ATTACHMENT I
TO
JPN-88-023

PROPOSED ADMINISTRATIVE CHANGES
TO THE
TECHNICAL SPECIFICATIONS
(JPTS-86-004)

NEW YORK POWER AUTHORITY
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
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TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Reportable Event - A reportable event shall be any of those conditions specified in Section 50.73 to 10 CFR Part 50.

B. Core Alteration - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation is not defined as a core alteration.

1.0 (cont'd)

C. Cold Condition - Reactor coolant temperature $\leq$212°F.

D. Hot Standby Condition - Hot Standby condition means operation with coolant temperature $>$ 212°F, the Mode Switch in Startup/Hot Standby and reactor pressure $<$ 1,005 psig.

E. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

F. Instrumentation

1. Functional Test - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

2. Instrument Channel Calibration - An instrument channel calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument channel including actuation, alarm or trip.

3. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

4. Instrument Check - An instrument check is a qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

5. Instrument Channel Functional Test - An instrument channel functional test means the injection of a simulated signal into the instrument primary sensor where possible to verify the proper instrument channel response, alarm and/or initiating action.

6. Logic System Function Test - A logic system functional test means a test of relays and contacts of a logic circuit from sensor to activated device to ensure components are operable per design intent. Where practicable, action will go to completion: i.e., pumps

1.0 (cont'd)

1. Refuel Mode - The reactor is in the refuel mode when the Mode Switch is in the Refuel Mode position. When the Mode Switch is in the Refuel position, the refueling interlocks are in service.

2. Run Mode - In this mode the reactor system pressure is at or above 850 psig and the Reactor Protection System is energized with APRM protection (excluding the 15 percent high flux trip) and the RBM interlocks in service.

3. Shutdown Mode - The reactor is in the shutdown mode when the Reactor Mode Switch is in the Shutdown Mode position.

   a. Hot shutdown means conditions as above with reactor coolant temperature $>$ 212°F.

   b. Cold shutdown means conditions as above with reactor coolant temperature $\leq$ 212°F. and the reactor vessel vented.

4. Startup/Hot Standby - In this mode the reactor protection scram trip initiated by main steam line isolation valve closure is bypassed when reactor pressure is less than 1,005 psig, the low pressure main steam line isolation valve closure trip is bypassed, the Reactor Protection System is energized with APRM (15 percent) and IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

J. Operable - A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

K. Operating - Operating means that a system or component is performing its intended functions in its required manner.

L. Operating Cycle - Interval between the end of one refueling outage and the end of the subsequent refueling outage.

M. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

   1. All manual containment isolation valves on lines connected to the Reactor Coolant System or containment which are not required to be open during plant accident conditions are closed. These valves may be

1.0 (cont'd)

opened to perform necessary operational activities.

2. At least one door in each airlock is closed and sealed.

3. All automatic containment isolation valves are operable or de-activated in the isolated position.

4. All blind flanges and manways are closed.

N. <u>Rated Power</u> - Rated power refers to operation at a reactor power of 2,436 MWt. This is also termed 100 percent power and is the maximum power level authorized by the operating license. Rated steam flow, rated coolant flow, rated nuclear system pressure, refer to the values of these parameters when the reactor is at rated power.

O. <u>Reactor Power Operation</u> - Reactor power operation is any operation with the Mode Switch in the Startup/Hot Standby or Run position with the reactor critical and above 1 percent rated thermal power.

P. <u>Reactor Vessel Pressure</u> - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space sensor.

Q. <u>Refueling Outage</u> - Refueling outage is the period of time between the shutdown of the unit prior to refueling and the startup of the Plant subsequent to that refueling.

R. <u>Safety Limits</u> - The safety limits are limits within which the reasonable maintenance of the fuel cladding integrity and the reactor coolant system integrity are assured. Violation of such a limit is cause for unit shutdown and review by the Atomic Energy Commission before resumption of unit operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

S. <u>Secondary Containment Integrity</u> - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The Standby Gas Treatment System is operable.

3. All automatic ventilation system isolation valves are operable or secured in the isolated position.

T. <u>Surveillance Frequency</u> - Periodic surveillance tests, checks, calibrations, and examinations shall be performed within the specified surveillance intervals. These intervals may be adjusted ± 25 percent. The interval as pertaining to instrument and electric surveillance shall never exceed one operating cycle. In cases where the elapsed interval has exceeded 100 percent of the specified interval, the next surveillance interval shall commence at the end of the original specified interval.

U. Thermal Parameters

1. Minimum critical power ratio (MCPR)-Ratio of that power in a fuel assembly which is calculated to cause some point in that fuel assembly to experience boiling transition to the actual assembly operating power as calculated by application of the GEXL correlation (Reference NEDE-10958).

2. Fraction of Limiting Power Density - The ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR. The design LHGR is 14.4 KW/ft for GE8x8EB fuel and 13.4 KW/ft for the remainder.

3. Maximum Fraction of Limiting Power Density - The Maximum Fraction of Limiting Power Density (MFLPD) is the highest value existing in the core of the Fraction of Limiting Power Density (FLPD).

4. Transition Boiling - Transition boiling means the boiling region between nucleate and film boiling. Transition boiling is the region in which both nucleate and film boiling occur intermittently with neither type being completely stable.

V. Electrically Disarmed Control Rod

To disarm a rod drive electrically, the four amphenol type plug connectors are removed from the drive insert and withdrawal solenoids rendering the rod incapable of withdrawal. This procedure is equivalent to valving out the drive and is preferred. Electrical disarming does not eliminate position indication.

W. High Pressure Water Fire Protection System

The High Pressure Water Fire Protection System consists of: a water source and pumps; and distribution system piping with associated post indicator valves (isolation valves). Such valves include the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler or water spray subsystem.

X. Staggered Test Basis

A Staggered Test Basis shall consist of:

a. A test schedule for "n" systems, subsystems, trains or other designated components obtained by dividing the specified test interval into "n" equal subintervals.

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

Y. Rated Recirculation Flow

That drive flow which produces a core flow of 77.0×10^6 lb/hr.

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:

$S \leq 0.66\ W + 54\%$ for two loop operation

or:

$S \leq (0.66\ W + 54\% - 0.66\Delta W)$ for single loop operation

where:

S = Setting in percent of rated thermal power (2436 MWT)

W = Recirculation flow in percent of rated

ΔW= Difference between two loop and single loop effective drive flow at the same core flow. ($\Delta W = 0$ for two loop operation. ΔW for single loop operation is to be determined upon implementation of single loop operation.)

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.

3.0 BASES

A. This specification states the applicability of each specification in terms of defined OPERATIONAL CONDITION (mode) and is provided to delineate specifically when each specification is applicable.

B. This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

C. This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. Under the terms of Specification 3.0, the facility is to be placed in COLD SHUTDOWN within the following 24 hours. It is assumed that the unit is brought to the required OPERATIONAL CONDITION (mode) within the required times by promptly initiating and carrying out the appropriate ACTION statement.

D. This specification provides that entry into an OPERABLE CONDITION (mode) must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

D. Continued

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other limits being exceeded.

Exceptions to this provision may be made for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

E. This specification delineates what additional conditions must be satisfied to permit operation to continue, consistent with the ACTION statements for power sources, when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the ACTION statements associated with individual systems, subsystems, trains, components or devices to be consistent with the ACTION statement of the associated electrical power source. It allows operation to be governed by the time

3.1 BASES (cont'd)

is discharged from the reactor by a scram can be accommodated in the discharge piping. Each scram discharge instrument volume accommodates in excess of 34 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level detection instruments have been provided in each instrument volume which alarm and scram the reactor when the volume of water reaches 34.5 gallons. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A Source Range Monitor (SRM) System is also provided to supply additional neutron level information during startup but has no scram functions (reference paragraph 7.5.4 FSAR).

The IRM high flux and APRM $\leq$ 15% power scrams provide adequate coverage in the startup and intermediate range. Thus, the IRM and APRM systems are required to be operable in the refuel and startup/hot standby modes. The APRM $\leq$120% power and flow referenced scrams provide required protection in the power range (referemce FSAR Section 7.5.7). The power range is covered only by the APRMs. Thus, the IRM system is not required in the run mode.

The high reactor pressure, high drywell pressure, reactor low water level and scram discharge volume high level scrams are required for startup and run modes of plant operation. They are, therefore, required to be operational for these modes of reactor operation.

The requirement to have the scram functions indicated in Table 3.1-1 operable in the refuel mode assures that shifting to the refuel mode during reactor power operation does not diminish the protection provided by the Reactor Protection System.

Turbine stop valve closure occurs at 10 percent of valve closure. Below 217 psig turbine first stage pressure (30 percent of rated), the scram signal due to turbine stop valve closure is bypassed because the flux and pressure scrams are adequate to protect the reactor.

4.1 BASES (cont'd)

The bi-stable trip circuit which is a part of the Group (B) devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in the Group (B) devices to calculate their unsafe failure rates. The non-ATTS (Analog Transmitter Trip System) analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than $20x10^{-6}$ failures/hr. The non-ATTS bi-stable trip circuits are predicted to have unsafe failure rate of less than $2x10^{-6}$ failures/hr. The ATTS analog devices (sensors), bi-stable devices (master and slave trip units) and power supplies have been evaluated for reliability by Mean Time Between Failure analysis or state-of-the-art qualification type testing meeting the requirements of IEEE 323-1974. Considering the 2-hour monitoring interval for analog devices as assumed above, the instrument checks and functional tests as well as the analyses and/or qualification type testing of the devices, the design reliability goal for system reliability of 0.9999 will be attained with ample margin.

The bi-stable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1-1. There are numerous identical bi-stable devices used throughout the Plant's instrumentation system. Therefore, significant data on the failure rates for the bi-stable devices should be accumulated rapidly.

The frequency of calibration of the APRM flow biasing network has been established as each refueling outage. The flow biasing network is functionally tested at least once/month and, in addition, cross calibration checks of the flow input to the flow biasing network can be made during the functional test by direct meter reading. There are several instruments which must be calibrated and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRM's resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instruments, such as those in the flow biasing network, is not significant and therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable: Refuel (6) (16)	Startup	Run	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
1	Mode Switch in Shutdown		X	X	X	1 Mode Switch (4 Sections)	A
1	Manual Scram		X	X	X	2 Instrument Channels	A
3	IRM High Flux	$\leq$120/125 of full scale	X	X		8 Instrument Channels	A
3	IRM Inoperative		X	X		8 Instrument Channels	A
2	APRM Neutron Flux-Startup (15)	$\leq$15% Power	X	X		6 Instrument Channels	A
2	APRM Flow Referenced Neutron Flux (Not to exceed 117%) (12)(13)(14)(17)	S$\leq$(0.66W+54%)(FRP/MFLPD)			X	6 Instrument Channels	A or B
2	APRM Fixed High Neutron Flux (14)	$\leq$120% Power			X	6 Instrument Channels	A or B
2	APRM Inoperative	(10)	X	X	X	6 Instrument Channels	A or B

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TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
			Refuel (6) (16)	Startup	Run		
2	Turbine Control Valve Fast Closure	$500 < P < 850$ psig Control oil pressure between fast closure solenoid and disc dump valve			X(4)	4 Instrument Channels	A or C

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TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting	Modes in Which Function Must be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
			Refuel (6))16)	Startup	Run		
4	Turbine Stop Valve Closure	≤ 10% valve closure			X(4)(5)	8 Instrument Channels	A or C

NOTES OF TABLE 3.1-1

1. There shall be two operable or tripped trip systems for each function, except as specified in 4.1.D. From and after the time that the minimum number of operable instrument channel for a trip system cannot be met, that affected trip system shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   A. Initiate insertion of operable rods and complete insertion of all operable rods within four hours.

   B. Reduce power level to IRM range and place Mode Switch in the Startup Position within eight hours.

   C. Reduce power to less than 30 percent of rated.

2. Permissible to bypass, if Refuel and Shutdown positions of the Reactor Mode Switch.

3. By passed when reactor pressure is less than 1005 psig.

4. Bypassed when turbine first stage pressure is less than 217 psig or less than 30 percent of rated.

5. The design permits closure of any two lines without a scram being initiated.

6. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need to be operable:
   A. Mode Switch in Shutdown
   B. Manual Scram

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3.2 BASES (cont'd)

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop accident. With the established setting of 3 times normal background, and main steam line isolation valve closure, fission product release is limited so that 10 CFR 100 guidelines are not exceeded for this accident. Reference Section 14.6.1.2 FSAR. During the Hydrogen Addition Test, the normal background Main Steam Line Radiation Level is expected to increase by approximately a factor of 5 at the peak hydrogen concentration as indicated in note 16, Table 3.1-1. With the hydrogen addition, the fission product release would still be well within the 10 CFR 100 guidelines in the event of a control rod drop accident.

Pressure instrumentation is provided to close the main steam isolation valves in the run mode when the main steam line pressure drops below 825 psig. The reactor pressure vessel thermal transient due to an inadvertent opening of the turbine bypass valves when not in the run mode is less severe than the loss of feedwater analyzed in Section 14.5 of the FSAR, therefore, closure of the main steam isolation valves for thermal transient protection when not in the run mode is not required.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

The trip settings of $\leq$300 percent of design flow for high flow or 40°F above maximum ambient for high temperature are such that uncovering the core is prevented and fission product release is within limits.

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip settings of $\leq$300 percent for high flow or 40°F above maximum ambient for temperature are based on the same criteria as the HPCI.

The reactor water cleanup system high temperature instrumentation are arranged similar to that for the HPCI. The trip settings are such that uncovering the core is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not decrease to the Safety Limit. The trip

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3.2 BASES (cont'd)

logic for this function is 1 out of n: e.g., any trip on one of six APRM's, eight IRM's, or four SRM's will result in a rod block.

The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This time period is only three percent of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence.

The RBM rod block function provides local protection of the core; i.e., the prevention of boiling transition in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trips are set so that MCPR is maintained greater than the Safety Limit.

The IRM rod block function provides local as well as gross core protection.

The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus, control rod motion is prevented. The downscale trips are set at 2.5 indicated on scale.

The flow comparator and scram discharge volume high level components have only one logic channel and are not required for safety. The flow comparator must be bypassed when operating with one recirculation water pump.

The refueling interlocks also operate one logic channel, and are required for safety only when the Mode Switch is in the Refueling position.

For effective emergency core cooling for small pipe breaks, the HPCI system must function since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The Automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in

4.2 BASES

The instrumentation listed in Table 4.2-1 through 4.2-6 will be functionally tested and calibrated at regularly scheduled intervals. The same design reliability goal as the Reactor Protection System is generally applied. Sensors, trip devices and power supplies are tested, calibrated and checked at the same frequency as comparable devices in the Reactor Protection System.

Those instruments which, when tripped, result in a rod block have their contacts arranged in a 1 out of n logic, and all are capable of being bypassed. For such a tripping arrangement with bypass capability provided, there is an optimum test interval that should be maintained in order to maximize the reliability of a given channel (7). This takes account of the fact that testing degrades reliability and the optimum interval between tests is approximately given by:

$$i = \sqrt{\frac{2t}{r}}$$

Where: i = the optimum interval between tests.

t = the time the trip contacts are disabled from performing their function while the test is in progress.

r = the expected failure rate of the relays.

To test the trip relays requires that the channel be bypassed, the test made, and the system returned to its initial state. It is assumed this task requires an estimated 30 min. to complete in a thorough and workmanlike manner and that the relays have a failure rate of 10^{-6} failures per hr. Using this data and the above operation, the optimum test interval is:

$$i = \sqrt{\frac{2(0.5)}{10^{-6}}} = 1 \times 10^3 \text{ hr.} = 40 \text{ days}$$

For additional margin a test interval of once/month will be used initially.

The sensors and electronic apparatus have not been included here as these are analog devices with readouts in the control room and the sensors and electronic apparatus can be checked by comparison with other like instruments. The checks which are made on a daily basis are adequate to assure operability of the sensors and electronic apparatus, and the test interval given above provides for optimum testing of the relay circuits.

TABLE 3.2-2 (cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Item No.	Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Remarks
9	1	Reactor Low Pressure	$50 \leq p \leq 75$ psig	2 Inst. Channels	In conjunction with PCIS signal permits closure of RHR (LPCI) injection valves.
10		THIS ITEM INTENTIONALLY BLANK			
11		THIS ITEM INTENTIONALLY BLANK			
12	1 (See Note 3)	Core Spray Pump Start Timer (each loop)	11 ± 0.6 sec.	1 Inst. Channel	Initiates starting of core spray pumps. (each loop)

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TABLE 3.2-2 (cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Item No.	Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Remarks
22		THIS ITEM INTENTIONALLY BLANK			
23		THIS ITEM INTENTIONALLY BLANK			
24		THIS ITEM INTENTIONALLY BLANK			
25	1	Core Spray Sparger to Reactor Pressure vessel d/p	$\leq$0.5 psid	2 Inst. Channels	Alarm to detect core spray sparger pipe break.
26	2	Condensate storage Tank Low Level	$\geq$59.5 in. above tank bottom (=15,600 gal avail)	2 Inst. Channels	Provides interlock to HPCI suction valves.
27	2	Suppression Chamber High Level	$\leq$6 in. above normal level	2 Inst. Channels	Transfers HPCI pump suction to suppression chamber.
28	1	RCIC Turbine Steam Line High Flow	$\leq$282 in. H_2O dp	2 Inst. Channels	Close Isolation Valves in RCIC Subsystem.

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TABLE 3.2-2 (cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Item No.	Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Trip Systems	Remarks
29	1	RCIC Steam Line/ Area Temperature	$\leq$40°F Above max. ambient	2 Inst. Channels	Close Isolation Valve in RCIC Subsystem
30	1	RCIC Steam Line Low Pressure	$100 > P > 50$ psig	2 Inst. Channels	Close Isolation Valves in RCIC Subsystem
31	1	HPCI Turbine Steam Line High Flow	$\leq$106 in H_2O dp	2 Inst. Channels	Close Isolation Valves in HPCI Subsystem
32	1	RCIC Turbine High Exhaust Diaphragm Pressure	$\leq$10 psig	2 Inst. Channels	Close Isolation Valves in RCIC Subsystem
33	1	HPCI Turbine High Exhaust Diaphragm Pressure	$\leq$10 psig	2 Inst. Channels	Close Isolation Valves in HPCI Subsystem
34	1	LPCI Cross-Connect Position	NA	1 Inst. Channels	Initiates annunciation when valve is not closed
35	1	HPCI Steam Line Low Pressure	$100 > P > 50$ psig	2 Inst. Channels	Close Isolation Valve in HPCI Subsystems
36	1	HPCI Steam Line/ Area Temperature	$\leq$40°F above max. ambient	2 Inst. Channels	Close Isolation Valve in HPCI Subsystem

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NOTES FOR TABLE 3.2-6 (CONTINUED)

2. In the event that all indications of this parameter is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in six (6) hours and a Cold Shutdown condition in the following eighteen (18) hours.

3. Three (3) indicators from level instrument channel A, B, & C. Channel A or B are utilized for feedwater control, reactor water high and low level alarms, recirculation pump runback. High level trip of main turbine and feedwater pump turbine utilizes channel A, B, & C.

4. One (1) recorder utilized the same level instrument channel as selected for feedwater control.

5. Three (3) indicators from reactor pressure instrument channel A, B, & C. Channel A or B are utilized for feedwater control and reactor pressure high alarm.

6. One (1) recorder. Utilizes the same reactor pressure instrument channel as selected for feedwater control.

7. The position of each of the 137 control rods is monitored by the Rod Position Information System. For control rods in which the position is unknown, refer to Paragraph 3.3.A.

8. Neutron monitoring operability requirements are specified by Table 3.1-1 and Paragraph 3.3.B.4.

9. A minimum of 3 IRM or 2 APRM channels respectively must be operable (or tripped) in each safety system.

10. Each Safety Relief Valve is equipped with two acoustical detectors of which one is in service and a backup thermocouple detector. In the event that a thermocouple is inoperable SRV performance shall be monitored daily with the associated acoustical detector.

11. From and after the date that none of the acoustical detectors is operable but the thermocouple is operable, continued operation is permissible until the next outage in which a primary containment entry is made. Both acoustical detectors shall be made operable prior to restart.

12. In the event that both primary and secondary indications of this parameter for any one valve are disabled and neither indication can be restored in forty-eight (48) hours, an orderly shutdown shall be initiated and the reactor shall be in a Hot Shutdown condition in twelve (12) hours and in a Cold Shutdown within the next twenty-four (24) hours.

13. From and after the date that the minimum number of operable instrument channels is one less than the minimum number specified for each parameter, continued operation is permissible during the succeeding 7 days unless the minimum number specified is made operable sooner.

TABLE 3.2-7

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minimum Number of Operable Instrument Channels per trip System (1)	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Action
1	Reactor High Pressure	$\leq$1120 psig	4	(2)
1	Reactor Low-Low Water Level	$\geq$ -38 in. indicated level ($\geq$126.5 in. above the top of active fuel)	4	(2)

Notes for Table 3.2-7

1. Whenever the reactor is in the run mode, there shall be one operable trip system for each parameter for each operating recirculation pump. From and after the time it is found that this cannot be met, the indicated action shall be taken.

2. Reduce power and place the Mode Selector Switch in a Mode other than the Run Mode within 24 hours.

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3.3 (cont'd)

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

d. Control rods with a failed "Full-in" or "Full-out" position switch may be bypassed in the Rod Sequence Control System and considered operable if the actual rod position is known. These rods must be moved in sequence to their correct positions (full in on insertion and full out on withdrawal).

e. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be inoperable.

f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. In addition, during reator power operation, no more than one control rod in any 5 X 5 array may be inoperable (at least 4 operable control rods must separate any 2 inoperable ones). If this specification cannot be met the reactor shall not be started, or if at power, the reactor shall be brought to a cold condition within 24 hr.

4.3 (cont'd)

e. When it is initially determined that a control rod is incapable of normal insertion, an attempt to fully insert the control rod shall be made. If the control rod cannot be fully inserted, a shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined, highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted. If Specification 3.3.A.1 and 4.3.A.1 are met, reactor startup may proceed.

f. The scram discharge volume drain and vent valves shall be full-travel cycled at least once per quarter to verify that the valves close in less than 30 seconds and to assure proper valve stroke and operation.

g. At least once per operating cycle, the operability of the entire scram discharge system as an integrated whole shall be demonstrated by a scram of control rods from a normal control rod configuration of less than or equal to 50% rod density by verifying that the drain and vent valves:

1. Close upon receipt of a signal for control rods to scram and:

2. Open when the scram signal is reset.

This requirement may be satisfied as part of any scram originating from the rod density conditions specified above, provided that Specification 4.3.A.2.f is independently satisfied during the quarter in which the scram occurs.

3.3 (cont'd)

B. Control Rods

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be in place during reactor power operation or

4.3 (cont'd)

B. Control Rods

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

a. When a rod is withdrawn the first time after each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is above 20 percent power shall be performed to verify instrumentation response.

b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

c. During each refueling outage and after each control rod maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

3.3 and 4.3 BASES (cont'd)

At power levels below 20% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calories per gram drop limit. In this range, the RWM and RSCS constrain the control rod sequence and patterns to those which involve only acceptable rod worths.

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviance from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences which limit the maximal reactivity worth of control rods, in the event that the Rod Worth Minimizer is out of service, when required, a second licensed operator or other qualified technical plant employee

can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural control to assure conformance.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a control rod drop. At low powers, below 20%, these devices force adherence to acceptable rod patterns. Above 20% of rated power, no constraint on rod pattern is required to assure that rod drop accident consequences are acceptable. Control rod pattern constraints above 20% of rated power are imposed by power distribution requirements as defined in Section 3.3.3.5 of these Technical Specifications. Power level for automatic cutout of the RSCS function is sensed by first stage turbine pressure. Because the instrument has an instrument error of $\pm$ 2% of full power, the nominal instrument setting is 22% of rated power. Power level for automatic cutout of the RWM function is sensed by steam flow and is set manually at 30% of rated power to be consistent with the RCSC setting.

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 20% rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

The RSCS can be functionally tested prior to control rod withdrawal for reactor startup. By selecting, for example, $^A12$ and attempting to withdraw, by one notch, a rod or all rods in each other group, it can be determined that the $^A12$ group is exclusive. By bypassing to full-out all $^A12$ rods, selecting $^A34$ and attempting to withdraw, by one notch, a rod or all rods in group B, the $^A34$ group is determined exclusive. The same procedure can be repeated for the B groups. After 50% of the control

B. Operation with Inoperable Components

From and after the date that a redundant component is made or found to be inoperable, Specification 3.4.A shall be considered fulfilled, and continued operation permitted, provided that:

1. The component is returned to an operable condition within 7 days.

4.4 (Cont'd)

pump solution in the recirculation path.

Explode one of three primer assemblies manufactured in same batch to verify proper function. Then install the two remaining primer assemblies of the same batch in the explosive valves.

Demineralized water shall be injected into the reactor vessel to test that valves (except explosive valves) not checked by the recirculation test are not clogged.

Test that the setting of the system pressure relief valves is between 1,400 and 1,490 psig.

3. Disassemble and inspect one explosive valve so that it can be established that the valve is not clogged. Both valves shall be inspected in the course of two operating cycles.

B. Operation with Inoperable Components

When a component becomes inoperable its redundant component shall be demonstrated to be operable immediately and daily thereafter.

3.5 (cont'd)

2. From and after the date that one of the Core Spray Systems is made or found inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the system is made operable earlier, provided that during the 7 days all active components of the other Core Spray System and the LPCI System shall be operable.

3. The LPCI mode of the RHR System shall be operable whenever irradiated fuel is in the reactor and prior to reactor startup from a cold condition, except as specified below.

   a. From the time that one of the RHR pumps is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the pump is made operable earlier provided that during such 7 days the remaining active components of the LPCI, containment spray mode, and all active components of both Core Spray Systems are operable.

4.5 (cont'd)

2. When it is determined that one Core Spray System is inoperable, the operable Core Spray System, and the LPCI System, shall be demonstrated to be operable immediately. The remaining Core Spray System shall be demonstrated to be operable daily thereafter.

3. LPCI System testing shall be as specified in 4.5.A.1.a, b, c, d, f and g except that three RHR pumps shall deliver at least 23,100 gpm against a system head corresponding to a reactor vessel pressure of 20 psig.

   a. When it is determined that one of the RHR pumps is inoperable, the remaining active components of the LPCI, containment spray subsystem and both Core Spray Systems required for operation shall be demonstrated to be operable immediately, and the remaining RHR pumps shall be demonstrated to be operable daily thereafter.

3.5 (cont'd)

5. All recirculation pump discharge valves shall be operable prior to reactor startup (or closed if permitted elsewhere in these specifications).

6. If the requirements of 3.5.A cannot be met, the reactor shall be placed in the cold condition within 24 hrs.

B. Containment Cooling Subsystem Mode (of the PHR System)

1. Both subsystems of the containment cooling mode, each including two RHR, one ESW pump and two RHRSW pumps shall be operable whenever there is irradiated fuel in tne reactor vessel, prior to startup from a cold condition, and reactor coolant temperature $\geq$ 212°F except as specified below:

2. Continued reactor operation is permissible for 30 days with one spray loop inoperable and with reactor water temperature greater than 212°F.

4.5 (cont'd)

5. All recirculation pump discharge valves shall be tested for operability any time the reactor is in the cold condition exceeding 48 hours, if operability tests have not been performed during the preceding 31 days.

B. Containment Cooling Subsystem Mode (of the PHR System)

1. Subsystems of the containment cooling mode are tested in conjunction with the test performed on the LPCI System and given in 4.5.A.1.a, b, c, and d. Residual heat removal service water pumps, each loop consisting of two pumps operating in parallel, will be included in testing, supplying 8,000 gpm. The Emergency Service Water System, each loop of which consists of a single operating emergency service water pump will be tested in accordance with Section 4.11D.

   During each five-year period, an air test shall be performed on the containment spray headers and nozzles.

2. When it is determined that one RHR pump and/or one RHRSW pump of the components required in 3.5.B.1 above are inoperable, the remaining redundant active components of the containment cooling mode subsystems shall be demonstrated to be operable immediately and daily thereafter.

3.5 (cont'd)

a. From and after the date that the HPCI System is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 7 days unless such system is sooner made operable, provided that during such 7 days all active components of the Automatic Depressurization System, the Core Spray System, LPCI System, and Reactor Core Isolation Cooling System are operable.

b. If the requirements of 3.5.C.1 cannot be met, the reactor shall be placed in the cold condition and pressure less than 150 psig within 24 hrs.

2. Low power physics testing and reactor operator training shall be permitted with reactor coolant temperature $\leq$212°F with an inoperable component(s) as specified in 3.5.C.1 above.

4.5 (cont'd)

a. When it is determined that the HPCI subsystem is inoperable the RCIC, the LPCI subsystem, both core spray subsystems, and the ADS subsystem actuation logic shall be demonstrated to be operable immediately. The RCIC system and ADS subsystem logic shall be demonstrated to be operable daily thereafter.

3.5 (cont'd)

D. Automatic Depressurization System (ADS)

1. The ADS shall be operable whenever the reactor pressure is greater than 100 psig, and irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition, except as specified below:

   a. From and after the date that one of the seven safety/relief valves of the ADS is made or found to be inoperable for any reason while it is required, continued reactor operation is permissible only during the succeeding 30 days unless repairs are made and provided that during such time the HPCI System is operable.

   b. From the time that more than one of the seven safety/relief valves of the ADS are made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 24 hrs. unless repairs are made and provided, that

4.5 (cont'd)

D. Automatic Depressurization System (ADS)

1. Surveillance of the Automatic Depressurization System shall be performed during each operating cycle as follows:

   a. A simulated automatic initiation which opens all pilot valves.

   b. Manually open each safety/relief valve while bypassing steam to the condenser and observe a $\geq$10% closure of the turbine bypass valves, to verify that the safety/relief valve has opened.

   c. A simulated automatic initiation which is inhibited by the override switches.

3.5 (Cont'd)

F. Minimum Emergency Core and Containment Cooling System Availability

1. Any combination of inoperable components in the Core and Containment Cooling Systems shall not defeat the capability of the remaining operable components to fulfill the core and containment cooling functions.

2. When the irradiated fuel is in the reactor vessel and the reactor is in the cold condition all LPCI, core spray, and containment cooling subsystems may be inoperable provided no work is being done which has the potential for draining the reactor vessel.

G. Maintenance of Filled Discharge Pipe

Whenever core spray subsystems, LPCI subsystems, HPCI, or RCIC are required to be operable, the discharge piping from the pump discharge of these systems to the last block valve shall be filled.

a. From and after the time that the pump discharge piping of the HPCI, RCIC, LPCI, or Core Spray Systems cannot be maintained in a filled

4.5 (Cont'd)

F. Minimum Emergency Core and Containment Cooling System Availability

Not Applicable.

G. Maintenance of Filled Discharge Pipe

The following surveillance requirements shall be adhered to, in order to assure that the discharge piping of the core spray subsystem, LPCI subsystem, HPCI, and RCIC are filled:

1. Every month prior to the testing of the LPCI subsystem and core spray subsystem, the discharge piping of these systems shall be vented from the high point, and water flow observed.

3.5 (cont'd)

condition, that pump shall be considered inoperable for purposes 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. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) shown in Figures 3.5-10 through 3.5-12 during two recirculation loop operation. During single loop operation, the APLHGR for each fuel type shall not exceed the above values multiplied by 0.84 (see Bases 3.5.K, Reference 1). 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 required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI, RCIC, or Core Spray System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI, RCIC, and Core Spray 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 (cont'd)

E. Safety and Safety/Relief Valves

1. During reactor power operating conditions and prior to startup from a cold condition, or whenever reactor coolant pressure is greater than atmosphere and temperature greater than 212°F, the safety mode of all safety/relief valves shall be operable, except as specified by Specification 3.6.E.2. The Automatic Depressurization System valves shall be operable as required by specification 3.5.D.

4.6 (cont'd)

E. Safety and Safety/Relief Valves

1. At least one half of all safety/relief valves shall be bench checked or replaced with bench checked valves once each operating cycle. The safety/relief valve settings shall be set as required in Specification 2.2.B. All valves shall be tested every two operating cycles.

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(cont'd)

2. a. From and after the date that the safety valve function of one safety/relief valve is made or found to be inoperable, continued operation is permissible only during the succeeding 30 days unless such valve is sooner made operable.

   b. From and after the time that the safety valve function on two safety/relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding 7 days unless such valves are sooner made operable.

3. If Specification 3.6.E.1 and 3.6.E.2 are not met, the reactor shall be placed in a cold condition within 24 hr.

4. Low power physics testing and reactor operator training shall be permitted with inoperable components as specified in Item B.2 above, provided that reactor coolant temperature is $\leq$ 212°F and the reactor vessel is vented or the reactor vessel head is removed.

4.6 (cont'd)

2. At least one safety/relief valve shall be disassembled and inspected once/operating cycle.

3. The integrity of the safety/relief valve bellows shall be continuously monitored.

   a. The bellows monitoring pressure switches shall be removed and bench checked once/operating cycle. Modified safety/relief valves with two-stage assemblies do not have a bellows arrangement and are, therefore, not subject to this requirement.

4. The integrity of the nitrogen system and components which provide manual and ADS actuation of the safety/relief valves shall be demonstrated at least once every 3 months.

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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 inservice inspection program shall consist of 100 percent inspection of these welds per inspection interval.

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)

E. Safety and Safety/Relief Valves

Experiences in safety valve operation show that the testing of 50 percent of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code an $\pm$ 1 percent of design pressure. An analysis has been performed which shows that with all safety valves set 1 percent higher, the reactor coolant pressure safety limit of 1,375 psig is not exceeded.

The safety/relief valves have two functions; i.e., power relief or self-actuated by high pressure. Power relief is a solenoid actuated function (Automatic Depressurization System) in which external instrumentation signals of low-low-low water level initiate the valves to open. This function is discussed in specification B.3.5.D. In addition, the valves can be operated manually.

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9 of the ASME Pressure Vessel Code Section III-Nuclear Vessels, requires that these bellows be monitored for failure, since this would defeat the safety function of the safety/relief valve.

The modified version of the safety/relief valves function with a direct-acting pilot arrangement with no integral bellows.

It is realized that there is no way to repair or replace the bellows during operation, and the plant must be shutdown to do this. The 30-day and 7-day periods to do this allow the operator flexibility to choose his time for shutdown; meanwhile, because of the redundancy present in the design and the continuing monitoring of the integrity of the other valves, the overpressure pressure protection has not been compromised in either case. The auto-relief function would not be impaired by a failure of the bellows. However, the self-actuated overpressure safety function would be impaired by such a failure. There is no provision for testing the bellows leakage pressure switch during plant operation. The bellows leakage pressure switches will be removed and bench checked once/operating cycle. These bench checks provide adequate assurance of bellows integrity. For those modified safety/relief valves with the direct-acting pilot arrangement, bellows failures and bellows related calibrations do not apply.

Low power physics testing and reactor operator training with inoperable components will be conducted only when the safety/relief and safety valves are

3.6 and 4.6 BASES (cont'd)

not required to be operable (reactor coolant temperature less than or equal to 212°F and the reactor vessel vented or the reactor vessel head removed). Permitting physics testing and operator training under these conditions would not place the plant in an unsafe condition.

F. Structural Integrity

A pre-service inspection of the ASME Code Class 1 components was performed after site erection to assure the system was free of gross defects. An initial inspection program as detailed in Appendix F of the FSAR was developed and based on an approved edition of the ASME Code.

The program has been expanded to include the requirements of later, approved ASME Code editions and addenda as far as practicable. The importance of these inspections is recognized, and efforts to develop practical new alternative methods of assuring plant inservice integrity will continue. This inspection program should assure the detection of problem areas well before they represent a significant impact on safety.

Several locations on the main steam lines and feedwater lines are not restrained to prevent pipe whip in the event of pipe failure at these locations. The physical layout within the drywell precludes restraints at these points. Unrestrained high stress areas have been identified in these lines where breaks could result in pipe whip such that the pipe could impact the primary containment wall. Augmented inservice inspection of these weld locations shall be performed during each inspection period.

In addition, visual inspection in accordance with the approved ASME code will be made during periodic pressure and hydrostatic tests of critical systems. The inspection program specified encompasses the major areas of the vessel and piping system within the drywell. The inspection period is based on the observed rate of defect growth from fatigue studies sponsored by the AEC.

These studies show that thousand of stress cycles, at stresses beyond any expected to occur in a Reactor Coolant System, were required to propagate a crack. The test

4.7 (cont'd)

c. Type C tests

(1.) Type C tests shall be performed by local pressurization. The pressure shall be applied in the same direction as that when the valve would be required to perform its safety function, except as listed in Table 4.7-2 unless it can be determined that the results from the tests for a pressure applied in a different direction will provide equivalent or more conservative results. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments.

(2.) Valves, unless pressurized with fluid from a seal system, shall be pressurized with air or nitrogen at a pressure of Pa, and the gas flow to maintain Pa shall be measured.

(3.) Valves, which are sealed with fluid from a seal system, such as the liquid in the suppression chamber shall not be tested.

4.7 (cont'd)

(4.) See table 4.7-2 for exceptions.

(5.) Acceptance criterion - The combined leakage rate for all penetrations and valves subject to type B and C tests shall be less than 0.60 La. Leakage from containment isolation valves that are sealed with fluid from a seal system may be excluded when determining the combined leakage rate provided that the installed isolation valve seal-water system fluid inventory is sufficient to assure the sealing function for at least 30 days.

d. Other leak rate tests

(1) The leakage rate for containment isolation valves 10-AOV-68A, B (penetration X-13A, B) for Low Pressure Coolant Injection system and 14-AOV-13A, B (penetration X-16A, B) for Core Spray System shall be less than 11 cubic feet per minute per valve (pneumatically tested at 45 psig with ambient temperature) or 10 gallons per minute per valve (hydrostatically) tested at 1000 psig with ambient temperature.

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.

(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 pre-operational 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 (cont'd)

4. Pressure Suppression Chamber Reactor Building Vacuum Breakers

a. Except as specified in 3.7.A.4.b below, two Pressure Suppression Chamber Reactor Building Vacuum Breakers shall be operable at all times when the primary containment integrity is required. The setpoint of the differential pressure instrumentation which actuates the pressure suppression chamber reactor building vacuum breakers shall be less than or equal to 0.5 psid.

b. From and after the date that one of the pressure suppression chamber reactor building vacuum breakers is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 7 days, unless such vacuum

4.7 (cont'd)

When the primary containment is inerted, it shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. The monitoring system may be taken out of service for maintenance, but shall be returned to service as soon as possible.

4. Pressure Suppression Chamber - Reactor Building Vacuum Breakers

a. The pressure suppression chamber reactor building vacuum breakers and associated instrumentations including setpoint shall be checked for proper operation every three months.

3.7 (cont'd)

breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

5. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. When primary containment integrity is required, all drywell suppression chamber vacuum breakers shall be operable and positioned in the fully closed position except during testing and as specified in 3.7.A.5.b below.

b. One drywell suppression chamber vacuum breaker may be non-fully closed so long as it is determined to be not more than 1° open as indicated by the position lights.

c. One drywell suppression chamber vacuum breaker may be determined to be inoperable for opening.

d. If specifications 3.7.A.5.a, b, and c cannot be met, an orderly shutdown will be initiated, and the reactor shall be placed in a cold condition.

4.7 (cont'd)

5. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. Each drywell suppression chamber vacuum breaker shall be exercised through an opening - closing cycle monthly.

b. When it is determined that one vacuum breaker is inoperable for fully closing when operability is required, the operable breakers shall be exercised immediately, and every 15 days thereafter until the inoperable valve has been returned to normal service.

c. Once each operating cycle, each vacuum breaker valve shall be visually inspected to insure proper maintenance and operation.

d. A leak test of the drywell to suppression chamber structure shall be conducted once per operating cycle; the acceptable leak rate is 0.25 in. water/min, over a 10 min period, with the drywell at 1 psid.

3.7 (cont'd)

e. Leakage between the drywell and suppression chamber shall not exceed a rate of 71 scfm as monitored via the suppression chamber 10 min pressure transient of 0.25 in. water/min.

f. The self actuated vacuum breakers shall open when subjected to a force equivalent to 0.5 psid acting on the valve disc.

g. From and after the date that one of the pressure suppression chamber/drywell vacuum breakers is made or found to be inoperable for any reason, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding sever days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

4.7 (cont'd)

e. Not applicable

f. Not applicable

g. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.7.A.5.f and each vacuum breaker shall be inspected and verified to meet design requirements.

3.7 (cont'd)

2. In the event any isolation valve specified in Table 3.7-1 becomes inoperable, reactor power operation may continue, provided at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold condition within 24 hours.

4.7 (cont'd)

(2.) With the reactor at reduced power level, trip main steam isolation valves and verify closure time.

d. At least twice per week, the main steam line power-operated isolation valves shall be exercised by partial closure and subsequent reopening.

2. Whenever an isolation valve listed in Table 3.7-1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

4.7 BASES (cont'd)

assumption of no holdup in the secondary containment, resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide additional margin between expected offsite doses and 10CFR100 guidelines.

The maximum allowable test leak rate at the peak pressure of 45 psig (Pa) is 0.5 weight percent per day (Lam). The maximum allowable test leak rate at the reduced pressure of 23 psig (P_t) will be verified to be conservative by actual primary containment leak rate measurements at both 45 psig and 23 psig upon completion of the containment structure.

To allow a margin for possible leakage deterioration between intervals, the maximum allowable leak rate (Ltm), which will be met to remain on the normal test schedule, is 0.75 L_t. In addition, it is intended to operate the primary containment structure at a slight positive pressure to continuously monitor primary containment leakage.

As most leakage and deterioration of integrity is expected to occur through penetrations, especially those with resilient seals, a periodic leak rate test program of such penetrations is conducted at the peak pressure of 45 psig to insure not only that the leakage remains acceptably low but also that the sealing materials can withstand the accident pressure. For airlock leak test, a seal test at the peak pressure could be substituted for the complete airlock test, if no maintenance work is done which could affect the sealing capability of the airlock.

The leak rate testing program was originally based on Commission guidelines for development of leak rate testing and surveillance schedules for reactor containment vessels, (16) and discussed in Question 5.4 of the FSAR. With the exceptions listed in Table 4.7-2, the system conforms to the latest Commission guidelines (17). The exceptions stated in Table 4.7-2 are necessary since additional requirements were added after the system was designed.

B. <u>Standby Gas Treatment System and</u>
C. <u>Secondary Containment</u>

Initiating reactor building isolation and operation of the Standby Gas Treatment System to maintain at least a 1/4 in. of water vacuum within the secondary containment provides an adequate test of the operation of the reactor

Table 3.7-1 (Cont'd)
PROCESS PIPELINE PENETRATING PRIMARY CONTAINMENT
(Numbers in parentheses are keyed to numbers on following pages: signal codes are listed on following pages)

Line Isolated	Drywell Penetration	Valve Type (6)	Power to Open (5) (6)	Group	Location Ref. to Drywell	Power to Close (5) (6)	Isolation Signal	Closing Time (7)	Normal Status	Remarks and Exceptions
Core Spray, Minimum Pump Flow	X-210A,B	MO Gate	Ac	B	Outside	Ac	RM	Not applicable	Closed	
Core Spray to Reactor	X-16A,B	MO Gate	Ac	A	Outside	Ac	RM	Not applicable	Open	Note (10)
Core Spray to Reactor	X-16A,B	MO Gate	Ac	A	Outside	Ac	RM	Not applicable	Closed	Note (10)
Core Spray to Reactor	X-16A,B	AO Check	(3)	A	Inside	Note (3)	Rev. flow	Not applicable	Closed	Testable Check Valve Note (3,16)
Core Spray Test to Suppression Pool	X-210A,B	MO Globe	Ac	B	Outside	Ac	G,RM	45 Sec	Closed	
Core Spray Pump Suction	X-227A,B	MO Gate	Ac	B	Outside	Ac	RM	Not Applicable	Open	
Drywell Equipment Drain Sump Discharge	X-19	MO Plug	Ac	B	Inside	Ac	A,F,RM	30 Sec	Open	
Drywell Equipment Drain Sump Discharge	X-19	AO Plug	Air/Ac	B	Outside	Spring	A,F,RM	Not Applicable	Closed (17)	
Drywell Floor Drain Sump Discharge	X-18	MO Plug	Ac	B	Inside	Ac	A,F,RM	30 Sec	Open	
Drywell Floor Drain Sump Discharge	X-18	AO Plug	Air/Ac	B	Outside	Spring	A,F,RM	Not Applicable	Open	
Traveling Incore Probe	X-35A,B,C,D	Explosive Shear	Dc	A	Outside	Dc	RM	Not Applicable	Open	One valve on each line
Traveling Incore Probe	X-35A,B,C,D	SO Ball	Ac	A	Outside	Ac	A,F,RM	Not Applicable	Open	One valve on each line Note (14)
Traveling Incore Probe Purge	X-35E	SO Valve	Ac	A	Outside	Spring	A,F,RM	Not Applicable	Closed	
Traveling Incore Probe Purge	X-35E	Check	Fwd. Flow	A	Inside	Process	Rev. Flow	Not Applicable	Closed	
HPCI - Turbine Steam Supply	X-11	MO Gate	Ac	A	Inside	Ac	L,RM	20 Sec	Open	) Signal "G" opens) valve.
HPCI - Turbine Steam Supply	X-11	MO Gate	Dc	A	Outside	Dc	L,RM	20 Sec	Closed	) Signal "L") overrides and) closes valve.

Amendment No. ~~40~~, ~~48~~, ~~108~~

TABLE 4.7-2

EXCEPTION TO TYPE C TESTS

Certain Type C test will be performed or omitted as follows:

Penetration	System	Valve	Local Leak Rate Test Performed
X-7A, B, C, and D	Main Steam	29-AOV-80A, B C, and D 29-AOV-86A, B C, and D	These valves are air-operated globe valves - pressurized in reverse direction and measurement of leakage will be equivalent to results from pressure applied in the same direction as when the valves would be required to perform its safety function. Therefore, pressure will be applied between the isolation valves and leakage measured. A water seal of 25 psig will be used on the inboard valve to determine the outboard valve's leak rate. (limit 11.5 scfh at 25 psig (a)
X-10	RCIC	13-MOV-15	See X-25 (27-AOV-131A, B)
X-11	HPCI	23-MOV-15	See X-25 (27-AOV-131A, B)
X-25	Dry Well Inerting CAD and Purge	27-AOV-112	This valve is a butterfly valve - pressurization in reverse direction and measurement of leakage will be equivalent to results from pressure applied in the same direction as that when the valve would be required to perform its safety function.
X-25	Dry Well Inerting CAD and Purge	27-AOV-131A 27-AOV-131B	These valves will be tested in the reverse direction, since the system was not designed for pressure to be applied in the same direction as that when the valve would be required to performs its safety function. Basis - The pressurization direction was not a requirement at the time of plant designs; to redesign the system to permit this is not feasible as it would delay plant operation.
X-26 A/B	Dry Well Inerting CAD and Purge	27-AOV-113 27-MOV-113	See X-25 (27-AOV-112) This globe valve will be tested in the reverse direction. See X-25 (27-AOV-131A, B)

(a) During cycle 3 the plant may operate with valve 29-AOV-86A type C leakage not to exceed 300 SCFD and valves 29-AOV-86A, B, C and D total leakage not to exceed 1104 SCFD.

Amendment No. ~~40~~, ~~44~~

TABLE 4.7-2 (CONT'D)

Penetration	System	Valve	Local Leak Rate Test Performed
		27-SOV-120B 27-SOV-121B 27-SOV-122B	See X-25 (27-AOV-131A, B)
X-31 Bd	Dry Well Inerting CAD and Purge	27-SOV-125B	See X-25 (27-AOV-131A)
X-39A	Cont. Spray	10-MOV-31A	This valve will be pressurized in the reverse direction and leakage measured. See X-25 (27-SOV-131A, B)
X-39B	Cont. Spray	10-MOV-31A	See X-39A
X-45	ILRT	VSM-100T	See X-25 (27-AOV-131A, B)
X-59	Dry Well Inerting Cad and Purge	27-SOV-123A	See X-25 (27-AOV-131A, B)
X-202	Torus Vacuum Breakers	AOV-101A/B	See X-25 (27-AOV-112)
X-203A	Dry Well Inerting CAD and Purge	27-SOV-119B	See X-25 (27-AOV-131A, B)
X-203B	Dry Well Inerting CAD and Purge	27-SOV-124A	See X-25 (27-AOV-131A)
X-205	Dry Well Inerting CAD and Purge	27-AOV-117 27-MOV-117	See X-25 (27-AOV-112) See X-25 (27-MOV-113)
X-210 A/B	RCIC, RHR		Will not be tested as lines are water ...ed by suppression chamber water ...-25 (27-AOV-131A, B)
X-211A	RHR	10-MOV-38A	Th... valve will be tested in the reverse dir...ion. See X-25 (27-AOV-131A, B)
X-211B	RHR	10-MOV-38B	This valve will be tested in the reverse direction.
X-212	RCIC	13-MOV-130	See X-25 (27-AOV-131A/B)
X-218	ILRT	VSM-100T	See X-25 (27-AOV-131A/B)
X-220	Dry Well Inerting CAD and Purge	27-AOV-116 27-SOV-132A 27-SOV-132B	See X-25 (27-AOV-112) See X-25 (27-AOV-131A/B)
X-222	HPCI		See X-210 A/B
X-224	RHR		See X-210 A/B
X-225	RHR		See X-210 A/B

TABLE 4.7-2 (CONT'D)

Penetration	System	Valve	Local Leak Rate Test Performed
X-226	HPCI		See X-210 A/B
X-227	Core Spray		See X-210 A/B
X-228	Condensate		See X-210 A/B

3.8 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Source Leakage Test

Specification:

Radioactive sources shall be leak tested for contamination. The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample. If the test reveals the presence of 0.005 microcurie or more of removable contamination, the source shall be decontaminated, and repaired, or be disposed of in accordance with Commission regulations.

Those quantities of by-product material that exceed the quantities listed in 10 CFR 30.71 Schedule B are to be leak tested in accordance with the schedule shown in Surveillance Requirements. All other sources (including alpha emitters) containing greater than 0.1 microcuries are also to be leak tested in accordance with the Surveillance Requirements.

Bases

Ingestion or inhalation of source material may give rise to total body or organ irradiation. This specification assures that leakage from radioactive material sources does not exceed allowable limits. In the unlikely event that those quantities of radioactive by-product materials of interest to this specification which are exempt from leakage testing are ingested or inhaled, they represent less than one maximum permissible body burden for total body irradiation. The limits for all other sources (including alpha emitters) are based upon 10 CFR 70.39(c) limits for plutonium.

4.8 SURVEILLANCE REQUIREMENTS

Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

1. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than thirty days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date or use or transfer. In the absence of a certificate from a transferor indicating that a test has been within six months prior to the transfer, sealed source shall not be put into use until tested.

3. Startup sources shall be leak tested prior to and following any repair or maintenance and before being subjected to core flux.

3.9 Continued

2. The Diesel Fuel Oil Transfer System shall be operable whenever the diesel generator it supplies is required to be operable, except as specified below:

   a. From and after the time that one fuel oil transfer pump per Diesel Generator System is made or found to be inoperable for any reason, continued reactor operation is permissible for a period not to exceed 60 days; provided that the remaining fuel oil transfer pumps be demonstrated to be operable immediately and weekly thereafter.

   b. From and after the time that only two fuel oil transfer pumps per Diesel Generator System are operable, continued reactor opertion is permissible for a period not to exceed 30 days total per pair of diesels, provided that the remaining fuel oil transfer pumps are demonstrated to be operable and daily thereafter.

4.9 Continued

2. During the monthly diesel generator testing, the diesel fuel oil transfer systems shall be checked for proper operation.

4.9 BASES (cont'd)

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.

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.

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.

3.10 (cont'd)

control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

7. In the "refuel" mode, there are interlocks which prevent the refueling bridge (if loaded) from moving toward the core unless all control rods are fully inserted. Those interlocks may be bypassed during spiral loading except for those control cells which contain fuel or that control cell which is being loaded. Interlocks for all cells containing fuel, or for any cell about to be loaded, shall be operable.

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3.10 BASES (cont'd)

Switch is in the Refuel position only one control rod can be withdrawn except as noted in Specifications 3.10.A, D and E. The refueling interlocks, in combination with core nuclear design and refueling procedures limit the probability of an inadvertent criticality. The nuclear characteristics of the core assure that the reactor is subcritical even when the highest worth control rod is fully withdrawn. The combination of refueling interlocks for control rods and the refueling platform provide redundant methods of preventing inadvertent critically even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

For a new core the dropping of a fuel assembly into the vacant fuel location adjacent to a withdrawn control rod does not result in an excursion or a critical configuration, than adequate margin is provided.

A spiral unloading pattern in one in which the fuel in the outer-most cells (four fuel bundles surrounding a control blade) is removed first. Unloading continues by removing the remaining outermost fuel by cell no that the center cell with be removed last. Spiral loading is the reverse of unloading. Spiral loading and unloading preclude the formation of flux traps (moderator filled cavities surrounded on all sides by fuel). It is not necessary to accomplish a full core offload or onload in order to utilize the spiral movement procedure as long as the partial unloading/reloading.

3.10 BASES (cont'd)

The maintenance is performed with the Mode Switch in the Refuel position to provide the refueling interlocks normally available during Part A of these Bases. In order to withdraw a second control rod after withdrawal of the first rod, it is necessary to bypass the refueling interlock on the first control rod, which prevents more than one control rod from being withdrawn at the same time. The requirement that an adequate shutdown margin be demonstrated with the control rods remaining in-service insures that inadvertent criticality cannot occur during this maintenance. The shutdown margin is verified by demonstrating that the core is shutdown even if the strongest control rod remaining in-service is fully withdrawn. Disarming the directional control valves does not inhibit control rod scram capability.

The requirement for SRM operability during the maintenance is covered in Part B above.

The intent of this Specification is to permit the unloading of a significant portion of the reactor core for such purposes as in-service inspection requirements, examination of the core support plate, etc.

This Specification provides assurance that inadvertent criticality does not occur during such operation.

This operation is performed with the Mode Switch in the Refuel position to provide the refueling interlocks normally available during refueling as explained in Part A above. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod which prevents more than one control rod from being withdrawn at a time. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed ensures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with that control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.

The requirement for SRM operability during these operations is covered in Part B above.

3.11 (cont'd)

B. Crescent Area Ventilation

Crescent area ventilation and cooling equipment shall be operable on a continuous basis whenever specification 3.5.A, 3.5.B, and 3.5.C are required to be satisfied.

1. From and after the date that more than one unit cooler serving ECCS compartment are made or found to be inoperable, all ECCS components in that compartment shall be considered to be inoperable for purposes of specification 3.5.A, 3.5.B, and 3.5.C.

C. Battery Room Ventilation

Battery room ventilation shall be operable on a continuous basis whenever specification 3.9.E is required to be satisfied.

1. From and after the date that one of the battery room ventilation systems is made or found to be inoperable, its associated battery shall be considered to be inoperable for purposes of specification 3.9.E.

4.11 (cont'd)

B. Crescent Area Ventilation

Unit coolers serving ECCS components shall be checked for operability once/3 months.

1. When it is determined that two unit coolers serving ECCS components in the same compartment are made or found inoperable, reactor operation may continue for 7 days unless one is made operable earlier.

2. Temperature indicator controllers shall be calibrated once/operating cycle.

3. If 3.11.B.1 cannot be met, the reactor shall be placed in a cold condition within 24 hours.

C. Battery Room Ventilation

Battery room ventilation equipment shall be checked for operability once/week.

1. When it is determined that one battery room ventilation system is inoperable, the remaining ventilation system shall be checked for operability and daily thereafter.

2. Temperature transmitters and differential pressure switches shall be calibrated once/operating cycle.

3.11 (cont'd)

D. Emergency Service Water System

1. To ensure adequate equipment and area cooling, both ESW systems shall be operable when the requirements of specification 3.5.A and 3.5.B must be satisfied, except as specified below in specification 3.11.D.2.

4.11 (Cont'd)

D. Emergency Service Water System

1. Surveillance of the ESW system shall be performed as follows:

Item	Frequency
a. Simulated Automatic Actuation Test	Each operating cycle
b. Flow Rate Test - ESW pumps shall deliver at least 3,250 gpm against a system head corresponding to a total pump head of $\geq$80 psi, as determined from the pump certification curve by measuring the pump shutoff head which shall be $\geq$117 psi.	Once/ 3 months
c. Pump Operability	Once/month
d. Motor Operated Valves	Once/month

3.11 (cont'd)

E. <u>Intake Deicing Heaters</u>

Intake heaters are required to be operable when intake water temperature is ≤37°F. A minimum of 18 out of 88 heaters are required to be operable to maintain the required flow for the ESW and RHRSW System.

1. If specification 3.11.E.1 cannot be met the reactor shall be placed in a cold condition within 24 hours.

4.11 (cont'd)

E. <u>Intake Deicing Heaters</u>

1. The six heater feeder ammeters shall be checked weekly whenever the intake water temperature is ≤37°F.

2. The individual heaters shall be monitored once/6 months for rated heater current or as required by large deviations in the feeder checks in 4.11.E.1 above.

3. Resistance to ground shall be checked once/ operating cycle.

LIMITING CONDITIONS FOR OPERATION

3.12 FIRE PROTECTION SYSTEMS

Applicability:

Applies to the Operational Status of the Fire Protection Systems.

Objective:

To assure operability of the Fire Protection Systems.

Specification:

A. High Pressure Water Fire Protection System

1.a. Both high pressure water fire protection pumps and associated automatic and manual initiation logic shall be operable and aligned to the high pressure water fire header.

b. The high pressure water fire protection system shall be operable with an operable flow path capable of taking suction from the lake and transferring the water through distribution piping with operable sectionalizing control or isolation valves to the yard hydrant curb valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser required to be operable per specifications 3.12.B and 3.12.D.

SURVEILLANCE REQUIREMENTS

4.12 FIRE PROTECTION SYSTEMS

Applicability:

Applies to the Surveillance of the Fire Protection System.

Objective:

To verify the operability of the Fire Protection Systems.

Specification:

A. High Pressure Water Fire Protection System

1. High pressure water fire protection system testing:

	Item	Frequency
a.	High pressure water fire protection system pressure check.	Once/week
b.	Each pump, on a STAGGERED TEST BASIS, by starting and operating it for at least 20 minutes on recirculating flow	Once/month
c.	Valve operational test	Once/12 months
d.	System flush	Once/6 months
e.	Functional test including:	Once/18 months

C. Carbon Dioxide Systems (Cont'd)

2. If the CO_2 protection for the areas listed in Table 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 to restore the system to an operable status shall be prepared and submitted within 30 days.

D. Manual Fire Hose Stations

1. a. The manual fire hose stations listed in Table 3.12.3 shall be operable except as specified below:

   b. From and after the date that any of the manual fire hose stations listed in Table 3.12.3 is made or found to be inoperable, additional hose lengths shall be added to adjacent operable manual hose stations such that the entire area of protection is maintained within one hour.

D. Manual Fire Hose Stations

1. The manual fire hose stations are inspected as listed in Table 4.12.3.

3.12 and 4.12 BASES (continued)

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.

Performance of the periodic tests and inspections listed in Table 4.12.2 are in accordance with NFPA-12, 1973, will verify the integrity of system nozzles and distribution headers as well as detect and remove any accumulation of rust or scale. The use of "puff test" rather than full flow tests will demonstrate proper valve operation without the attendant potential equipment and personnel hazards associated with full flow tests.

D. Manual hose stations provide backup fire protection throughout the Plant. Those hose stations that are in or near areas with safety related equipment are listed in Table 3.12.3. Hose station location and hose length selection provides the capability of reaching any fire in a safety related area with the hose stream. When any of the hose stations listed in Table 3.12.3 is inoperable, providing additional hose lengths from other operable hose stations assures maintenance of this capability. Periodic inspection and tests are in accordance with NFPA Code guidelines and assures prevention, detection and correction of hose, nozzle, valve and/or gasket damage or deterioration to maintain high levels of operability.

E. Early fire detection and fire fighting activity is essential to ensuring that any fire will result in minimum damage to safety related equipment. Since each area monitored utilizes a number of smoke and/or heat detectors when more than one detector is inoperable, early fire detection is assured by establishing a patrolling fire watch which check the area where the detectors are inoperable at least hourly.

Testing of smoke and heat detectors and associated circuitry every 6 months, in accordance with manufacturers and NFPA 72E-1974 recommendations ensures a high level of operability.

F. Fire barrier penetration seals are designed to give 3 hours or more protection and to meet the requirements of IEEE - 383, "Fire Test of Building Construction and Materials". Visual inspection and leak testing ensure that seals are intact. Leak testing with open flame or combustion generated smoke is prohibited.

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 The minimum qualifications with regard to educational background and experience for plant staff positions shown in Fig. 6.2-1 shall meet or exceed the minimum qualifications of ANSI NI8.1-1971 for comparable positions; except for the Radiation and Environmental Services Superintendent 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 this license amendment, 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 Superintendent 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 NI8.1-1971.

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).

ATTACHMENT II
TO
JPN-88-023

PROPOSED ADMINISTRATIVE CHANGES
TO THE
TECHNICAL SPECIFICATIONS
(JPTS-86-004)

NEW YORK POWER AUTHORITY
JAMES A. FITZPATRICK NUCLEAR POWER PLANT
DOCKET NO. 50-333
DPR-59

Section I DESCRIPTION OF THE PROPOSED CHANGE

The proposed changes to the James A. FitzPatrick Technical Specifications affect several pages and specifications. Specifically, the changes are:

Section	Page	Description
[a] Table of Contents	i	·Add section 3.0 titled 'General'.
		·Change page number for Specification 3.1 from page '30' to page '30f'.
[b] List of Tables	vi	·Insert title for Table 4.6-2.
		·Insert title for Tables 3.12-1, 3.12-2, 3.12-3, 4.12-1, 4.12-2, and 4.12-3.
[c] List of Figures	vii	·Insert 'τ' for Figure 3.1-2 title.
		·Add 'and 3.5.J.3' to title of Figure 3.5-1.
		·Change Figure 3.5-9 title to 'Deleted'.
		·Insert title for Figure 3.5-12.
[d] Spec. 1.0	1	·Correct the spelling of 'explicitly'.
[e] Spec. 1.0.F.4	2	·Insert 'a' in first sentence defining instrument check.
[f] Spec. 1.0.I.4	4	·Add 'Amendment No. 83' to bottom of page.
		·Replace 'trips' with 'trip' in first sentence of spec. 1.0.4.
[g] Spec. 1.0.Q	5	·Correct the abbreviation of 'cont'd'.
		·Delete 'a' from definition of refueling outage.

Section	Page	Description
[h] Spec. 1.0.X.a	6	·Replace 'a' from definition of staggered test basis with '"n"'. Place n inside quotes (last line).
[i] Spec. 1.1.B	8	·Replace 'less than' with 'less than or equal to' prior to 10%.
[j] Spec. 1.2 & 2.2 BASES	29	·Correct spelling of 'resulting'.
[k] Spec. 3.0.E BASES	30b	·Correct spelling of 'inoperable'.
[l] Spec. 3.1 BASES	34	·Rewrite the top paragraph, right column, to clarify nuclear instrumentation coverage for the reactor modes of operation.
[m] Spec. 4.1 BASES	38	·Insert '44' into Amendment No.
[n] Table 3.1-1	41	·Insert 'X' to require Manual Scram Operability in Run Mode.
	41b	·Insert two '$<$' symbols in Trip Level Setting for Turbine Control Valve Fast Closure.
	42	·Insert a '$\leq$' symbol before 10% valve closure.
		·Insert 'less than' prior to 1005 psig in Note 3 to correct the condition for which the MSIV closure scram is bypassed.
[o] Spec. 3.2 BASES	57	·Correct five grammatical errors: 1) replace 'drop' with 'drops' 2) remove 'this' 3) replace 'of' with 'or' 4) replace 'and' with 'or' 5) replace 'setting' with 'settings'
		·Delete 'flow' from first sentence of the third paragraph in the right-hand column.
	58	·Correct spelling of 'channel'.
[p] Spec. 4.2 BASES	61	·Insert 't' into formula for optimum interval between tests.

	Section	Page	Description
[q]	Table 3.2-2	68	·Insert 'THIS ITEM INTENTIONALLY BLANK' for item no. 10.
		70a	·Replace 'psid' with 'dp' in Item No. 28.
			·Insert 'THIS ITEM INTENTIONALLY BLANK' for item nos. 22, 23 & 24.
		70b	·Insert 'dp' in Item No. 31.
[r]	Notes for Table 3.2-6	76c	·Correct spelling of 'permissible'.
[s]	Table 3.2-7	77	·Replace '>' with '≥' under the trip level setting column for the reactor low-low water level instrument prior to -38.
[t]	Specs. 3.3.A.2 & 4.3.A.2	89a 90	·Retype such that paragraph 4.3.A.2.e resides in the right side column and paragraph 3.3.A.2.d continues with text currently on page 90.
[u]	Spec. 3.3.B.1	91	·Replace 'rods' with 'rod' in first sentence.
[v]	Spec. 4.3.B.3 BASES	101	·Delete 'feedwater and' from the fourth complete sentence in the first paragraph in the right-hand column.
[w]	Spec. 3.4.B	106	·Insert 'B' for paragraph number.
[x]	Spec. 3.5.A.2	114	·Insert change bars for Amendment 95.
[y]	Spec. 4.5.B.1	115a	·Delete 'of 3,700 gpm' from third sentence.
[z]	Spec. 3.5.C.b	118	·Change referenced paragraph from '3.5.C' to '3.5.C.1'.
[aa]	Specs. 3.5.D & 4.5.D	119	·Replace 'relief/safety' with 'safety/relief'.
[bb]	Spec. 3.5.F	122	·Insert 'F' for paragraph identification.

Section	**Page**	**Description**
[cc] Spec. 3.5.H	123	·Insert 'at' into fourth sentence describing APLHGR.
[dd] Spec. 3.6.E & 4.6.E	142a	·Delete note for effective date.
	142b	·Delete entire expired page.
	143	·Delete note for effective date.
		·Change referenced specification from '3.6.B.1' to '3.6.E.1' and from '3.6.B.2' to '3.6.E.2'.
	143a,b	·Delete both expired pages.
[ee] Spec 3.6.G	144	·Replace 'operable' with 'inoperable' in second sentence of paragraph G (Jet Pumps).
[ff] Specs. 3.6.E & 4.6.E BASES	152	·Delete 'coincident high drywell pressure and' (second sentence, second paragraph) and replace 'low-low' with 'low-low-low' from the same sentence and paragraph in the left hand column.
		·Replace 'relief/safety' with 'safety/relief'.
	153	·Insert 'less than or equal to' into the second line (left-hand column) prior to 212°F.
[gg] Spec. 3.6.F & 4.6.F BASES	153	·Correct spelling of 'will'.
[hh] Spec. 4.7.A.2.c.1	171	·Change referenced Table '3.7-1' to '4.7-2'.
[ii] Spec. 4.7.A.2.c.4	172	·Change referenced Table '3.7-2' to '4.7-2'.
[jj] Spec. 4.7.A.2.e,f	174	·Insert '40' in Amendment No.

Section	Page	Description
		·Replace 'on' with 'of' in fourth sentence of spec. 4.7.A.2.f.
[kk] Spec. 3.7.4.a	177	·Insert 'less than or equal to ' into the last line prior to 0.5.
[ll] Spec. 3.7.A.5.a	178	·Replace 'The' with 'When' in first line.
[mm] Spec. 3.7.A.5.g	179	·Replace 'chamberreactor building' with 'chamber/drywell' in first sentence.
[nn] Spec. 4.7.D.1.c.2	186	·Insert ',' after 'level' and delete ',' after 'trip'.
[oo] Spec. 4.7.A BASES	194	·Replace 'AEC' with 'Commission' in first and second sentences.
		·Reposition of a comma in first sentence.
[pp] Table 3.7-1	201	·Replace Drywell Penetration for the TIP Purge from 'X-35B' to 'X-35E'.
[qq] Table 3.7-2	211 - 213	·Replace '3.7-2' to '4.7-2'.
[rr] Spec. 3.8 & 4.8.2	214	·Insert 'contamination, the source shall be decontaminated,' into the third sentence of the first paragraph.
		·Insert an 'a' into Specification 4.8.2 (third sentence).
[ss] Spec. 3.9.C.2.a	219	·Insert 'one' into first sentence of spec. 3.9.C.2.a
[tt] Spec. 4.9.F BASES	226	·Correct spelling of 'tests'.
[uu] Specification 3.10.8	230	·Insert '59' into Amendment No.
		·Delete expired spec. 3.10.8.
	230a	·Delete entire expired page.

Section	Page	Description
[vv] Spec. 3.10 BASES	235	·Correct spelling of 'not'.
[ww] Spec. 3.10 BASES	236	·Correct spelling of 'BASES' at the top of the page.
[xx] Spec. 3.11.B.1	239	Replace '3.5.C' with '3.5.B' and '3.5.D' with '3.5.C' in last two lines.
[yy] Spec. 4.11.C.1	239	·Correct spelling of 'ventilation'.
[zz] Spec. 3.11.D.1	240	·Insert 'both ESW systems shall be' into first sentence.
		·Delete 'the' from first line.
[aA] Spec. 4.11.E.2	242	·Replace '3.11.E.1' with '4.11.E.1'.
[bB] Spec. 4.12.A.1.b	244a	·Replace 'once/week' with 'once/month'.
[cC] Spec. 3.12.D.1.a & b	244f	·Replace '3.12.2' with '3.12.3'.
[dD] Spec. 3.12.D BASES	244i	·Replace '3.12.2' with '3.12.3' in second sentence.
[eE] Spec. 6.4	248	·Replace 'Coordinator' with 'Superintendent'.

Section II PURPOSE OF THE PROPOSED CHANGE

The purpose of the proposed changes are to correct typographical and other errors in the Technical Specifications. These errors were discovered during the process of obtaining previous license amendments. The types of errors within the Technical Specifications include typographical errors (e.g., misspelled words), grammatical errors (e.g., unnecessary words), and expired pages. These changes are administrative and will improve the clarity of the Technical Specifications.

The proposed change (item [a]) to page i adds the title for Specification [illegible] and changes the page number for Specification [illegible]. Both of these changes are due to the addition of Specification 3.0 which was approved and issued as Amendment 83.

The proposed changes (item [b]) to page vi adds the title for Tables 4.6-2, 3.12-1, 3.12-2, 3.12-3, 4.12-1, 4.12-2, and 4.12-3. Table 4.6-2 was added to the Specifications by Amendment 28 but never entered into the List of Tables. Tables 3.12-1 - 3.12-3 and 4.12-1 - 4.12-3 were added to the Specifications by Amendment 34 but never entered into the List of Tables.

The proposed changes (item [c]) to page vii updates the List of Figures. The titles for Figures 3.1-2 and 3.5-1 are incomplete and need to be completed. Figure 3.5-9 was deleted by Amendment 109 but its entry in the List of Figures was never removed. Figure 3.5-12 was added to the Specifications by Amendment 109 but never entered into the List of Figures.

The proposed change (item [d]) to page 1 corrects the spelling of 'explicitly'.

The proposed change (item [e]) to page 2 inserts an 'a' into the first sentence defining instrument check. This change will make the sentence grammatically correct.

The proposed change (item [f]) to page 4 adds 'Amendment No. 83' to the bottom of the page. This page was updated by Amendment 83 but never noted on the page. Also, a grammatical change is made to the first sentence of the startup/hot standby mode of operation description.

The proposed change (item [g]) to page 5 corrects the spelling of the abbreviation of 'cont'd' and deletes 'a' from the definition of refueling outage. This correction makes the definition grammatically correct.

The proposed change (item [h]) to page 6 replaces 'a' with '"n"' in Specification 1.0.X.a. In this case, '"n"' means number of systems. Also, the n that exists in the last line of this Specification is placed inside quotes.

The proposed change (item [i]) to page 8 inserts an inequality expression to account for core flow less than or equal to 10% of rated.

The proposed change (item [j]) to page 29 corrects the spelling of 'resulting'.

The proposed change (item [k]) to page 30b corrects the spelling of 'operable'.

The proposed change (item [l]) to the top paragraph, right-hand column on page 34 is intended to clarify the existing paragraph for nuclear instrumentation coverage for the reactor modes of operation. The rewritten paragraph does not change the meaning of the paragraph but makes it more understandable.

The proposed change (item [m]) to page 38 adds '44' into Amendment No. to signify this page was previously updated by this amendment.

The proposed change (item [n]) to Table 3.1-1 (page 41) adds a requirement that a manual scram trip function be operable in the run mode. This requirement was inadvertently deleted when the table was updated for a submittal which was subsequently approved and issued as Amendment 98 to the FitzPatrick Technical Specifications.

The proposed change (item [n]) to Table 3.1-1 (page 41b) adds two '<' symbols for the turbine control valve fast closure trip level setting. The symbols were inadvertently omitted when the table was last updated.

The proposed changes (item [n]) to Table 3.1-1 (page 42) inserts '≤' prior to 10% valve closure and inserts 'less than' to Note 3. Both omissions occurred while updating the page for a previous submittal.

The proposed changes (item [o]) to page 57 correct five grammatical errors. Also, 'flow' is deleted from the paragraph describing reactor water cleanup instrumentation since this system is independent of flow.

The proposed change (item [o]) to page 58 corrects the spelling of 'channel'.

The proposed change (item [p]) to page 61 inserts a 't' into the mathematical equation for the optimum interval between tests.

The proposed change (item [q]) to Table 3.2-2 (page 68) inserts 'THIS ITEM INTENTIONALLY BLANK'. This line item description should have been included in the submittal for Amendment 84.

The proposed change (item [q]) to Table 3.2-2 (page 70a) replaces 'psid' with 'dp' in item no. 28. This is the appropriate expression for differential pressure in this instance. Also, insert a line item description that the instrumentation for item nos. 22 - 24 has been deleted. The instrumentation was removed by a plant modification which was approved by Amendment 48.

The proposed change (item [q]) to Table 3.2-2 (page 70b) inserts 'dp' in item no. 31 to complete the trip level setting for the HPCI turbine steam line high flow trip function.

The proposed change (item [r]) to page 76c corrects the spelling of 'permissible'.

The proposed change (item [s]) to page 77 corrects a wrong inequality symbol.

The proposed change (item [t]) to pages 89a and 90 rearrange the pages so that the appropriate paragraphs fall under the appropriate specifications (Specs. 3.3.A.2 and 4.3.A.2).

The proposed change (item [u]) to page 91 (first sentence of Spec. 3.3.B.1) corrects a typographical error.

The proposed change (item [v]) to page 101 eliminates the statement indicating that the power level for automatic cutout of the rod worth minimizer is sensed by feedwater flow. Automatic cutout of the RWM is sensed only by steam flow.

The proposed change (item [w]) to page 106 inserts 'B.' to identify Specification 3.4.B.

The proposed change (item [x]) to page 114 adds the change bars from Amendment 95. The change bars are necessary to indicate what text has been previously changed.

The proposed change (item [y]) to page 115a deletes 'of 3700 gpm' from Specification 4.5.B.1. This deletion should have been included in the submittal for Amendment 71. Amendment 71 pertained to the emergency service water pump surveillance requirement.

The proposed change (item [z]) to page 118 changes a reference specification. The Specification currently referenced (3.5.C) does not exist and is replaced with the correct Specification (3.5.C.1).

The proposed change (item [aa]) to page 119 replaces 'relief/safety' with 'safety/ relief'. This change is necessary for the specifications to be consistent.

The proposed change (item [bb]) to page 122 inserts 'F' to identify the specification.

The proposed change (item [cc]) to page 123 grammatically corrects the fourth sentence of Specification 3.5.H.

The proposed change (item [dd]) to page 142a deletes the note for the effective date.

The proposed changes (item [dd]) to pages 142b, 143a, and 143b delete the pages in their entirety. These pages are no longer applicable.

The proposed change (item [dd]) to page 143 removes the note for effective date. This note is no longer necessary.

The proposed change (item [ee]) to page 144 changes operable to inoperable in the second sentence of Specification 3.6.G. The existing specification reads:

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

The second sentence contradicts the first sentence but is in agreement and correct if operable is changed to inoperable. The specification with the proposed change reads:

> "Whenever the reactor is in the start-up/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."

The proposed change (item [ff]) to page 152 eliminates "...coincident high drywell pressure..." from the second paragraph of Specification 3.6.E BASES. This same change was made to pages 66, 68, and 69 by Amendment 84. This change is a result of a plant modification (F1-83-034) which removed the high drywell pressure permissive for ADS actuation from the ADS logic. The change to this page was inadvertently omitted from the submittal for Amendment 84.

Also, on this same page and paragraph, "low-low" is being replaced by "low-low-low". This error has existed since the initial issuance of the Technical Specifications. As evidenced in Table 3.2-2, ADS actuation results due to a trip caused by reactor low-low-low water level. This change will make the specification consistent with the correct condition listed in the table.

Also on page 152, 'relief/safety' is replaced with 'safety/relief' to maintain consistency in the Technical Specifications.

The proposed change (item [ff]) to page 153 inserts an inequality expression regarding safety/relief valves. The expression is replacing its corresponding inequality sign.

The proposed change (item [gg]) to page 153 corrects the spelling of 'will'.

The proposed change (item [hh]) to page 171 correctly identifies a referenced table.

The proposed change (item [ii]) to page 172 correctly identifies a referenced table.

The proposed change (item [jj]) to page 174 corrects two errors. A missing amendment no. is added and a grammatical error is corrected.

The proposed change (item [kk]) to page 177 inserts a missing inequality expression.

The proposed change (item [ll]) to page 178 corrects a grammatical error.

The proposed change (item [mm]) to page 179 replaces 'chamberreactor' with 'chamber/drywell'. This is a necessary change in order to achieve consistency with Specification 3.7.A.4.

The proposed change (item [nn]) to page 186 repositions a punctuation mark to make the sentence grammatically correct.

The proposed change (item [oo]) to page 194 replaces 'AEC' with 'Commission'. Also, a punctuation mark is repositioned.

The proposed change (item [pp]) to page 201 replaces drywell penetration 'X-35B' with 'X-35E'. This change will correctly identify the drywell penetration for the TIP purge.

The proposed change (item [qq]) to pages 211 - 213 changes Table 3.7-2 to Table 4.7-2. This was the original number of the table and was renumbered by Amendment 40. This revised table number is incorrect and is being changed back to the original number.

The proposed change (item [rr]) to page 214 inserts text into the third sentence (first paragraph) of Specification 3.8. This text has been missing since the initial issuance of the Technical Specifications. Also, an 'a' is inserted into the third sentence of Specification 4.8.[illegible] to make it grammatically correct.

The proposed change (item [ss]) to page 219 inserts 'one' into the first sentence of specification 3.9.C.2.a. This word was inadvertently omitted when the page was retyped for the submittal of Amendment 83.

The proposed change (item [tt]) to page 226 corrects the spelling of 'tests'.

The proposed change (item [uu]) to page 230 deletes the expired Specification 3.10.8. The entire page 230a is deleted which contains the remainder of this specification. Page 230a will be replaced with page stating that the page has been intentionally deleted.

The proposed change (item [vv]) to page 235 corrects the spelling of 'not'.

The proposed change (item [ww]) to page 236 corrects the spelling of 'BASES'.

The proposed change (item [xx]) to page 239 correctly identifies two referenced specifications.

The proposed change (item [yy]) to page 239 corrects the spelling of 'ventilation'.

The proposed change (item [zz]) to page 240 inserts text missing since the initial issuance of the Technical Specifications to the first sentence of Specification 3.11.D.1. Also, a grammatical change is made to this same specification.

The proposed change (item [aA]) to page 242 correctly identifies a referenced specification.

The proposed change (item [bB]) to page 244a replaces the "once/week" surveillance frequency for the high pressure water fire protection system pumps to "once/month". This error was introduced in the submittal approved and issued as Amendment 80. This change to the surveillance frequency was never analyzed by the Authority or NRC. As a result, this change restores the surveillance frequency to its pre-Amendment 80 value.

The proposed change (item [cC]) to page 244f corrects the referenced Tables in Specifications 3.12.D.1.a and b.

The proposed change (item [dD]) to page 244i correctly identifies the referenced Table in Specification 3.12.D BASES.

The proposed change (item [eE]) to page 248, Specification 6.4, replaces 'Coordinator' to 'Superintendent' to be consistent with Figure 6.2-1 (organizational chart).

Section III IMPACT OF THE PROPOSED CHANGE

The proposed changes to the FitzPatrick Technical Specifications will not impact plant safety. All of the changes are administrative or editorial in nature. The proposed changes involve no limiting conditions for operations, surveillance requirement, setpoint or safety limit changes, nor do they affect the environmental monitoring program. These changes clarify and improve the quality of the Technical Specifications by correcting editorial errors.

The majority of the proposed changes are the correction of typographical errors. These errors consist primarily of misspelled words but also include inadvertent deletion or misplacement of text. In the process of updating a page for a previous amendment, portions of text were inadvertently deleted. In the case of misplaced text, which occurred once, a paragraph of one specification was moved into the adjacent specification. Resolution of these types of errors restore the Technical Specifications to their proper format. These proposed changes do not effect plant operations since they are administrative in nature.

Grammatical errors exist as well in the Technical Specifications. These errors include missing words and improper punctuation. Correction of these errors make the Technical Specifications easier to read. As a result, plant operations or safety is not impacted.

Other errors are strictly administrative. These include expired pages and specifications. Removal of the expired pages and specifications eliminate the possibility of referring to an inapplicable specification. These proposed changes do not impact plant operations or safety.

The proposed change to Table 3.1-2 (page 41) adds a requirement that a manual scram trip function be operable in the run mode. This requirement has existed since the initial issuance of the Technical Specifications but was removed inadvertently when updating the table for a previous submittal. This proposed change returns the table to its original condition and does not impact plant operations or safety.

The proposed change to page 144 will correct a condition for plant shutdown resulting from jet pump operability. The existing specification states that the reactor shall be placed in a cold condition if a jet pump is operable. This is in direct contrast to the preceding sentence which states that whenever the reactor is in a mode other than refueling that all jet pumps be operable. This proposed change will correct the specification to state that whenever a jet pump is inoperable that the reactor be placed in a cold condition. This proposed change will not affect plant operations.

The proposed changes to page 152 are needed to achieve consistency throughout the Technical Specifications. The elimination of "...coincident drywell pressure..." from specification 3.6.E BASES is consistent with the changes brought about as a result of a plant modification which eliminated the high drywell pressure permissive for ADS actuation from the ADS logic. This change has previously been approved and issued by Amendment 84. The inclusion of this change in the submittal for this amendment was inadvertently omitted. The proposed change has no impact on plant safety or operations.

Replacement of "low-low" in the same specification with "low-low-low" is necessary in order for the specification to agree with Table 3.2-2. This error has existed since the initial issuance of the Technical Specifications. The proposed change will have no impact on plant operations or safety.

The proposed change to page 244a replaces the "once/week" surveillance frequency for the high pressure water fire protection system pumps to "once/month". This change was erroneously introduced in the process of updating the page for a submittal which was approved and issued as Amendment 80. This submittal did not include a change to this surveillance frequency. This surveillance frequency of once/month is a condition that has always existed. This change will return the surveillance interval to its original value and does not impact plant operations or safety.

The proposed changes to the Technical Specifications do not change any system or subsystem and will not alter the conclusions of either the FSAR or SER accident analysis.

Section IV EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

The proposed changes to the James A. FitzPatrick Technical Specifications involve no significant hazards considerations. They are all administrative or editorial in nature and include: typographical errors; grammatical errors; and clarification of a specification. Operation of the FitzPatrick Plant in accordance with the proposed amendment would not involve a significant hazards consideration as stated in 10 CFR 50.92 since it would not:

(1) involve significant increase in the probability or consequences of an accident previously evaluated. The intent of the proposed changes are to clarify and correct the Technical Specifications. The changes are administrative and include: correction of misspelled words; deletion of expired pages; and correction of grammatical errors. There are no setpoint changes, safety limit changes, surveillance requirement changes, or limiting

conditions for operation. These changes have no impact on plant safety or plant operations. The changes will have no impact on previously evaluated accidents.

(2) create the possibility of a new of different kind of accident previously evaluated. The proposed changes are purely administrative in nature and involve only the correction of typographical and other errors. These proposed changes are intended to clarify and improve the quality of the Technical Specifications. This cannot create the possibility of a new or different kind of accident.

(3) involve a significant reduction in the margin of safety. The proposed changes correct errors which currently exist in the Technical Specifications. The changes are all administrative in nature and will clarify the specifications by eliminating errors such as typographical errors. 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 no affect the plant's safety systems and do not reduce any safety margin.

In the April 6, 1983 Federal Register (48FR14870), NRC published examples of license amendments that are not likely to involve a significant hazards consideration. Example (i) of that list is applicable to this change and states:

"A purely administrative change to the Technical Specifications: for example, ...correction of an error...".

The Authority considers that the proposed changes can be classified as not likely to involve significant hazards considerations, since the changes are administrative or editorial in nature and do not involve hardware changes nor any changes to the plant's safety related structures, systems, or components. The proposed changes are designed to improve the quality of the Technical Specifications.

Section V Implementation of the Proposed Change

The proposed change will not adversely impact the ALARA, Security or Fire Protection programs at the FitzPatrick plant, nor will it impact the environment.

Section VI Conclusion

The change, as proposed does not constitute an unreviewed safety question as defined in 10 CFR 50.59, that is, it:

a. will not increase the probability or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report:

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

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

d. does not constitute an unreviewed safety question; and

e. involves no significant hazards consideration, as defined in 10 CFR 50.92.

Section VII REFERENCES

1. James A. FitzPatrick Nuclear Power Plant Final Safety Analysis Report (FSAR).

2. James A. FitzPatrick Nuclear Power Plant Safety Evaluation Report (SER).