ATTACHMENT A

Technical Specification Change to
Core Spray Sparger Break Detection Setpoint

References: a) IE Circular No. 79-24 titled, "Proper Installation and Calibration of Core Spray Pipe Break Detection Equipment on BWR's"

b) General Electric Service Information Letter No. 300, titled "Instrumentation for Core Spray Sparger Line Break Detection"

Proposed Change

Reference is made to PNPS Technical Specification Appendix A, Table 3.2.B "Instrumentation that Initiates or Controls the Core and Containment Cooling Systems". The desired change consists of replacing the current Core Spray Sparger to Reactor Pressure Vessel d/p Trip Level Setting of 5(± 1.5) psid with a new setpoint of ≤ .5 psid.

Reason for Change

IE Circular 79-24 identified potential problems related to BWR core spray pipe break detection systems and recommended that licensees investigate this problem as it relates to setpoint, function, and installation of the d/p instruments used as monitors for core spray pipe breaks. As a result of our investigation and recommendations by General Electric Company, it was determined that the current alarm setpoint on instruments used to measure core spray differential pressure should be changed to take into account the effect of density changes of the water in the pressure leg connections.

Safety Considerations

Changing the core spray differential pressure switch setpoint from 5(± 1.5) psid to -1.0 (± 1.5) psid with a limit of 0.5 psid increasing, takes into consideration the density fluctuations experienced in the instrument reference leg during normal rated power operation not previously accounted for in setpoint determination. Actual instrument readings including reference leg density considerations together with a predicted 5.3 psid increase in differential pressure following a core spray header break (Reference b) provide the basis and justification for this change.

This change has been reviewed by the Nuclear Safety Review and Audit Committee and has been reviewed and approved by the Operations Review Committee.

Schedule of Change

Approval of this change is requested prior to startup from our current refueling outage.

Boston Edison Company proposes that pursuant to 10 CFR Part 170 this is a Class II Amendment.

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PNPS

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
1	RHR (LPCI) Trip System bus power monitor	NA	Monitors availability of power to logic systems.
1	Core Spray Trip System bus power monitor	NA	Monitors availability of power to logic systems.
1	ADS Trip System bus power monitor	NA	Monitors availability of power to logic systems and valves.
1	HPCI Trip System bus power monitor	NA	Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	NA	Monitors availability of power to logic systems.
2	Recirculation Pump A d/p	$\leq$2 psid	Operates RHR (LPCI) break detection logic which directs cooling water into unbroken recirculation loop.
2	Recirculation Pump B d/p	$\leq$2 psid	
2	Recirculation Jet Pump Riser d/p A>B	0.5<p<1.5 psid	
1	Core Spray Sparger to Reactor Pressure Vessel d/p	$\leq$0.5 psid	Alarm to detect core spray sparger pipe break.

ATTACHMENT B

Technical Specification Change Concerning Hydraulic Snubbers

Proposed Change

Reference is made to PNPS Technical Specification Appendix A, Page 137 . The desired change consists of removing Snubber No. SS-23-3-33 (HPCI) from the table of Safety Related Shock Suppressors (Snubbers).

Reason For Change

Per IE Bulletin #79-14, the pipe stress reanalysis performed by Bechtel on the HPCI piping system has shown that this snubber is not required to protect the HPCI system piping. Thus removal of this snubber was recommended by Bechtel to ensure that the proper design configuration is maintained in accordance with the reanalysis.

Safety Considerations

Removal of this snubber from the HPCI piping system does not decrease the level of safety as originally designed to provide protection from structural damage to the piping as a result of a seismic or other event initiating dynamic loads. Through pipe stress reanalysis, the remaining snubbers have been determined to be of correct size and location to ensure proper system rigidity for the prevention of unrestrained pipe motion under dynamic loads.

This change has been reviewed by the Nuclear Safety Review and Audit Committee and reviewed and approved by the Operations Review Committee

Schedule of Change

Approval of this change is requested prior to startup from our current refueling outage.

Boston Edison Company proposes that pursuant to 10 CFR 170 this is a Class II amendment.

Attachment

Table 3.6.1

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Snubber in High Radiation Area During Shutdown	Snubbers Especially Difficult to Remove	Snubbers Inaccessible During Normal Operation	Snubbers Accessible During Normal Operation
SS-6-10-1	Feedwater System	42'			X (Drywell)	
SS-6-10-2	Feedwater System	42'			X (Drywell)	
SS-6-10-3	Feedwater System	42'			X (Drywell)	
SS-6-10-4	Feedwater System	42'			X (Drywell)	
SS-6-10-5	Feedwater System	42'			X (Drywell)	
SS-13-3-1	RCIC	38'			X (Drywell)	
SS-13-3-2	RCIC	38'			X (Drywell)	
SS-14-3-1	Core Spray	65'			X (Drywell)	
SS-14-3-2	Core Spray	65'			X (Drywell)	
SS-14-3-3	Core Spray	65'			X (Drywell)	
SS-14-3-4	Core Spray	65'			X (Drywell)	
SS-23-10-1	H.P.C.I.	42'			X (Drywell)	
SS-23-10-2	H.P.C.I.	42'			X (Drywell)	
S-23-3-30	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-3-31	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-10-32	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-10-34	H.P.C.I.	-6'				X H.P.C.I. Quadrant
S-23-10-35	H.P.C.I.	-6'				X H.P.C.I. Quadrant
S-23-3-36	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-23-3-37	H.P.C.I.	-3'09"				X H.P.C.I. Quadrant
S-10-3-43	RHR	-3'06"				X RHR Pump Room
S-10-20-44	RHR	-3'06"				X RHR Pump Room
S-30-3-45	RBCCW	83'5"				X Reactor Building
S-10-10-46	RHR	6"				X Torus Compartment

Modifications to this Table due to changes in high radiation areas should be submitted to the NRC as part of the next license amendment.

ATTACHMENT C

Supplement to August 24, 1977 "Proposed
Electrical Power Systems Technical Specifications"

Proposed Change

Reference is made to BECo letter (J. E. Howard) to NRC (D. K. Davis) dated August 24, 1977, Attachment A (proposed Tech. Specs.), Page 194A. The proposed change consists of supplementing our original proposal with additional surveillance requirements on the Shutdown transformer.

Reason for Change

During telephone conversations on February 4, 6 and 7, 1980, members of your staff involved in the review of the above stated correspondence requested Boston Edison Company to supplement the proposed August 1977 technical specification with a section which would ensure surveillance testing for the timers on the shutdown transformer. Enclosure A provides this supplement.

Safety Considerations

This testing on the shutdown transformer does not present any hazard considerations not described or implied in the license application as amended, as this testing has always been in effect via station procedures.

This change has been reviewed by the Nuclear Safety Review and Audit Committee and reviewed and approved by the Operations Review Committee.

Schedule of Change

This change will be put into effect upon receipt of approval from the Commission

Boston Edison Company proposes that no submittal fee is required since this is a supplement to changes submitted prior to implementation of 10 CFR 170 and is made at the request of the Commission.

Attachment

1. Verifying de-energization of the emergency buses and load shedding from the emergency buses.

2. Verifying the diesel starts from ambient condition on the auto-start signal energizes the emergency buses with permanently connected loads, energizes the auto-connected emergency loads through the load sequence and operates for ≥ 5 minutes while its generator is loaded with the emergency loads.

   The results shall be logged.

C. Once per operating cycle with the diesel loaded per 4.9.A.1.b verify that on diesel generator trip secondary (off-site) a-c power is automatically connected to the emergency service buses and emergency loads are energized through the load sequencer in the same manner as described in 4.9.A.1.b.1.

   The results shall be logged.

2. Secondary Off-Site Power

A. A test will be performed once per operating cycle to verify that the shutdown transformer breakers will close on to the safety related buses within 12 to 14 seconds.

August 1977

March 1980

ATTACHMENT D

Technical Specification Concerning Modifications To Safety Relief Valves

Ref. (a) USNRC Letter (B. K. Grimes) to All BWR Licensees dated July 26, 1979

(b) NUREG-0462

Proposed Change

Reference is made to PNPS Technical Specifications, Appendix A, Pages 126, 127 & 146. The desired changes consist of deleting the requirement associated with the relief/safety valve bellows (as shown in Exhibit A), since this feature will be removed from the valves during our current refueling outage.

Reason for Change

The Target Rock SRV will be modified this outage in that the 3 stage topworks will be replaced with 2 stage topworks, thereby eliminating the bellows and its associated pilot valve leakage. This in turn creates the need to remove valve bellows operability requirements from the Technical Specifications.

Safety Considerations

Replacing the original 3 stage pilot operated activator with a redesigned 2 stage pilot operated actuator is consistent with General Electric's long term program to reduce the likelihood of failures of BWR safety relief valves, as described and recommended in a letter from B. K. Grimes to all Boiling Water Reactor Licensees, dated July 16, 1979. Reference (a).

This change has been reviewed and approved by the Operations Review Committee and reviewed by the Nuclear Safety Review and Audit Committee.

Schedule of Change

Approval of this change is requested prior to startup from our current refueling outage.

Boston Edison Company proposes that pursuant to 10 CFR Part 170 this is a Class II amendment.

Attachments

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.C Coolant Chemistry (Cont'd)	4.6
power operation is permissible only during the succeeding seven days.	
3. If the conditions in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	
D. Safety and Relief Valves	D. Safety and Relief Valves
1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than 104 psig and temperature greater than 340°F, both safety valves and the safety modes of all relief valves shall be operable.	1. At least one safety valve and two relief/safety valves shall be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles. The set point of the safety valves shall be as specified in Specification 2.2.
	2. At least one of the relief/safety valves shall be disassembled and inspected each refueling outage.
3. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENT
3.6.D Safety and Relief Valves (Cont'd)	
pressure shall be below 104 psig within 24 hours.	
E. Jet Pumps	E. Jet Pumps
1. Whenever the reactor is in the startup or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	Whenever there is recirculation flow with the reactor in the startup or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously: 1. The two recirculation loops have a flow imbalance of 15% or more when the pumps are operated at the same speed. 2. The indicated value of core flow rate varies from the value derived from loop flow measurements by more than 10%. 3. The diffuser to lower plenum differential pressure reading on an individual jet pump varies from established jet pump P characteristics by more than 10%.
F. Jet Pump Flow Mismatch	F. Jet Pump Flow Mismatch
1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than or equal to 80%.	Recirculation pump speeds shall be checked and logged at least once per day.
G. Structural Integrity	G. Structural Integrity
1. The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1974	The nondestructive inspections listed in Table 4.6.1 shall be performed as specified. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with AEC.

(DELETED)

ATTACHMENT E

Technical Specification Change to ECCS
High Drywell Pressure Setpoint

Proposed Change

Reference is made to PNPS Technical Specifications, Appendix A, pages 27, 45 and 48 The desired change consists of raising the High Drywell Pressure trip level setting from ≤2 psig to ≤2.5 psig.

Reason for Change

Current Technical Specifications require that a Δ P between the drywell and suppression chamber be maintained at equal to or greater than 1.50 psid. This requirement coupled with the High Drywell Pressure trip setting of ≤2 psig, impose operational hardships that could be lessened by raising this setpoint to ≤2.5 psig.

Safety Considerations

The ECCS high drywell setpoint was originally set a) low enough to give proper safety system response, and b) high enough to avoid spurious safety system activation (Ref: FSAR Section 7.4 and Bases for PNPS Technical Specification 3.2). Accident analyses (e.g. LOCA Analysis, NED 21696 and FSAR Amendment #20) have been performed with a setpoint of 2.0 psig. These analyses assumed initiation of an accident with a unpressurized drywell followed by analyzed pressurization to 2.0 psig at which time ECCS actuation is initiated.

Mark I Containment studies and resultant analyses have resulted in a Technical Specification requirement of normal drywell pressurization of 1.5 psi above the containment wetwell which is nominally atmospheric. This has resulted in the normal operating pressure of the drywell being substantially closer to the ECCS set point. It is therefore desireable to increase the drywell Technical Specification setpoint to 2.5 psig to reduce the potential for spurious ECCS actuation.

The initiation time of the ECCS system depends primarily on the rate of pressure rise in the drywell and the pressure difference between the setpoint and normal operation. Since the first parameter is relatively unaffected by the initial drywell pressure and since the proposed setpoint change will result in a pressure difference between normal operation and the setpoint less than that used in previous LOCA analyses, the change will result in a more responsive performance of ECCS Systems following drywell pressurization than in the original analyses.

This change has been reviewed by the Nuclear Safety Review and Audit Committee and has been reviewed and approved by the Operations Review Committee.

Schedule of Change

Approval of this change is requested prior to startup from our current refueling outage.

Boston Edison Company proposes that pursuant to 10CFR Part 170 this is a Class II Amendment.

Attachments

REACTOR PROTECTION SYSTEM (SCRAM) . INSTRUMENTATION REQUIREMENT

Minm. Number Operable Inst. Channels per Trip (1) System	Trip Function	Trip Level Setting	Modes in Which Function Must Be Operable			Action (1)
			Refuel (7)	Startup/Hot Standby	Run	
1	Mode Switch in Shutdown		X	X	X	A
1	Manual Scram		X	X	X	A
	IRM					
3	High Flux	≤ 120/125 of full scale	X	X	(5)	A
3	Inoperative		X	X	(5)	A
	APRM					
2	High Flux	* (14) (15)	(17)	(17)	X	A or B
2	Inoperative		X	X(9)	X	A or B
2	Downscale	≥ 2.5 Indicated on Scale	(11)	(11)	X(12)	A or B
2	High Flux (15%)	≤ 15% of Design Power	X	X	(16)	A or B
2	High Reactor Pressure	≤ 1085 psig	X(10)	X	X	A
2	High Drywell Pressure	≤ 2.5 psig	X(8)	X(8)	X	A
2	Reactor Low Water Level	≥ 9 In. Indicated Level	X	X	X	A
2	High Water Level in Scram Discharge Tank	≤ 39 Gallons	X(2)	X	X	A
2	Turbine Condenser Low Vacuum	≥ 23 In. Hg Vacuum	X(3)	X(3)	X	A or C
2	Main Steam Line High Radiation	≤ 7X Normal Full Power Background	X	X	X	A or C
4	Main Steam Line Isolation Valve Closure	≤ 10% Valve Closure	X(3) (6)	X(3) (6)	X(6)	A or C
2	Turb. Cont. Valve Fast Closure	≥ 150 psig Control Oil Pressure at Acceleration Relay	X(4)	X(4)	X(4)	A or D
4	Turbine Stop Valve Closure	≤ 10% Valve Closure	X(4)	X(4)	X(4)	A or D

*APRM high flux scram setpoint $\leq (.65W + 55)\left[\frac{A}{MTPF}\right]$ Two recirc. pump operation

PNPS

TABLE 3.2.A

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum # of Operable Instrument Channels Per Trip System (1)	Instrument	Trip Level Setting	Action (2)
2(7)	Reactor Low Water Level	≥9" indicated level (3)	A and D
1	Reactor High Pressure	≤110 psig	D
2	Reactor Low-Low Water Level	at or above -49 in. indicated level (4)	A
2	Reactor High Water Level	≤48" indicated level (5)	B
2(7)	High Drywell Pressure	≤2.5 psig	A
2	High Radiation Main Steam Line Tunnel	≤7 times normal rated full power background	B
2	Low Pressure Main Steam Line	≥880 psig (8)	B
2(6)	High Flow Main Steam Line	≤140% of rated steam flow	B
2	Main Steam Line Tunnel Exhaust Duct High Temperature	≤170°F	B
2	Turbine Basement Exhaust Duct High Temperature	≤150°F	B
1	Reactor Cleanup System High Flow	≤300% of rated flow	C
2	Reactor Cleanup System High Temperature	≤150°F	C

PNPS

TABLE 3.2.B (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum # of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Remarks
2	High Drywell Pressure	$\leq$2.5 psig	1. Initiates Core Spray; LPCI; HPCI. 2. In conjunction with Low-Low Reactor Water Level, 120 second time delay and LPCI or Core Spray pump running, initiates Auto Blowdown (ADS). 3. Initiates starting of Diesel Generators.
1	Reactor Low Pressure	400 psig $\pm$ 25	Permissive for Opening Core Spray and LPCI Admission valves.
1	Reactor Low Pressure	$\leq$110 psig	In conjunction with PCIS signal permits closure of RHR (LPCI) injection valves.
1	Reactor Low Pressure	400 psig $\pm$ 25	In conjunction with Low-Low Reactor Water Level initiates Core Spray and LPCI.
2	Reactor Low Pressure	900 psig $\pm$ 25	Prevents actuation of LPCI break detection circuit.

ATTACHMENT F

PROPOSED TECHNICAL SPECIFICATIONS
FOR
PILGRIM NUCLEAR POWER STATION
ATWS RPT/ARI

PROPOSED CHANGES

Page 44a Add a new LCO as follows:

"G. Recirculation Pump Trip/Alternate Rod Insertion Initiation

Whenever the reactor is in the RUN mode, the limiting conditions for operation for the instrumentation listed in Table 3.2.G shall be met."

Add a new Surveillance Requirement as follows:

"G. Recirculation Pump Trip/Alternate Rod Insertion

Surveillance for instrumentation which initiates Recirculation Pump Trip and Alternate Rod Insertion shall be as specified in Table 4.2-G."

Page 59a - Insert a new page as shown in Exhibit B. This page includes Table 3.2.G and notes pertaining to the table.

Page 66a - Insert an additional item in table 4.2.G as shown in Exhibit B which provides RPT/ARI instrumentation surveillance requirements.

Page 67 - Modify notes to include Table 4.2.G. *

Page 73 Insert the paragraphs shown in Exhibit B which state the bases for the recirculation pump trip and alternate rod insertion systems and the limiting conditions for operation. Additionally three references are added.

Page 77 - Insert the paragraph shown in Exhibit B which states the bases for the surveillance frequency of the recirculation pump trip and alternate rod insertion instrumentation.

REASON FOR CHANGES

Numerous studies have been performed to assess the probability and consequences of the failure to scram immediately following an abnormal operational transient. In volume 3 of NUREG 0460, the Staff proposed requirements including installation of a reciculation pump trip system as a means of substantially reducing maximum reactor vessel pressure in the unlikely event of a failure to scram. The proposed technical specifications for the new trip system are modeled after the specifications for existing protective instrumentation. In addition, an alternate rod insertion system has been proposed to provide a diverse means for initiation of control rod insertion.

SAFETY CONSIDERATIONS

A safety evaluation for the ATWS Recirculation Pump Trip and Alternate Rod Insertion systems ATWS RPT/ARI was submitted on the docket for the Monticello Nuclear Generating Plant (Docket No. 50263, License No. DPR-22). This evaluation was reviewed by the NRC Staff and a favorable Safety Evaluation Report was issued on February 23, 1977 for the RPT function. Since the RPT/ARI systems proposed for the Pilgrim Nuclear Power Station are essentially identical to that described in the Monticello evaluation only the minor differences and facets unique to the technical specifications will be considered in depth here.

The proposed limiting condition for operation requires the recirculation pump trip system to be operable when the reactor is in the RUN mode and the alternate rod insertion system to be operable in all modes except REFUELING. Since the capacity of the safety/relief valves is far in excess of the steam generation rate achievable in any other mode, there is no potential for vessel overpressurization in modes other than RUN. Restricting the LCO to the RUN mode for the RPT function is therefore appropriate.

The proposed operability requirements are similar to those of like systems. These requirements were assumed in the design and reliability analysis of the trip system.

The proposed surveillance requirements incorporate the fact that analog transmitters are used in ATWS RPT/ARI systems. These devices are a new, improved line of BWR instrumentation. The calibration frequency is therefore proposed to be once per operating cycle which is consistent with both the equipment capabilities and the requirements for similar equipment used by other reactor vendors. The calibration frequency for the trip units is proposed to be quarterly, the same as other similar protective instrumentation. Likewise, the test frequency is specified as monthly like that of other protective instrumentation. A sensor check is proposed once per day; this is considered to be an appropriate frequency, commensurate with the design applications and the fact that the recirculation pump trip/alternate rod insertion systems are backups to existing protective instrumentation.

With the implementation of the above proposed technical specification changes, there is adequate assurance that the ATWS RPT/ARI systems will perform to provide the intended plant protection in the extremely low probability of a plant transient with a failure to scram.

As discussed above, the ATWS RPT/ARI systems proposed are essentially identical with the systems as described on the Monticello docket. Several changes have been made to improve the system which are described below:

The Monticello ATWS RPT design as approved by the NRC Staff was not coupled with an ARI system. The ARI system design is; however, identical with the ARI design provided by Monticello in their safety evaluation report, with the exceptions noted below:

The Monticello RPT design includes a "Manual Initiation" push button on the operator control console. The proposed RPT/ARI design removes this push button but does provide manual control of the ARI function from the operator control console. Manual initiation of RPT at the console is unnecessarily redundant due to the variety of means already available to the operator for manually tripping the recirculation pumps or otherwise reducing recirculation flow.

The addition of the ARI function results in additional crowding of the operator control console. In order to reduce this crowding the manual reset push buttons have been eliminated and automatic reset logic substituted. Although the Monticello RPT design included a seal-in logic with manual reset it is unncessarily redundant. Once a trip signal actuates the field breakers, it can be removed without affecting the state of the field breakers. The field breakers must be manually reset. Therefore, the automatic reset feature only reduces the manual actions required to reset the pumps for operation and does not affect the trip function. The ARI automatic reset logic includes a seal-in logic for a 30 second interval to assure sufficient time to blow down the pilot air header and insure complete rod insertion. The automatic reset cannot function, however, if the trip signal is still present. In this case an additional 30 seconds of delay will occur before reset and this sequence will continue until the trip signal is removed.

The high pressure setpoint for ATWS RPT/ARI as proposed is higher than the specification for the Monticello RPT. With the current plant configuration initiation of the ATWS RPT function is predicted during certain pressurization transients if the ATWS RPT setpoint is not raised. Since the initiation of ATWS RPT causes an increase in severity of the transients this is an undesirable condition. Raising the setpoint decouples the more frequent pressurization transients from ATWS RPT effects. With the proposed setpoint, the only events which will initiate ATWS RPT are the turbine/generator trip with bypass failure and the ASME overpressure protection event (MSIV closure with trip scram failure). The turbine/generator trip events will not result in exceeding the vessel pressure limit despite the increased severity due to ATWS RPT initiation and the limiting event will remain the MSIV closure with trip scram failure.

The MSIV closure event, with ATWS RPT, has a margin of 34 psi to the ASME code limit with the higher setpoint of 1160 psig.

Increasing the ATWS RPT setpoint will also affect the peak pressure during an ATWS event. The selected setpoint will cause an increased peak pressure but it is expected to result in less than 25 psi increase.

The qualified DC power supplies and DC to AC inverter specified for the ATWS RPT/ARI systems are not currently available. Replacement of the DC power supplies and elimination of the inverter are necessary changes pending resolution of the supply difficulty. The design of each ATWS cabinet includes two qualified 24 VDC power supplies. One power supply is sourced by offsite 115 VAC which transfers to onsite generator power in the event of loss of offsite power. The second power supply is sourced by the station batteries (125 VDC) through an inverter (125 VDC to 115 VAC). The uninterruptible power supplied by the inverter allows the system logic to remain energized during the event of loss of offsite power until the transfer to onsite generators occurs. The temporary modifications will require manual action to initiate a pump trip and ARI during a loss of offsite power event.

In the unlikely event of loss of offsite power, a trip of the recirculation pumps and the ARI function can be initiated without the inverter. A pump trip occurs when the offsite AC power is interrupted deenergizing the motor of the recirculation pump MG sets. The ARI function can be initiated manually by the operator following an alarm from the ATWS loss of power annunciator. The ARI manual trip utilizes power from the 125 VDC station batteries to actuate the solenoid-operated instrument air valves.

A failure of one or both power supplies alerts the operator via the annunciator. He can utilize the MG manual controls to trip the recirculation pumps. The ARI can be manually initiated as explained above.

In order to maintain the diversity of the original design the following steps are proposed:

1. The ATWS cabinets would include provisions for installation of the inverters at a later date.

2. The ATWS cabinets would include non-safety qualified power supplies to be replaced or qualified later.

The changes which have been incorporated into the proposed design do not change the basic functions of the ATWS systems as described for the Monticello RPT/ARI system.

These changes have been reviewed by the Nuclear Safety Review and Audit Committee and reviewed and approved by the Operations Review Committee.

SCHEDULE OF CHANGE

Approval of this change is requested prior to startup from our current refueling outage.

Boston Edison Company proposes that pursuant to 10 CFR Part 170 this is a Class II Amendment.

Attachments

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
G. Recirculation Pump Trip/Alternate Rod Insertion Initiation The recirculation pump trip system causes a pump trip on a signal of high reactor dome pressure or low-low reactor water level when the mode select switch is in the RUN mode. The alternate rod insertion system provides for initiating control rod insertion whenever the mode switch is in the RUN, STARTUP or SHUTDOWN mode. The limiting conditions for operation for the instrumentation are listed in Table 3.2-G.	G. Recirculation Pump Trip/Alternate Rod Insertion Surveillance for instrumentation which initiates Recirculation Pump Trip and Alternate Rod Insertion shall be as specified in Table 4.2-G.

TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP
AND
ALTERNATE ROD INSERTION

Minimum Number of Operable or Tripped Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Mode Select Requirements(2)
2	High Reactor Pressure	1160 psig	A/B
2	Low-Low Reactor Water Level	$\geq$ 78.5" above the top of the active fuel	A/B

NOTES

1. Minimum number of operable trip systems shall be two.

2. Required mode select when minimum number of operable or tripped channels are not satisfied:

   A. Reactor in STARTUP, REFUEL or SHUTDOWN mode for recirculation pump trip system.

   B. Reactor in REFUEL mode for alternate rod insertion system.

Table 4.2-G

Minimum Test and Calibration Frequency for
ATWS RPT/ARI Instrumentation

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
1. Reactor High Pressure	(1)	Once/Operating Cycle-Transmitter	Once/day
		Once/3 months - Trip unit	Once/day
2. Reactor Low-Low Water Level	(1)	Once/Operating Cycle-Transmitter	Once/day
		Once/3 months - Trip unit	Once/day

NOTES FOR TABLES 4.2.A THROUGH 4.2.G

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months.

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations of IRMs and SRMs shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is exempted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

   These instrument channels will be calibrated using simulated electrical signals once every three months.

4. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

5. Reactor low water level, high drywell pressure and high radiation main steam line tunnel are not included on Table 4.2.A since they are tested on Table 4.1.2.

6. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

3.2 BASES (Cont'd)

For each parameter monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between the two (2) channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

The recirculation pump trip/alternate rod insertion systems are consistent with the "Monticello RPT/ARI" design described in NEDO 25016 (Reference 1) as referenced by the NRC as an acceptable design (Reference 2) for RPT. Reference 1 provides both system descriptions and performance analyses. The pump trip is provided to minimize reactor pressure in the highly unlikely event of a plant transient coincident with the failure of all control rods to scram. The rapid flow reduction increases core voiding providing a negative reactivity feedback. High pressure sensors and low water level sensors initiate the trip. 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 in this unlikely postulated event. Requiring the trip to be operable only when in the RUN mode is therefore conservative. The low water level trip function includes a time delay of nine (9) seconds ± one (1) second to avoid increasing the consequences of a postulated LOCA. This delay has an insignificant effect on ATWS consequences.

Alternate rod insertion utilizes the same initiation logic and functions as RPT and provides a diverse means of initiating a reactor scram. ARI uses sensor diverse from the reactor protection system to depressurize the scram pilot air header, which in turn causes all control rods to be inserted.

References

1. NEDO-25016, "Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant," September 1976.

2. NUREG 0460, Volume 3, December 1978.

4.2 BASES (Cont'd)

instruments of similar design, a testing interval of once every three months has been found adequate.

The automatic pressure relief instrumentation can be considered to be a 1 out of 2 logic system and the discussion above applies also.

The instrumentation which is required for the recirculation pump trip and alternate rod insertion systems incorporate analog transmitters and are a new, improved line of BWR instrumentation. The calibration frequency is once per operating cycle which is consistent with both the equipment capabilities and the requirements for similar equipment used by other reactor vendors. The calibration frequency of the trip units is proposed to be quarterly, the same as other similar protective instrumentation. Likewise, the test frequency is specified at monthly like that of other protective instrumentation. A sensor check is proposed once per day; this is considered to be an appropriate frequency, commensurate with the design applications and the fact that the recirculation pump trip and alternate rod insertion systems are backups to existing protective instrumentation.

ATTACHMENT G

Technical Specification Change to Containment Isolation Valve Logic

Proposed Change

Reference is made to Pilgrim Nuclear Power Station Technical Specification Appendix A, Table 3.7.1 "Primary Containment Isolation Valves". The proposed change consists of providing high drywell pressure isolation signal to existing isolation logics for reactor water sample valves.

Reason For Change

NUREG 0578 Section 2.1.4 (1) requires a diversity in parameters sensed for the initiation of containment isolation as described in Standard Review Plan 6.2.4. In particular, isolation of all non-essential systems must be accomplished based on diverse signals indicative of a LOCA obtained from qualified Class IE systems.

The reactor water sample valves presently receive only one isolation signal (low low reactor water level) which meets the above criteria. A second isolation signal containing high drywell pressure will be added to the existing logics to provide the diverse signals required.

Safety Considerations

The new isolation signals have been obtained from new auxiliary relays wired in parallel to existing isolation trip relays in Panels C941 and C942. The addition of these Class IE relays to the existing circuits, the only interface with other safety components, does not degrade the logic circuit, as the wiring changes are accomplished in separate divisional Class IE Panels.

These design modifications do not diminish the ability of the affected components from performing their required safety function assuming a single active failure. The installation is designed to withstand the affects of the most severe natural phenomena postulated, and does not increase the probability of occurrence or consequences of an accident or malfunction of equipment important to safety.

This change has been reviewed by the Nuclear Safety Review and Audit Committee and has been reviewed and approved by the Operations Review Committee.

Schedule of Change

Approval of this change is requested prior to startup from our current refueling outage.

Boston Edison Company proposes that since this change is part of our implementation to the Commission's January 2, 1980, Confirmatory Order (NUREG 0578), it is exempt from any fees associated with 10 CFR 170.

TABLE 3.7.1

PRIMARY CONTAINMENT ISOLATION VALVES

Group	Valve Identification	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (sec.)	Normal Position	Action on Initiating Signal
1	Main Steam Line isolation valves	4	4	$3 \leq T \leq 5$	O	GC
1	Main steam line drain isolation valves	1	1	30	C	SC
1**	Reactor Water sample line isolation valves	1	1	10	C	SC
2	Drywell purge supply isolation valves		2	15	C(1) O(1)	SC GC
2	Suppression chamber purge supply isolation valves		2	15	O	GC
2	Nitrogen purge isolation valve		1	10	O	GC
2	Nitrogen makeup isolation valve		1	10	C	SC
2	Suppression chamber nitrogen makeup isolation valve		1	10	C	SC
2	Drywell purge exhaust isolation valves		2	15	C	SC
2	Drywell exhaust isolation valves		2	10	C	SC
2	Suppression chamber purge exhaust isolation valves		2	15	C	SC
2	Suppression chamber exhaust isolation valves		2	10	C	SC

NOTES FOR TABLE 3.7.1 (Cont'd)

2. High reactor vessel pressure

3. High drywell pressure

GROUP 4: Isolation valves in the high pressure coolant injection system (HPCI) are closed upon any one of the following signals:

1. HPCI steam line high flow

2. High temperature in the vicinity of the HPCI steam line

3. Low reactor pressure

GROUP 5: Isolation valves in the RCIC system are closed upon any one of the following signals:

1. RCIC steam line high flow

2. High temperature in the vicinity of the RCIC steam line

3. Low reactor pressure

GROUP 6: Actions in Group 6 are initiated by any one of the following:

1. Reactor low water level

2. Cleanup area high temperature

3. Cleanup inlet high flow

*The RHRS shutdown cooling injection isolation valves require a Group 2 signal plus high reactor vessel pressure.

**The Reactor Water Sample Line Isolation Valves initiate on a Group 1 signal plus high drywell pressure.

ATTACHMENT H

Proposed Technical Specifications Related To

Fire Protection Alternate Shutdown Stations

Proposed Technical Specifications

Reference is made to PNPS Technical Specifications, Appendix A, pages 103, 104, 106, 107, 108, 109 and 194. The proposed technical specifications provide surveillance requirements for equipment not already covered by existing technical specifications.

Reason for Change

Amendment #35 (Fire Protection) to our Operating License (DPR-35) required that an alternate shutdown system, independent of cabling and equipment in the cable spreading room be provided. As a result, modifications have been necessary which in turn require the inclusion of additional surveillance requirements into the technical specifications.

Safety Considerations

This proposal has been reviewed and approved by the Operations Review Committee and reviewed by the Nuclear Safety Review and Audit Committee.

Schedule of Change

This change will be put into effect upon receipt of approval by the Commission.

As stated in Amendment #35, no license amendment fee is required for technical specifications made necessary as a result of Commission requested changes related to Fire Protection.

Attachments

3.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the operational status of the core and suppression pool cooling subsystems.

Objective

To assure the operability of the core and suppression pool cooling subsystems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification

A. Core Spray and LPCI Subsystems

1. Both core spray subsystems shall be operable whenever irradiated fuel is in the vessel and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.2 below.

4.5 CORE AND CONTAINMENT COOLING SYSTEMS

Applicability

Applies to the Surveillance Requirements of the core and suppression pool cooling subsystems which are required when the corresponding Limiting Condition for operation is in effect.

Objective

To verify the operability of the core and suppression pool cooling subsystems under all conditions for which this cooling capability is an essential response to station abnormalities.

Specification

A. Core Spray and LPCI Subsystem

1. Core Spray Subsystem Testing.

	Item	Frequency
a.	Simulated Automatic Actuation test.	Once/Operating Cycle
b.	Pump Operability	Once/month and Once/cycle from the Alternate Shutdown Panel
c.	Motor Operated Valve Operability	Once/month and Once/cycle from the Alternate Shutdown Panel
d.	Pump flow rate Each pump shall deliver at least 3600 gpm against a system head corresponding to a reactor vessel pressure of 104 psig.	
e.	Core Spray Header Δ p Instrumentation	

3.5.A Core Spray and LPCI Subsystems (cont'd)

2. From and after the date that one of the core spray subsystems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding seven days, provided that during such seven days all active components of the other core spray subsystem and active components of the LPCI subsystem and the diesel generators are operable.

3. The LPCI Subsystems shall be operable whenever irradiated fuel is in the reactor vessel, and prior to reactor startup from a Cold Condition, except as specified in 3.5.A.4, 3.5.A.5 and 3.5.F.5.

4.5.A Core Spray and LPCI Subsystems (cont'd)

Check	Once/day
Calibrate	Once/3 months
Test	Once/3 months

2. When it is determined that one core spray subsystem is inoperable, the operable core spray subsystem, the LPCI subsystem and the diesel generators shall be demonstrated to be operable immediately. The operable core spray subsystem shall be demonstrated to be operable daily thereafter.

3. LPCI Subsystem Testing shall be as follows:

a.	Simulated Automatic Actuation Test	Once/Operating Cycle
b.	Pump Operability	Once/month and Once/cycle from the Alternate Shutdown Panel
c.	Motor Operated valve operability	Once/Month and Once/cycle from the Alternate Shutdown Panel
d.	Pump Flow	Once/3 months

Three LPCI pumps shall deliver 14,400 gpm against a system head corresponding to a vessel pressure of 20 psig

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.5.B Containment Cooling Subsystem

1. Except as specified in 3.5.B.2, 3.5.B.3, and 3.5.F.3 below, both containment cooling subsystem loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F, and prior to reactor startup from a Cold Condition.

4.5.B Containment Cooling Subsystem

1. Containment Cooling Subsystem Testing shall be as follows:

	Item	Frequency
a.	Pump & Valve Operability	Once/3 months and Once/cycle from the Alternate Shutdown Station
b.	Pump Capacity Test Each RBCCW pump shall deliver 1700 gpm at 70 ft. TDH. Each SSWS pump shall deliver 2700 gpm at 55 ft. TDH.	After pump maintenance and every 3 months
c.	Air test on drywell and torus headers and nozzles	Once/5 years

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3.5.B Containment Cooling Subsystem (Cont'd)	4.5.B Containment Cooling Subsystem (Cont'd)
2. From and after the date that one containment cooling subsystem loop is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such subsystem loop is sooner made operable, provided that the other containment cooling subsystem loop, including its associated diesel generator, is operable.	2. When one containment cooling subsystem loop becomes inoperable, the operable subsystem loop and its associated diesel generator shall be demonstrated to be operable immediately and the operable containment cooling subsystem loop daily thereafter.
3. If the requirements of 3.5.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.	
C. HPCI Subsystem	C. HPCI Subsystem
1. The HPCI Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, reactor pressure is greater than 104 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.C.2 and 3.5.C.3 below.	1. HPCI Subsystem testing shall be performed as follows:

a.	Simulated Automatic Actuation Test	Once/operating cycle
b.	Pump Operability	Once/month and Once/cycle from the Alternate Shutdown Station
c.	Motor Operated Valve Operability	Once/month and Once/cycle from the Alternate Shutdown Station
d.	Flow Rate at 1000 psig	Once/3 months
e.	Flow Rate at 150 psig	Once/operating cycle

3.5.C HPCI Subsystem (Cont'd)

2. From and after the date that the HPCI Subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, providing that during such seven days all active components of the ADS subsystem, the RCIC system, the LPCI subsystem and both core spray subsystems are operable.

3. If the requirements of 3.5.C cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to or below 104 psig within 24 hours.

4.5.C HPCI Subsystem (Cont'd)

The HPCI pump shall deliver at least 4250 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig.

2. 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.D Reactor Core Isolation Cooling (RCIC) Subsystem

1. The RCIC Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel, the reactor pressure is greater than 104 psig, and prior to reactor startup from a Cold Condition, except as specified in 3.5.D.2 below.

4.5.D Reactor Core Isolation Cooling (RCIC) Subsystem

1. RCIC Subsystem testing shall be performed as follows:

a.	Simulated Automatic Actuation Test	Once/operating cycle
b.	Pump Operability	Once/month and Once/cycle from the Alternate Shutdown Station
c.	Motor Operated Valve Operability	Once/month and Once/cycle from the Alternate Shutdown Station

3.5.D Reactor Core Isolation Cooling (RCIC) Subsystem (Cont'd)

2. From and after the date that the RCICS is made or found to be inoperable for any reason, continued reactor power operation is permissible only during the succeeding seven days provided that during such seven days the HPCIS is operable.

3. If the requirements of 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to or below 104 psig within 24 hours.

4.5.D Reactor Core Isolation Cooling (RCIC) Subsystem (Cont'd)

d. Flow Rate at 1000 psig — Once/3 months

e. Flow Rate at 150 psig — Once/operating cycle

The RCIC pump shall deliver at least 400 gpm for a system head corresponding to a reactor pressure of 1000 to 150 psig

2. When it is determined that the RCIC subsystem is inoperable, the HPCIS shall be demonstrated to be operable immediately and weekly thereafter.

3.5.E Automatic Depressurization System (ADS)

1. The Automatic Depressurization Subsystem shall be operable whenever there is irradiated fuel in the reactor vessel and the reactor pressure is greater than 104 psig and prior to a startup from a Cold Condition, except as specified in 3.5.E.2 below.

4.5.E Automatic Depressurization System (ADS)

1. During each operating cycle the following tests shall be performed on the ADS:

a. A simulated automatic actuation test shall be performed prior to startup after each refueling outage.

This test shall also be performed from the Alternate Shutdown Station within the same time frame.

b. With the reactor at pressure, each relief valve shall be manually opened until a corresponding change in reactor pressure or main turbine bypass valve positions indicate that steam is flowing from the valve

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the auxiliary electrical power system.

Objective:

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification:

A. Auxiliary Electrical Equipment

The reactor shall not be made critical unless all of the following conditions are satisfied:

1. At least one offsite transmission line and the startup transformer are available and capable of automatically supplying auxiliary power to the emergency buses.

2. An additional source of offsite power consisting of one of the following:

   a. A transmission line and shutdown transformer capable of supplying power to the emergency 4160 volt buses.

   b. The main transformer and unit auxiliary transformer available and capable of supplying power to the emergency 4160 volt buses.

3. Both diesel generators shall be operable. Each diesel generator shall have a minimum of 19,800 gallons of diesel fuel on site.

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Auxiliary Electrical Equipment Surveillance

1. Diesel Generators

   a. Each diesel generator shall be manually started and loaded once each month to demonstrate operational readiness. The test shall continue for at least a one hour period at rated load.

   During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps shall be demonstrated, and the diesel starting time to reach rated voltage and frequency shall be logged.

   Also, once per operating cycle the diesel generator shall be manually started and loaded from the Alternate Shutdown Statio

   b. Once per operating cycle the condition under which the diesel generator is required will be simulated and a test conducted to demonstrate that it will start and accept the emergency load within the specified time sequence. The results shall be logged.