EXHIBIT A

Monticello Nuclear Generating Plant

License Amendment Request dated February 15, 1983

Proposed changes to the Technical Specifications
Appendix A of Operating License DPR-22

Pursuant to 10 CFR 50.59 and 50.90 the holders of Operating License DPR-22 hereby propose the following changes to Appendix A, Technical Specifications:

1. Turbine Bypass Valves Pages (ii, 113, 119)

Proposed Change

Add new limiting conditions for operation, 3.5.H, and surveillance requirements, 4.5.H, for the Turbine Bypass Valves.

Reason for Change

These valves mitigate the pressure transient following turbine trips and are assumed in the reload analysis to function during a feedwater controller failure.

This change was requested in the Safety Evaluation Report from the NRC dated December 6, 1982.

Safety Evaluation

Below 30% reactor power this system is not required to prevent fuel damage. The addition of this specification to the Technical Specifications will not affect plant safety.

2. Feedwater/Turbine Trip Instrumentation (pages i,vi,46,60b,62,69,70,71)

Proposed Change

Add the new Table 3.2.7: "Instrumentation for Plant Systems." Add the feedwater/turbine trip on high reactor water level to the table. The setpoint of 14'6" above the top of the active fuel is equivalent to 48" on the level instrumentation and 525.5" above the lower head. Add the surveillance requirements on the feedwater/turbine trip to Table 4.2.1. The requirements are very similar to the requirements of other reactor water level instrumentation. Renumber the "Trip Functions and Deviations" Table to be Table 4.2.2 The allowed deviation of +6 inches is listed in Table 4.2.2.

8302180146 830215
PDR ADOCK 05000263
P PDR

Reason for Change

This trip mitigates the feedwater controller failure and is assumed to function during that transient.

This change was requested in the Safety Evaluation Report from the NRC dated December 6, 1982.

Safety Evaluation

The Cycle 10 analysis used 525.3" above the lower head as the high reactor water level trip setpoint. General Electric believes that the analysis is not sensitive to the high water level trip setpoint. In future analyses we plan to use 525" + 6" = 531" as the high reactor water level trip setpoint.

The addition of this specification to the Technical Specification will not affect plant safety.

3. RPS Power Monitoring System Setpoint Time Delays (page 27)

Proposed Changes

Change the RPS Power Monitoring System Setpoint time delays (section 3.1.C.1) from "100 milliseconds" to "4 seconds".

Reasons for Change

A 4 second time delay is needed to prevent premature trips from occurring during routine switching operations.

Safety Evaluation

This extension of the electrical protection assembly time delays reflects the conclusion of a study of the RPS bus components which found it acceptable to allow an under voltage, over voltage, or under frequency condition to exist for up to four (4) seconds. This change will increase the stability of the alternate source supplying the RPS Bus Power and help alleviate unnecessary tripping that could result from normal switching operations. It does not remove any of the protections provided by this system.

4. Error in Control Rod Withdrawal Bases (page 89)

Proposed Change

Change "10%" to "10^{-8}".

Reason for Change

Correct Error.

Safety Evaluation

None required.

5. Screen Wash/Fire Pump (pages 224, 225, 228, 229)

Proposed Change

Change "screen wash pump" to "screen wash/fire pump" and reword bases.

Reason for Change

The pump name was changed to indicate its use as a fire pump.

Safety Evaluation

None required

6. Snubbers (pages 131, 132, 132a)

Proposed Change

Add the following snubbers:

RV26-H3A	SS-191
RV25-N2	SS-185
SS-121	SS-187
SS-184	SS-211

Delete the following snubbers:

RV25-NS3	RV26A-NS3

Note: These pages also include changes requested in our September 24, 1982 License Amendment Request.

Reason for Change

Installation of the Combustible Gas Control System, modifications to the Safety/Relief Valve discharge lines, and modifications to the Torus-attached piping necessitates the addition of the above 8 snubbers to Table 3.6.1.

Two snubbers have been replaced by restraints and therefore should be deleted from the table.

Safety Evaluation

Non required.

EXHIBIT B

License Amendment Request dated February 15, 1983

Docket No. 50-263
License No. DPR-22

Exhibit B consists of revised pages for the Monticello Nuclear Generating Plant Technical Specifications as listed below:

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3.0 LIMITING CONDITIONS FOR OPERATION

B. Upon discovery that the requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated to:

1. Satisfy the minimum requirements by placing appropriate devices, channels, or trip systems in the tripped condition, or

2. Place and maintain the plant under the specified required conditions using normal operating procedures.

C. RPS Power Monitoring System

1. Except as specified below, both channels of the power monitoring system for the MG set or alternate source supplying each reactor protection system bus shall be operable with the following setpoints:

		Time Delay
a. Over-voltage	- ≤128 VAC	≤ 4 seconds
b. Under-voltage	- ≥104 VAC	≤ 4 seconds
c. Under-frequency	- ≥57 HZ	≤ 4 seconds

2. With one RPS electric power monitoring channels for the MG set or alternate source supplying each reactor protection system bus inoperable, restore the inoperable channel to Operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

3. With both RPS electric power monitoring channels for the MG set or alternate source supplying each reactor protection system bus inoperable, restore at least one to Operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

4.0 SURVEILLANCE REQUIREMENTS

B. Once per day during power operation the MFLPD (Maximum Fraction of Limiting Power Density) shall be checked and the scram setting given by the equation in Specification 2.3.A shall be adjusted if necessary.

C. RPS Power Monitoring System

1. Instrument Functional Tests of each RPS power monitoring channel shall be performed at least once every six months.

2. At least once each Operating Cycle an Instrument Calibration of each RPS power monitoring channel shall be performed to verify over-voltage, under-voltage, and under-frequency setpoints.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
B. Emergency Core Cooling Subsystems Actuation When irradiated fuel is in the reactor vessel and the reactor water temperature is above 212°F, the limiting conditions for operation for the instrumentation which initiates the emergency core cooling subsystems are given in Table 3.2.2. C. Control Rod Block Actuation The limiting conditions of operation for the instrumentation that initiates control rod block are given in Table 3.2.3. D. Instrumentation for Plant Systems Whenever the reactor is in the RUN Mode, the limiting conditions for operation for the instrumentation listed in Table 3.2.7 shall be met.	

Table 3.27

Instrumentation for Plant Systems

Function	Trip Setting	Minimum No. of Operable or Operating Trip Systems (1)	Total No. of Instrument Channels Per Trip System	Minimum No. of Operable or Operating Instrument Channels Per Trip System (1)	Required Conditions*
Feedwater/Turbine Trip Actuations:					
1. High Reactor Water Level	≤ 14' 6" above the top of the active fuel	2	2	2	A

NOTE:

1. Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied, action shall be initiated to:

   a. Satisfy the requirements by placing the appropriate channels or systems in the tripped condition, or

   b. Place the plant under the specified required condition using normal operating procedures.

* Required conditions when minimum conditions for operation are not satisfied:

   A. Reactor in Startup, Refuel or Shutdown mode.

Table 4.2.1 - Continued
Minimum Test and Calibration Frequency For Core Cooling
Rod Block and Isolation Instrumentation

Instrument Channel	Test (3)	Calibration (3)	Sensor Check (3)
3. Steam Line Low Pressure	Note 1	Once/3 months	None
4. Steam Line High Radiation	Once/week (5)	Note 6	Once/shift
HPCI ISOLATION			
1. Steam Line High Flow	Once/month	Once/3 months	None
2. Steam Line High Temperature	Once/month	Once/3 months	None
RCIC ISOLATION			
1. Steam Line High Flow	Once/month	Once/3 months	None
2. Steam Line High Temperature	Note 1	Once/3 months	None
REACTOR BUILDING VENTILATION			
1. Radiation Monitors (Plenum)	-	See Table 4.8.2	-
2. Radiation Monitors (Refueling Floor)	Note 1	Once/3 months	(4)
RECIRCULATION PUMP TRIP			
1. Reactor High Pressure	Note 1	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/Day
2. Reactor Low Water Level (Note 7)	Once/month	Once/Operating Cycle-Transmitter Once/3 Months-Trip Unit	Once/shift
FEEDWATER/TURBINE TRIP			
1. High Reactor Water Level	Once/month	Once/3 months	Once/day

Bases Continued:

increases core voiding, a negative reactivity feedback. High pressure sensors initiate the pump trip in the event of an isolation transient. Low level sensors initiate the trip on loss of feedwater (and the resulting MSIV closure). The recirculation pump trip is only required at high reactor power levels, where the safety/relief valves have insufficient capacity to relieve the steam which continues to be generated after reactor isolation in this unlikely postulated event, requiring the trip to be operable only when in the RUN mode is therefore conservative.

The feedwater/turbine trip is actuated by a high reactor water level signal. This trip mitigates the feedwater controller failure transient by tripping the feedwater pumps and the turbine.

Although the operator will set the set points within the trip settings specified in Tables 3.2.1 through 3.2.7, the actual values of the various set points can differ appreciably from the value the operator is attempting to set. The deviations could be caused by inherent instrument error, operator setting error, drift of the set point, etc. Therefore, these deviations have been accounted for in the various transient analyses and the actual trip settings may vary by the amount shown in Table 4.2.2.

Table 4.2.2
Trip Functions And Deviations

	Trip Function	Deviation
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Specification 3.2.E.3 and Table 3.2.4	Ventilation Plenum Radiation Monitors	+0.2 Mr/Hr
	Refueling Floor Radiation Monitors	+5 Mr/Hr
	Low Reactor Water Level	-6 inches
	High Drywell Pressure	+ 1 psi
Primary Containment Isolation Functions Table 3.2.1	Low Low Water Level	-3 inches
	High Flow in Main Steam Line	+2 %
	High Temp. in Main Steam Line Tunnel	+2°F
	Low Pressure in Main Steam Line	-10 psi
	High Drywell Pressure	+1 psi
	Low Reactor Water Level	-6 inches
	HPCI High Steam Flow	+7,500 lb/hr
	HPCI Steam Line Area High Temp.	+2°F
	RCIC High Steam Flow	+2250 lb/hr
	RCIC Steam Line Area High Temp	+2°F

Table 4.2.2- Continued
Trip Function and Deviations

	Trip Function	Deviation
Instrumentation That Initiates Emergency Core Cooling Systems Table 3.2.2	Low-Low Reactor Water Level	-3 inches
	Reactor Low Pressure (Pump Start) Permissive	-10 psi
	High Drywell Pressure	+1 psi
	Low Reactor Pressure (Valve Permissive)	-10 psi
Instrumentation That Initiates Rod Block Table 3.2.3	IRM Downscale	-2/125 of Scale
	IRM Upscale	+2/125 of Scale
	APRM Downscale	-2/125 of Scale
	APRM Upscale	See Basis 2.3
	RBM Downscale	-2/125 of Scale
	RBM Upscale	Same as APRM Upscale
	Scram Discharge Volume-High Level	+ 1 gallon
Instrumentation That Initiates Recirculation Pump Trip	High Reactor Pressure	+ 12 psi
	Low Reactor Water Level	-3 inches
Instrumentation that Initiates Feedwater/Turbine Trip	High Reactor Water Level	+6 inches

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip settings, or, when a sufficient number of devices have been affected by any means such that the automatic function is incapable of operating within the allowable deviation while in a reactor mode in which the specified function must be operable or when actions specified are not initiated as specified.

consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The consequences of a rod block monitor failure have been evaluated. These evaluations show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR's below the Safety Limit (T.S.2.1.A). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Engineer, Nuclear, to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit (T.S.2.1.A). This requires the negative reactivity insertion in any local region of the core and in the overall core to be equivalent to at least the scram reactivity curve used in the transient analysis. The required average scram times for three control rods in all two by two arrays and the required average scram times for all control rods are based on inserting this amount of negative reactivity at the specified rate locally and in the overall core. Under these conditions, the thermal limits are never reached during the transients requiring control rod scram. The limiting operational transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains above the Safety Limit (T.S.2.1.A).

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. When irradiated fuel is in the reactor vessel and reactor coolant temperature is less than 212°F, all low pressure core and containment cooling subsystems may be inoperable provided no work is being done which has the potential for draining the reactor vessel except as allowed by specification 3.5.G.4 below.	
4. When irradiated fuel is in the reactor vessel and the vessel head is removed, the suppression chamber may be drained completely and no more than one control rod drive housing or instrument thimble opened at any one time provided that the spent fuel pool gates are open and the fuel pool water level is maintained at a level of greater than or equal to 33 feet.	
H. Turbine Bypass Valves	H. Turbine Bypass Valves
1. Except as specified in 3.5.H.2 below, the turbine bypass valves shall be operable whenever the reactor power is above 30%.	1. The turbine bypass valves shall be cycled every week.
2. From and after the date that one or both of the turbine bypass valves are made or found inoperable for any reason, reactor operation above 30% reactor power is permissible only during the succeeding seven days unless such valve is sooner made operable.	2. During each operating cycle the valves shall be functionally tested including simulated automatic actuation and verifying that each valve actuates to its correct position.

G. Emergency Cooling Availability

The purpose of Specification G is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time tbat all core and containment cooling subsystems may be out of service. Specification 3.5.G.3 allows all core and containment cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.G.4 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

H. Turbine Bypass Valves

The purpose of specification H is to assure that the bypass valves will be available to mitigate the consequences of transients following a turbine trip from above 30% power. This system is not a cooling or containment system. Following the discovery of a condition of inoperability of the system, plant operation above 30% power is limited to a seven day period. This provides flexibility in choosing a time to take the unit out of service to make necessary repairs.

I. Recirculation System

The capacity of the Emergency Core Coolant System is based on the potential consequences of a double ended recirculation line break. Such a break involves 3.9 sq. ft. when the cross tie valves are closed and 5.3 sq.ft. when the cross tie valves are open. Specification 3.11.A is based on an ECCS evaluation assuming a break area of 3.9 sq. ft.; the limitations of 3.11.A do not apply to the larger break area. Therefore, at least one cross tie valve must remain closed with two pump operation to reduce the potential break area.

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	ACCESSIBLE -A INACCESSIBLE-I
PS1-H2	MAIN STEAM	DRYWELL	953	071	I
PS1-H3	MAIN STEAM	DRYWELL	950	148	I
PS2-H2	MAIN STEAM	DRYWELL	950	120	I
PS3-H2	MAIN STEAM	DRYWELL	950	240	I
PS4-H3	MAIN STEAM	DRYWELL	950	212	I
RV24-H3	SAFETY-RELIEF	DRYWELL	950	110	I
RV24-H4	SAFETY-RELIEF	DRYWELL	935	100	I
RV24-H4A	SAFETY-RELIEF	DRYWELL	935	100	I
RV24-H5	SAFETY-RELIEF	DRYWELL	935	110	I
RV24-NS2	SAFETY-RELIEF	DRYWELL	934	081	I
RV24-NS3	SAFETY-RELIEF	DRYWELL	962	090	I
RV24-N1	SAFETY-RELIEF	DRYWELL	953	090	I
RV24A-H4A	SAFETY-RELIEF	DRYWELL	947	048	I
RV24A-H7	SAFETY-RELIEF	DRYWELL	953	088	I
RV24A-H8	SAFETY-RELIEF	DRYWELL	939	032	I
RV24A-NS1	SAFETY-RELIEF	DRYWELL	952	050	I
RV24A-NS2	SAFETY-RELIEF	DRYWELL	952	055	I
RV24A-N1	SAFETY-RELIEF	DRYWELL	956	086	I
RV25-H1	SAFETY-RELIEF	DRYWELL	953	180	I
RV25-HIA	SAFETY-RELIEF	DRYWELL	953	180	I
RV25-H2	SAFETY-RELIEF	DRYWELL	948	190	I
RV25-H2A	SAFETY-RELIEF	DRYWELL	948	190	I
RV25-H3	SAFETY-RELIEF	DRYWELL	934	180	I
RV25-NS1	SAFETY-RELIEF	DRYWELL	952	160	I
RV25-NS2	SAFETY-RELIEF	DRYWELL	952	195	I
RV25-N2	SAFETY-RELIEF	DRYWELL	956	159	I
RV25A-H2	SAFETY-RELIEF	DRYWELL	945	120	I
RV25A-H2A	SAFETY-RELIEF	DRYWELL	945	120	I
RV25A-H7	SAFETY-RELIEF	DRYWELL	953	135	I
RV25A-NS1	SAFETY-RELIEF	DRYWELL	934	110	I
RV25A-NS2	SAFETY-RELIEF	DRYWELL	934	102	I
RV25A-NS3	SAFETY-RELIEF	DRYWELL	952	122	I
RV26-H1	SAFETY-RELIEF	DRYWELL	953	200	I
RV26-H1A	SAFETY-RELIEF	DRYWELL	953	200	I
RV26-H2	SAFETY-RELIEF	DRYWELL	947	200	I
RV26-H2A	SAFETY-RELIEF	DRYWELL	947	200	I
RV26-H3A	SAFETY-RELIEF	DRYWELL	935	200	I
RV26-N1	SAFETY-RELIEF	DRYWELL	956	200	I
RV26A-H2	SAFETY-RELIEF	DRYWELL	940	250	I
RV26A-H2A	SAFETY-RELIEF	DRYWELL	935	250	I
RV26A-NS1	SAFETY-RELIEF	DRYWELL	934	240	I
RV26A-NS2	SAFETY-RELIEF	DRYWELL	934	230	I
RV26A-N1	SAFETY-RELIEF	DRYWELL	950	250	I
RV26A-N2	SAFETY-RELIEF	DRYWELL	951	250	I
RV27-H1	SAFETY-RELIEF	DRYWELL	950	320	I
RV27-H1A	SAFETY-RELIEF	DRYWELL	950	230	I
RV27-H5	SAFETY-RELIEF	DRYWELL	945	270	I

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	ACCESSIBLE -A INACCESSIBLE-I
RV27-H6	SAFETY-RELIEF	DRYWELL	945	270	I
RV27-NS1	SAFETY-RELIEF	DRYWELL	934	250	I
RV27-NS2	SAFETY-RELIEF	DRYWELL	934	280	I
RV27-N1	SAFETY-RELIEF	DRYWELL	956	270	I
RV27A-H2A	SAFETY-RELIEF	DRYWELL	953	290	I
RV27A-H3	SAFETY-RELIEF	DRYWELL	953	290	I
RV27A-H9	SAFETY-RELIEF	DRYWELL	938	290	I
RV27A-NS1	SAFETY-RELIEF	DRYWELL	952	282	I
RV27A-NS2	SAFETY-RELIEF	DRYWELL	952	279	I
RV27A-NS3	SAFETY-RELIEF	DRYWELL	952	282	I
RV27A-N1	SAFETY-RELIEF	DRYWELL	956	270	I
R26-NS1	SAFETY-RELIEF	DRYWELL	952	200	I
SS-1	MAIN STEAM	DRYWELL	953	279	I
SS-1AR	RECIRCULATION	DRYWELL	922	315	I
SS-1BR	RECIRCULATION	DRYWELL	922	135	I
SS-11	FEEDWATER	DRYWELL	952	302	I
SS-12	FEEDWATER	DRYWELL	952	058	I
SS-13	FEEDWATER	DRYWELL	952	258	I
SS-14	FEEDWATER	DRYWELL	952	096	I
SS-17A	RHR	DRYWELL	964	072	I
SS-17B	RHR	DRYWELL	964	072	I
SS-18A	RHR	DRYWELL	964	288	I
SS-18B	RHR	DRYWELL	964	288	I
SS-19	RHR	DRYWELL	964	341	I
SS-2	MAIN STEAM	DRYWELL	953	081	I
SS-2AR	RECIRCULATION	DRYWELL	927	302	I
SS-2BR	RECIRCULATION	DRYWELL	927	122	I
SS-20	RHR	DRYWELL	964	019	I
SS-3	MAIN STEAM	DRYWELL	950	212	I
SS-3AR	RECIRCULATION	DRYWELL	927	328	I
SS-3BR	RECIRCULATION	DRYWELL	927	148	I
SS-4	MAIN STEAM	DRYWELL	950	148	I
SS-4AR(A)	RECIRCULATION	DRYWELL	934	302	I
SS-4AR(B)	RECIRCULATION	DRYWELL	934	323	I
SS-4BR(A)	RECIRCULATION	DRYWELL	934	120	I
SS-4BR(B)	RECIRCULATION	DRYWELL	934	149	I
SS-40	HPCI	MAIN STEAM CHASE			I
SS-5AR	RECIRCULATION	DRYWELL	941	315	I
SS-5BR	RECIRCULATION	DRYWELL	941	135	I
SS-6AR	RECIRCULATION	DRYWELL	953	261	I
SS-6BR	RECIRCULATION	DRYWELL	953	099	I
SS-7	MAIN STEAM	DRYWELL	953	240	I
SS-7AR	RECIRCULATION	DRYWELL	953	323	I
SS-7BR	RECIRCULATION	DRYWELL	953	032	I
SS-8	MAIN STEAM	DRYWELL	953	120	I
SS-8AR	RECIRCULATION	DRYWELL	927	270	I
SS-8BR	RECIRCULATION	DRYWELL	927	090	I

TABLE 3.6.1
SAFETY RELATED HYDRAULIC SNUBBERS

SNUBBER NO.	SYSTEM	LOCATION	ELEVATION	AZIMUTH (AIRLOCK 0 REF)	ACCESSIBLE -A INACCESSIBLE-I
SS-121	PCAC	TORUS ROOM	928	252	A
SS-184	PCAC	TORUS ROOM	928	261	A
SS-185	CGCS	TORUS ROOM	928	260	A
SS-187	CGCS	TORUS ROOM	928	264	A
SS-191	PCAC	TORUS ROOM	928	234	A
SS-21	RHR	TORUS FL LV - S WALL			A
SS-211	CGCS	TORUS ROOM	931	072	A
SS-22	RHR	TORUS FL LV - S WALL			A
SS-23	RHR	B RHR ROOM FL LV			A
SS-24	RHR	A RHR ROOM FL LV			A
SS-25	RHR	TORUS CATWK-SE WALL			A
SS-26	CORE SPRAY	B RHR ROOM FL LVL			A
SS-27	CORE SPRAY	B RHR ROOM FL LVL			A
SS-28A	CORE SPRAY	A RHR ROOM FL LVL			A
SS-28B	CORE SPRAY	A RHR ROOM FL LVL			A
SS-29	RHR	OVER N2 ANALYZER	954		A
SS-30	RHR	OVER N2 ANALYZER	954		A
SS-31	RHR	TORUS CATWK			A
SS-32A	RHR	A RHR ROOM - BY HX	916		A
SS-32B	RHR	A RHR RODM - BY HX	916		A
SS-33	RHR	ABOVE TORUS			A
SS-34	RHR	ABOVE TORUS			A
SS-35	HPCI	HPCI ROOM - N WALL	912		A
SS-36A	HPCI	HPCI ROOM - FL LVL			A
SS-36B	HPCI	HPCI ROOM - FL LVL			A
SS-37	HPCI	HPCI ROOM - W WALL	905		A
SS-38A	RCIC	RCIC ROOM - W WALL	906		A
SS-38B	RCIC	RCIC ROOM - W WALL	906		A
SS-41	CORE SPRAY	ABOVE TORUS CATWK	927		A
SS-42	HPCI	ABOVE TORUS RING HDR	906		A

3.0 LIMITING CONDITIONS FOR OPERATION

B. Fire Suppression Water System

1. Except as specified in 3.13.B.2 or 3.13.B.3 below, the system shall be operable at all times with:

   a. The following pumps, including automatic initiation logic, operable and capable of delivering at least 1500 gpm at a discharge pressure of 90 psig:

      1. Diesel-driven fire pump
      2. Motor-driven fire pump
      3. Screen wash/fire pump

   b. An operable flow path capable of taking suction from the river and transferring the water through distribution piping with operable sectionalizing control or isolation valves to the yard hydrant valves and the first valve ahead of each hose station or sprinkler system required to be operable.

4.0 SURVEILLANCE REQUIREMENTS

B. Fire Suppression Water System

1. The system shall be verified operable as follows:

   a. Operability of the diesel-driven fire pump starting battery shall be demonstrated by:

      1. Once each week verify electrolyte level and voltage is within specifications.

      2. Once every three months verify the specific gravity of each cell is within specifications.

      3. Once every 18 months inspect the batteries, battery racks, and electrical connections for damage or abnormal deterioration.

   b. 1. The motor-driven fire pump shall be started every month and run for at least 15 minutes on recirculation flow.

      2. The screen wash/fire pump shall be run for at least 15 minutes per month.

   c. The diesel-driven fire pump shall be started every month from ambient conditions and run for at least 20 minutes on recirculation flow.

   d. The level in the diesel-driven fire pump day tank shall be checked every month and verified to contain at least 65 gallons of fuel.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
2. With one of the pumps required by Specification 3.13.B.1.a inoperable, perform the Surveillance required by Specification 4.13.B.2 and restore the inoperable pump to operable status within seven days or provide a 30-day written report outlining the plans and procedures to be used to provide for the loss of redundancy in the Fire Suppression Water system.	e. Every three months verify that a sample of fuel from the diesel oil storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water, and sediment.
3. With the fire suppression water system otherwise inoperable: a. Establish a backup fire Suppression Water System within 24 hours. b. Provide prompt notification with a written followup report within 14 days outlining the actions taken and the plans and schedule for restoring the system to operable status.	f. Every 18 months subject the diesel-driven fire pump engine to an inspection in accordance with procedures prepared in conjunction with the manufacturer's recommendations for this class of standby service. g. A simulated automatic actuation of each fire pump and the screen wash/fire pump, including verification of pump capability, shall be conducted every 18 months. h. The yard main and the reactor building and turbine building headers shall be flushed every 12 months. i. System flow tests shall be performed every three years. j. Valves in flow paths supplying fire suppression water to safety related structures, systems, and component shall be cycled every 12 months.

3.13 BASES:

Elements of the fire detection and protection system are required to be operable to protect safety related structures, systems, and components whenever those structures, systems, or components are required to be operable. Fire detection and protection systems will normally be maintained operable at all times except for periods of maintenance and testing.

Fire detection instrumentation is installed throughout the plant to protect safety related structures, systems, and components. The detectors in each area initiate a local alarm and an alarm in the control room. All circuits are supervised and the installation meets the requirements of NFPA-72D. The Specifications require all detectors to be operable in those zones having only one detector (battery rooms). In other plant areas, Table 3.13.1 permits one detector in each zone to be inoperable. If more detectors are inoperable, a patrolling fire watch is established in the affected area until the required number of detectors are restored to operable status. The loss of one detector does not significantly degrade the ability to detect fires in areas of the plant having multiple detectors.

The fire suppression water system is supplied by three identical vertical centrifugal pumps rated at 1500 gpm at 100 psig each. Two of these pumps are motor driven and one is diesel driven . One of the motor driven pumps normally supplies the needs of the screen wash system and is designated the screen wash/fire pump. Transfer from screen wash duty to fire duty occurs automatically. All pumps are started automatically by instrumentation sensing header pressure. Any two pumps are capable of supplying all fire fighting water requirements in safety related areas of the plant. If a pump is inoperable, it must be repaired within seven days or a report is submitted to the Commission. If two pumps are inoperable, or if other circumstances interrupt the supply of water to any safety related area, a backup source of water must be provided within 24 hours and the Commission notified.

Automatic sprinkler systems are installed in both diesel-generator rooms and both day tank rooms. Other sprinkler and deluge systems are installed in turbine lube oil piping and storage areas and other non-safety related portions of the plant. An automatic Halon suppression system is installed in the cable spreading room. Inoperability of any of the automatic suppression systems in safety related areas of the plant requires the stationing of a continuous fire watch in the area equipped with backup manual fire suppression equipment. Hose stations and yard hydrant hose houses are provided in all safety related areas of the plant and surrounding all principal plant buildings. These stations are supplied from the fire suppression water system. If the water supply to these areas is interrupted, a hose supplied from an operable source is made available to protect the area having the inoperable station.

Piping and electrical penetrations are provided with seals where required by the fire severity. If a seal is made or found to be inoperable for any reason, the penetration area is continuously attended until an effective fire seal is restored. Seals have been qualified for the maximum fire severity present on either side of the barrier.

4.13 BASES:

Fire detectors are tested in accordance with the manufacturer's recommendations. All tests and inspections are performed by the plant staff. Every six months each detector is functionally tested. Combustion generated smoke is not used in these tests. Alarm circuits are functionally checked every six months. In addition, all circuitry is automatically supervised for open wiring and ground faults.

Fire pumps are tested each month to verify operability. Test starting of the screen wash/fire pump is not required since it is normally in service. Each fire pump is manually started and operated for at least 15 minutes with pump flow directed through the recirculation test line. Every 18 months the operability of the automatic actuation logic for the fire pumps and the screen wash/fire pump is verified and the performance of each pump is verified to meet system requirements. The specified flush and valve checks provide assurance that the piping system is capable of supplying fire suppression water to all safety related areas. When one of the fire pumps is inoperable, the operable pumps are run daily to verify operability until all pumps are once again available.

A system flow test is specified every three years. This test verifies the hydraulic performance of the fire suppression fire water header system. The testing will be performed using Section 11, Chapter 5 of the Fire Protection Handbook, 14th Edition, as a procedural guide. This test is generally performed in conjunction with a visit from insurance company inspectors.

Hose stations and yard hydrant hose houses are inspected monthly to verify that all required equipment is in place. Gaskets in hose couplings are inspected periodically and the hose is pressure tested. Pressure testing of outdoor hose is conducted more frequently than indoor hose because of the less favorable storage conditions. Operability of hose station isolation valves is verified every three years by partially opening each valve to verify flow. All of these tests provide a high degree of assurance that each hose station and yard hydrant hose house will perform satisfactorily after periods of standby service.

Simulated automatic actuation tests are conducted each 18 months to confirm the operability of the sprinkler and Halon systems. These tests consist of verification that all valves, dampers (Halon system only), alarms, and flow paths are functional.

Plant fire barrier walls are provided with seals for pipes and cables where necessary. Where such seals are installled, they must be maintained intact to perform their function. Visual inspection of each installed seal is required every 18 months and after seal repair. A visual inspection following repair of a seal is sufficient to assure that seal integrity will be within acceptable limits.