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January 5, 1981

BECo. Ltr. #81-01

Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

License No. DPR-35
Docket No. 50-293

Responses to NUREG-0737 January 1, 1981 Requirements

Dear Sir:

You have requested licensees of operating plants to submit documentation in accordance with schedule and criteria established in NUREG-0737. "Post TMI Requirements". Attached you will find our response to the following 0737 requirements.

I.A.1.1	Shift Technical Advisor
I.C.I	Guidance for the Evaluation and Development of Procedures for Transients and Accidents
I.C.5	Procedures for feedback of operating experience
I.C.6	Verifying Correct Performance of operating activities
II.B.2	Design Review of Plant Shielding and Environmental Equipment Qualification
II.B.4	Training for Mitigating Core Damage
II.E.4.2	Containment Isolation Dependability
II.F.2	Instrumentation for Detection of Inadequate Core Cooling
II.K.3.3	Reporting Safety and Relief Valve Failures and Challenges
II.K.3.13	Separation of HPCI and RCIC Initiation Setpoints
II.K.3.17	ECCS Outage Report
II.K.3.21	Restart of Core Spray & LPCI Injection Systems
II.K.3.22	Auto Switchover of RCIC

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II.K.3.44	Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure
II.K.3.45	Depressurization other than full ADS
III.D.3.4	Control Room Habitability

Several of our responses were developed in conjunction with the BWR Owners' Group and General Electric. These positions were not available for review until December 10, 1980, at which time an in-house review was commenced. As a result of our review, further development is required to satisfy plant specific requirements at Pilgrim Nuclear Power Station in response to II.K.3.13. This work has begun and a detailed response will be provided to your staff by January 31, 1981.

We trust this letter is responsive to your requirements; however, should you desire additional information or clarification, please feel free to contact us.

Very truly yours,

Attachment

Attachment

1.A.1.1 Shift Technical Advisor

In response to previous NRC requirements contained in NRC letter dated September 13, 1979 and October 30, 1979, Boston Edison Company has created the position of Shift Technical Advisor (STA) at PNPS Unit #1. A description of the STA program was provided in BECo letter #80-54, dated April 4, 1980. A copy of our response is attached for your convenience. As we indicated in our December 15, 1980 submittal as a result of a reorganization within the nuclear organization, the STA group no longer reports to the Assistant Station Manager. Presently, STA's report to the Staff Assistant-Nuclear Safety, who in term reports to the Nuclear Operations Manager.

In response to your requirement for a description of our STA training, the following is provided:

The STA Training Program has been developed using as a guide the recommendations contained in the INPO document entitled "Nuclear Power Plant Shift Technical Advisor - Recommendations for Position Description, Qualifications, Education and Training", Revision 0, dated April 30, 1980. The course content for the initial group of STA candidates is as follows:

A. Seven weeks (approximately 260 contact hours) of training on-site, administered by the Pilgrim Nuclear Power Station Training Group. This training includes:

   Plant Specific System Training covering all the BWR Specific Systems of Section 6.4 of the INPO Document.

   Plant Specific Reactor Technology

   Plant Chemistry

   Nuclear Instrumentation and Controls

   Reactor Plant Thermal Cycle

   Process Instrumentation and Control

   Review of Reactor Theory including Reactivity Control, Reactivity Coefficients, and Fission Product Poisons

   Review of Radiation Protection and Health Physics

B. Three weeks (120 contact hours) of simulator training, administered by a vendor on a BWR simulator. This training includes 40 hours of time on the simulator and 80 hours of classroom time. Primary emphasis is on reactor operations, both normal and abnormal, and response of the plant to transients and accidents.

C. Eight weeks (approximately 304 contact hours) of training, tailored specifically for Pilgrim STA's and administered by General Electric Company, on the following subjects:

	Contact Hours
Station Nuclear Engineering	120 hours
Abnormal Event Analysis	80 hours
Nuclear and Non-Nuclear Instrumentation	40 hours
Control Room Management	40 hours
Communications/Motivation	24 hours

D. One week (40 contact hours) on Mitigation of Core Damage and Emergency Procedures, administered by various BECo specialists.

By January 1, 1981, all initial STA candidates will have completed the training outlined under A and B above. It is the position of Boston Edison Company that the training provided as of January 1, 1981 and the qualifications of the STA candidates, each of whom holds a Bachelor's Degree or equivalent, are sufficient to demonstrate conformance with the criteria of your October 30, 1979 letter, particularly in view of the fact that as of January 1, 1981, all STA candidates will have had at least 520 hours on-duty experience serving the STA function. The balance of the STA training is presently scheduled to be completed by June 1, 1981. Plans for requalification training will be in place by January 1, 1982.

Scheduling of STA training has been impeded by an industry-wide crunch on hiring qualified STA candidates, conflicting regulatory requirements to both train STA's and have an STA on duty on each watch, and lack of availability of simulator time and qualified training instructors.

In response to your requirement for a description of our long term STA program, the following is provided:

The aim of the long term STA program is to provide an on-shift technical advisor to the Nuclear Watch Engineer, who has college level education in engineering and science subjects and specific training in the response and analysis of the plant for transients and accidents, until such time as these characteristics are attained by the Nuclear Watch Engineer. The STA is accountable for the following end results:

A. Contributes to maximizing safety of operations by independently observing plant status and advising shift supervision of conditions that could compromise plant safety.

B. Contributes to maximizing plant safety during transient or accident situations by independently assessing plant conditions and by providing the technical assistance necessary to mitigate the incident and minimize the effect on personnel, the environment, and plant equipment.

These accountabilities are consistant with those in the referenced INPO document. Boston Edison's long term STA program closely follows that described in the INPO document. Boston Edison agrees with the Commission's assessment that the descriptions as set forth in Sections 5 and 6 of Revision 0 to the INPO document are an acceptable approach for the selection and training of personnel to staff the STA position. However, it is our intent to use this document as a guideline only and to deviate from it at our discretion, so long as, in our judgement, such deviation does not substantially affect the quality or intent of the STA program.

For example, we would contemplate deviation, on a candidate by candidate basis, in the following subsections of the INPO document:

5.2 Experience - The document specifies at least 12 months experience shall be at the station at which the position is to be filled. All of our present STA candidates meet this requirement. However, considering the difficulty we have experienced in filling job vacancies and the potential high turnover rate for the STA position, we intend to maintain the option to waive this requirement.

5.3 Absences from STA Duties - The document requires certain training prior to assuming on-shift responsibilities for any STA who has not been actively performing that function for a period of 30 days or longer. Boston Edison agrees that the person absent from STA duties for periods greater than 30 days shall be briefed on significant procedure and facility changes during that absence. Absences from STA shifts as part of the STA training program or as part of the off-shift STA function, including the operating experience assessment function, will not be applicable to this definition of absence as these are considered to be integral to the STA function.

6.0 Education and Training Requirements - Section 6.1 of the document specifies certain prerequisite education and college level fundamental education considered necessary for successful completion of the advanced course work specified in Sections 6.2 through 6.8. It has been our experience that capable, highly motivated individuals can, in fact, pass the advanced course work without formal training in certain prerequisite or fundamental subjects, by initiating self-study programs or tutoring, as necessary. The selection criterion for assuming the STA function is based on successful completion of oral and written exams given along with the advanced course work. It is Boston Edison's position that candidates who pass these comprehensive exams, including certification, if required, have demonstrated an acceptable level of knowledge of the prerequisite subjects and need not document formal training in these areas.

However, in recognition that certain of the fundamentals are more important than others to the STA function, the extent of the advanced course work training for Boston Edison STA's has been increased to 19 weeks as compared to 15 weeks specified in the INPO document, with heavy emphasis on nuclear engineering, abnormal event analysis and mitigation of core damage. Also, the qualifications for an STA candidate require the individual to have a Bachelor's Degree in engineering or science plus 3 years power plant experience or equivalent. Or equivalent is defined as an Associate Degree in nuclear or mechanical engineering plus 5 years nuclear experience.

As regards licensing plans and plans for phase out of the STA program, these matters are still under review. The STA's serving in that position will be encouraged to attain SRO licenses, but this will be on a voluntary basis. Phase out of the STA program is dependent upon the success of the program to upgrade the educational background of current Watch Engineers. Because the program for the Watch Engineers is developmental at this time, the actual duration of the STA program is not determinable. It is an objective of the STA program to prepare STA's, who are amenable, for future positions as Watch Engineers. Standards of increased operational experience and NRC licensing will be part of the development of those STA's inclined to progress to that position. Those STA's who may obtain NRC licenses, but are not attracted to the Watch Engineer position may move to other positions in the nuclear organization when and if the STA program is phased out.

As stated in our December 15, 1980 submittal, we are reevaluating and will reissue our position related to Technical Specifications changes for this requirement.

2.2.1.b Shift Technical Advisor

The Boston Edison Company has created the position of Shift Technical Advisor (STA) at Pilgrim Station Unit #1. This position is accountable for the safe operation of Pilgrim Unit #1 through contributing to assessment of plant conditions during normal operation and transients consistent with technical specifications, procedures and regulatory requirements. The Shift Technical Advisor Group is independent of Operations and maintains direct line reporting to the Assistant Station Manager. The Shift Technical Advisor is assigned to a specific shift and there are three (3) shifts per day. One (1) STA is on duty at all times during normal plant operation. The STA maintains a high awareness of safety in plant operation and provides analysis and reporting of plant conditions during operation and transients. The STA provides technical expertise to the Watch Engineer in order to help the Watch Engineer recognize, diagnose and respond to unusual events. The STA provides the perspective and the time for assessing plant conditions by independently monitoring plant safety.

On a daily basis, the STA communicates with other STA's to report ongoing plant conditions and the status of any special circumstances. Also on a daily basis, the incumbent(s) interface with the Watch Engineer(s) to assess the effectiveness of operations. On a frequent basis (weekly) the incumbent(s) receive update reports of unusual occurrences from the Reliability and Safety Assessment Group. As required, the incumbent(s) provide technical comments and recommendations on plant operations and responses of operators to transients for the Chief Operating Engineer. The STA provides assessment of transients or conditions during an event to the On-Site Technical Support Center Staff. The role of the incumbent is essentially one of monitoring plant conditions and maintaining a close interface with the Watch Engineer regarding plant operations. The scope of the position also includes providing recommendations to the Watch Engineer as requested for actions to be taken by the Nuclear Plant Operators in response to an unusual event or transient.

Boston Edison believes that our program meets the intent of NUREG-0578 Position 2.2.1.b, Shift Technical Advisor.

1.C.1 Guidance for the Evaluation and Development of Procedures for Transients and Accidents

In a letter dated June 30, 1980 (letter #MFN-117-80, R. H. Buchholz to D. G. Eisenhut), G.E. submitted BWR Emergency Procedure Guidelines on behalf of the BWR Owners Group. Boston Edison is a participant in the BWR Owners Group and endorses this submittal. Following review of the guidelines and issuance of further guidance by the NRC, Boston Edison intends to implement these guidelines in emergency procedures at PNPS 1 on a schedule consistant with the requirements of item 1.C.1.

1.C.5 Procedures for Feedback of Operating Experience to Plant Staff

Procedures governing feedback of operating experience have been completed, and will be in effect by January 31, 1981.

I.C.6 Guidance on Procedures for Verifying Correct Performance of Operating Activities

The requirements of TAP Item I.C.6 are being reviewed against existing station policies and procedures. Where differences are identified, a decision will be made whether or not to incorporate the change. It is expected that this effort will be completed with the necessary procedure changes by June 1981.

11.B.2 Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems which may be used in Post Accident Operations

In a letter dated April 4, 1980 (BECo letter #80-54, G. C. Andognini to H. R. Denton) Boston Edison described the shielding design review performed at PNPS 1 and indicated potential system modifications and shielding additions under evaluation to increase accessibility to plant vital areas following the postulated accident. This review was performed to the requirements of NUREG-0578, Item 2.1.6b.

On the basis of clarification provided in NUREG-0737, Item 11.B.2, however, Boston Edison is reviewing this initial study and reevaluating previous results. A preliminary reanalysis of the Control Room, Post Accident Sampling location and Sample Analysis area indicates each would provide the required level of accessibility following the postulated accident. It is anticipated that reanalyses of other plant vital areas will result in similar findings. Modifications in progress or presently scheduled for completion by January 1, 1982 which are related to insuring vital area accessibility following an accident are as follows:

1. Remote Closure Capability for Reactor Building Truck Lock Door as described in Boston Edison's April 4, 1980 letter (Item 2.1.6b).
2. Post-Accident Sample Sink Installation as required by NUREG-0737, Item 11.B.3 and described in Boston Edison's April 4, 1980 letter (Item 2.1.8a).
3. Remote operation capability for Post Accident Combustible Gas Control valves as described in Boston Edison's April 4, 1980 letter (Item 2.1.5a).

Regarding deviations from position 11.B.2, Boston Edison does not intend to consider a "LOCA event in which the primary system may not depressurize" in determining dose rates. Emergency procedures in effect at PNPS 1 require that the RCS be promptly depressurized and cooled down with low pressure systems following an accident of large scale fuel damage.

II.B.4 Training for Mitigating Core Damage

A program has been developed at PNPS 1 to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severly damaged. This program is intended for training STA's and operating personnel from the plant manager through the operating chain to the licensed operators and includes all training indicated in enclosure 3 to H. R. Denton's March 28, 1980 letter. The initiation of this training program is tentatively scheduled for February 1981, with the initial program scheduled for completion by April 1, 1981.

II.E.4.2 Isolation Dependability

In response to previous NRC requirements contained in NRC letters, dated September 13 and October 30, 1979, Boston Edison has reviewed the containment isolation system in accordance with criteria established by the NRC requirements. Boston Edison believes its position contained in BECo letter #80-54 to be responsive to NRC positions 1-4 of Task Action Plan Item II.E.4.2. This position is attached for your convenience.

In response to position 5 of II.E.4.2 to reduce the containment setpoint pressure that initiates containment isolation for nonessential penetrations to the minimum compatible with normal operating conditions, the following is provided:

The BWR Owners Group and General Electric have prepared a response to this NRC requirement which demonstrate the adequacy of the present containment isolation setpoint of approximately 2 psi. Boston Edison endorses this owner's group position. In addition, BECo letter #80-57 requested and provided justification to raise the High Drywell Pressure trip level setting from 2 psig to 2.5 psig. NRC letter dated May 12, 1980 approved this change to PNPS's Technical Specifications. Boston Edison believes its present High Drywell Pressure 2.5 psig is adequate and no further change is required.

In response to position 6 of II.E.4.2, which requires containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CS136-4 or the Staff Interim position of October 23, 1979 to be sealed closed as defined in SRP6.2.4, the following is provided:

Boston Edison in response to the staff interim position implemented controls to satisfy this requirement, which were reviewed and approved by your staff in a letter dated September 9, 1980.

Subsequently, we have not been in compliance to limit the operation of the containment vent and purge valves to less than 90 hours a year. We are presently developing further procedural controls to prevent reoccurence of this condition which will be complete by January 15, 1981. We believe, this will satisfy the staff's interim position for containment purge and vent valve operation

2.1.4 Containment Isolation Provisions for PWR's and BWR's

Position

1. All containment isolation system designs shall comply with the recommendations of SRP 6.2.4; i.e, that there be diversity in the parameters sensed for the initiation of containment isolation.

2. All plants shall give careful reconsideration to the definition of essential and non-essential systems, shall identify each system determined to be essential, shall identify each system determined to be non-essential, shall describe the basis for selection of each essential system, shall modify their containment isolation designs accordingly, and shall report the results of the re-evaluation to the NRC.

3. All non-essential systems shall be automatically isolated by the containment isolation signal.

4. The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

Response

1. Diversity of parameters sensed for the initiation of containment isolation shall be provided in accordance with SRP-6.2.4.

Diversity in Parameters Sensed for Initiation of Containment Isolation

A. Secondary Containment Isolation (FSAR 5.3.3.3)

Either of two signals will initiate the secondary containment system. These signals, which indicate a loss-of-coolant accident inside the drywell are high drywell pressure or low reactor water level. In addition, radiation monitors in the operating(refueling) floor ventilation exhaust duct, which indicate a fuel handling accident, can initiate the secondary containment system. Secondary containment can also be initiated manually from the control room.

B. Table 1 summarizes the isolation signal codes (asterisk items only) used by the Primary Containment and Reactor Vessel Isolation System. Additional details may be found in PNPS 1 FSAR Section 7.3.4.7 (Isolation Functions and Settings)

Exceptions to the diverse isolation signals criteria have been identified to the NRC in response to IE Bulletin 79-08. The NRC has accepted the existing methods for isolation of all valves except the reactor water sample valves, the MSIV drains, and the RWCU supply and return valves.

- The reactor water sample valves presently receive only one isolation signal (low-low reactor water level) that meets the diverse isolation criteria. A second isolation signal containing high drywell pressure will be added to the existing logics to provide the diverse signals required as these valves have no effect on plant safety.

Changing the isolation signals to the MSIV drains, however, could affect plant safety. Operation of these valves to more restrictive isolating requirements than the MSIV's could possibly result in condensate buildup between the MSIV's thus preventing operation (opening) of the MSIV's or damaging the steamlines to the condenser. Either failure will needlessly eliminate the condenser as a heat sink after a unit scram thus removing one possible method of cool down.

RWCU suction and return line isolation valves are currently provided with only one containment isolation signal in addition to the process isolation signals. The RWCU system intentionally remains active to keep cleansing the vessel water during the situation where high drywell pressure exists because the drywell coolers are not operating or a small break LOCA occurs. The small break LOCA could also result in a high drywell pressure condition without reaching a low reactor vessel level condition. It is desirable to keep the RWCU operating under these conditions.

Response

2. Definition of Essential and Non-Essential Systems

A. Source of Definition is NUREG 0578, Pg. A-14

B. Definitions:

1. Essential Systems: Those systems that should be selectively isolated during containment isolations only after it is established that the use of these systems will not be needed for an accident or abnormal transients.

2. Non-Essential Systems: Those systems not needed for mitigation of an accident or abnormal transient and which should be immediately isolated during containment isolation.

C. FSAR (Sect. 1.5.2.6.2) Criteria for Implementing Definitions:

1. A primary containment shall be provided to completely enclose the reactor vessel. It shall be designed to act as a radioactive material barrier during or following accidents that release radioactive material into the primary containment. It shall be possible to test the primary containment integrity and leak tightness at periodic intervals.

2. A secondary containment that completely encloses both primary containment and fuel storage areas shall be provided and shall be designed to act as a radioactive material barrier.

3. The primary and secondary containments, in conjunction with other engineered safeguards, shall act to prevent the release of radioactive material from the containment volumes from exceeding the guideline values of applicable regulations.

4. Provisions shall be made for the removal of energy from within the primary containment as necessary to maintain the integrity of the containment system following accidents that release energy to the primary containment.

5. Piping that both penetrates the primary containment structure and could serve as a path for the uncontrolled release of radioactive material to the environs shall be automatically isolated whenever such uncontrolled radioactive material release is threatened. Such isolation shall be effected in time to prevent radiological effects from exceeding the guideline values of applicable regulations.

Classification of Systems

Essential Systems

a. RHR (except head spray)

b. Standby Liquid Control

c. RCIC

d. Core Spray (except test lines)

e. HPCI

f. Main Steam Flow Instrumentation

g. Drywell Pressure Instrumentation

h. RBCCW - see note

i. Containment Atmospheric Control System

Non-Essential Systems

a. Main Steam

b. Feedwater

c. Reactor Water Sample

d. Control Rod Drive Hydraulic Return

e. Control Rod Drive Inlet and Outlet

f. RHR Reactor Head Spray

g. Reactor Water Cleanup

h. Core Spray Test Line to Suppression Pool

i. Drywell Equipment Drain

j. Drywell Floor Drain

k. Traversing In-core Probe

l. Service Air

m. Instrument Air

Note: RBCCW has 2 Class C containment isolation valves (check valves and motor operated gate valve), one valve per containment penetration. Class C valves are on process lines that penetrate the primary containment but do not communicate directly with the reactor vessel, with the primary containment free space, or with the environs. Class C lines require only one valve which closes automatically by process action (i.e, reverse flow) or by remote manual operation from the control room.

Response

3. All Non-Essential Systems Shall be Automatically Isolated by the Containment Isolation Signal

Table 1 gives a listing of all non-essential systems and their respective isolation signals. Items which require greater detail are described in the next few paragraphs:

Tip Valves

Section 5.2.3.5.2 of FSAR

Tip system guide tubes are provided with an isolation valve which closes automatically upon receipt of proper signal and after the TIP cable and fission chamber have been retracted. In series with this isolation valve, an additional or backup isolation shear valve is included. Both valves are located outside the drywell. The function of the shear valve is to assure integrity of the containment in the unlikely event that the other isolation valve should fail to close or the chamber drive cable should fail to retract if it should be extended in the guide tube during the time that containment isolation is required. This valve is designed to shear the cable and seal the guide tube upon an actuation signal. Valve position (full open or full closed) of the automatic closing valves will be indicated in the control room. Each shear valve will be operated independently. The valve is an explosive type valve and each actuating circuit is monitored. In the event of a containment isolation signal, the TIP system receives a command to retract the traveling probes. Upon full retraction, the isolation valves are then closed automatically. If a traveling probe were jammed in the tube run such that it could not be retracted, instruments would supply this information to the operator, who would in turn investigate to determine if the shear valve should be operated.

Section 7.5.9.2.2. of PNPS 1, FSAR

A valve system is provided with a valve on each guide tube entering the primary containment. These valves are closed except when the TIP subsystem is in operation. A ball valve and a cable shearing valve are mounted in the guide tubing just outside of the primary containment. They prevent the loss of reactor coolant in the event a guide tube ruptures inside the reactor vessel. A valve is also provided for a gas purge line to the indexing mechanisms. A guide tube ball valve opens only when the TIP is being inserted. The shear valve is used only if a leak occurs when the TIP is beyond the ball valve and power to the TIPS fails. The shear valve, which is controlled by a manually operated protected switch, can cut the cable and close off the guide tube. The shear valves are actuated by detonation squibs. The continuity of the squib circuits is monitored by front panel indicator lights in the control room.

A guide tube ball valve is normally de-energized and in the closed position. When the TIP starts forward the valve is energized and opens. As it opens it actuates a set of contracts which gives a signal light indication at the TIPS control panel and bypasses an inhibit limit switch which automatically stops TIP motion if the ball valve does not open on command.

Compressed Air System Valves

Section 10.11.3.1 of PNPS 1 FSAR

Pressure loss in the high pressure system, sensed by several pressure switches, will cause valves in the service air header, the low pressure service air cross-around line, and the non-essential instrument air header to close in a cascading sequence thus leaving the essential instrument air header as the only header drawing air from the receivers in the event that supply pressure decreases.

Response

4. Reopening of Containment Isolation Valves

During meetings with the NRC on December 11, 1979 and March 18, 1980, the NRC accepted the PNPS reset circuits for the Balance of Plant Isolation valves with the exception of the primary containment vent and purge system. To reset the BOP isolation logic after a scram, it is first necessary to reset the General Electric isolation logic at panel C905 and then reset the BOP logics at panel C7. The only change required by the NRC to this reset function is the replacement of the existing reset pushbuttons on panel C7 with keylocked selector switches, and this has been done.

The control circuits for the primary containment vent and purge system isolation vlaves have been revised by wiring valve control switch contacts (either directly or through auxiliary relays) parallel to the normally closed reset selector switch contacts. The control switch contacts, closed whenever a control switch is in an open position, provide a path for maintaining the trip relays energized for isolation (independent of reset switch position) until all control switches are moved to close. At that time,the isolation logics can be reset by operation of the keylocked reset selector switch.

Also affected by NUREG 0578 were the MSIV's, the reactor water sample valves, the drywell sump effluent valves, and the "emergency open" position of the vent and N2 makeup valves.

The MSIV's control circuits were revised by wiring "close" contacts from each MSIV switch in series with the applicable trip logic reset contacts. This arrangement requires the operator to move all MSIV control switches to "close" before the trip logics can be reset after an automatic isolation.

The control circuits for the "emergency open" position of the vent and N2 makeup valves have been revised by wiring contacts from the applicable control switches (via auxiliary relays) in series across the trip relay sealin contact. This arrangement, similar to that used on the MSIV's, requires that all valve control switches be closed before the trip relay is reset. In addition, we are replacing the existing control switches as recommended by the NRC, to allow operation between the "Open" and "Close" without a key and require a key to get into the "Emergency Open" position.

The control circuits for the remaining valves with reset problems have been modified by replacing maintained contact control swiches with three position, spring return to normal switches. These switches used in conjunction with auxiliary relays provide sealin circuits which,when tripped, will require operator action to open the valves after the isolation logics have been reset.

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TABLE I

ISOLATION SIGNALS TO NON-ESSENTIAL SYSTEMS

System Penetration	Normal Status(1)	Isolation Signal
1a Main Steam Lines	Open	B, C, D, P, Q
b Main Steam Drains	Closed(2)	B, C, D, P, Q
2 Reactor Feedwater	Open	Rev. Flow (Check Valves)
3 Reactor Water Sample	Closed(2)	B, C, D, P, Q, F, A
4 CRD Return	Note 4	Rev. Flow (Check Valves)
5 CRD In and Outlet	Note 4	Note 4
6 RHR Head Spray	Closed	AUF
7 Reactor Water Cleanup	Open	A, W, Y, J, RM
8 CS Test Line	Closed(2)	G
9 Drywell Equip. Drains	Open	B, F
10 Drywell Floor Drains	Open	B, F
11a TIP Primary	Closed(2)(3)	F, A
b Backup	Open	RM (Explosive Shear Valve)
12 Service Air	Closed	Rev. Flow (Check Valve)
13 Instrument Air	Open	Inside - Rev. Flow (Check Valve)
	Open	Outside - RM

NOTES:

(1) Normal status position of a valve is the position during normal power operation of the reactor.

(2) Valve can be opened or closed by remote manual switch for operating convenience during any mode of reactor operation except when automatic signal is present.

(3) Signal "A" or "F" causes automatic withdrawal of TIP probe, then valve automatically closes by mechanical action.

(4) CRD solenoid valves are normally closed, but they open on rod movement and during scram.

ISOLATION SIGNAL CODES FOR TABLE I

SIGNAL	DESCRIPTION
A*	Reactor vessel low water level - scram and close isolation valves except main steam lines.
B*	Reactor vessel low low water level - initiate RCIC, HPCI and close main steam line isolation valves.
C*	High radiation - main steam line (also causes scram).
D*	Line break - main steam line (steam line high space temperature or high steam flow).
E	Reactor low low level or high drywell pressure - select LPCI and close other loop valves.
F*	High drywell pressure - close RHR/shutdown cooling and head spray plus the RHR to radwaste valves.
G	Reactor vessel low water level and low pressure; or high drywell pressure - initiate Core Spray and RHR systems.
J*	Line break in cleanup system - high space temperature, or high flow.
K*	Line break in RCIC system steam line to turbine (high steam line space temperature or high steam flow) or low steam pressure.
L*	Line break in HPCI system steam line to turbine (high steam line space temperature or high steam flow) or low steam line pressure.
M*	Line break in RHR shutdown and head cooling (high space temperature; alarm only; no auto closure).
P*	Low main steam line pressure at inlet to main turbine (RUN mode only).
S	Low drywell pressure - close containment spray valves.
T	Low reactor pressure permissive to open core spray and RHR-LPCI valves.
U	High reactor vessel pressure - close RHR shutdown cooling valves and head cooling valves.
W	High temperature at outlet of cleanup system nonregenerative heat exchanger.
Y	Standby liquid control system actuated.
RM*	Remote manual switch from control room.

Q Reactor high water level - Isolate main steam line (except in run mode).

X RCIC or HPCI steam supply valve (as applicable) not fully closed.

*These are the isolation functions of the primary containment and reactor vessel isolation control system; other functions are given for information only.

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II.F.2 Instrumentation for Detection of Inadequate Core Cooling

On December 28, 1979, General Electric submitted on behalf of the BWR Owners Group (letter #MFN-314-79, R. H. Buchholz to D. F. Ross) a prepublication version of Section 3.5.2.3 of NEDO-24708 in response to NUREG-0578, Item 2.1.3b. This submittal indicated that "the reactor vessel water level measurement technique provided on the General Electric BWR performs satisfactorily for all modes of normal operation, anticipated transient conditions and credible accident conditions". In a later submittal (BECo letter #80-268, dated October 29, 1980) Boston Edison indicated its concurrence with the BWR Owners Group position that existing instrumentation is adequate for detecting inadequate core cooling.

As stated in the December 15, 1980 submittal to the NRC, Boston Edison is in the process of reevaluating its position on this item on the basis of new information and clarification contained in NUREG-0737, Item II.F.2. The NRC will be notified promptly of any changes to Boston Edison's present position that installed instrumentation is satisfactory.

II.K.3.3 Reporting Safety and Relief Valve Failures and Challenges

Boston Edison has reviewed planr operating records since April 1, 1980 to identify all relief valve challenges or failures. We have defined a relief valve challenge to be anytime a relief or safety valve received a signal to operate via High reactor pressure, auto signal (ADS) or control switch (manual). Based on this definition, all relief and safety valve challenges at PNPS resulted from manual actuation except for two occurances when a relief valve opened due to high nitrogen pressure to the relief valve solenoid. The following is a list of relief and safety valve challenges since April 1, 1980 (1980 refuel outage ended May 16, 1980):

Challenge Date	S/RV #	Reason for Challenge	Remarks
5/17/80	RV203-3A 3B 3C 3D	Performed tech spec surveillance	All valves tested OK
5/25/80	RV203-3D	Perform oper test per temp. Procedure TP 80-65	Valve failed to open
5/25/80	RV203-3D	Test after maintenance	Valve would not open
5/25/80	RV203-3D	Test after maintenance	Valve would not open
5/25/80	RV203-3C	Test after maintenance related to problems with RV203-3D	Tested OK
5/25/80	RV203-3B	Test after maintenance related to problems with RV203-3D	Tested OK
5/26/80	RV203-3A	Test after maintenance related to problems with RV203-3D	Tested Ok
5/26/80	RV203-3D	Test after maintenance	Would not open
5/26/80	RV203-3D	Test after maintenance	Tested OK

Safety/Relief Valve Challenge

Challenge Date	S/RV #	Reason for Challenge	Remarks
5/26/80	RV203-3D	Re-test (reliability)	Tested OK
8/1/80	RV203-3A	Operability Test	Tested OK
8/1/80	RV203-3B	Operability Test	Tested Ok
8/1/80	RV203-3C	Operability Test	Tested OK
8/1/80	RV203-3D	Operability Test	Valve did not open
8/3/80	RV203-3D	Operability Test after maintenance	Tested OK
8/5/80	RV203-3D	Operability test 8 times	Test OK 8 times
8/30/80	RV203-3D	Accelerated test program	Tested OK
10/1/80	RV203-3D	Rx scram/operator open & closed valve via control switch	Valve opened-would not close
10/5/80	RV203-3D	Test after maintenance	Tested OK
10/7/80	RV203-3A	Valve opened due to hi instrument nitrogen pressure to the RV solenoid	
10/8/80	RV203-3A	Test after maintenance	Tested OK
10/31/80	RV203-3A	Valve opened due hi instrument nitrogen pressure to the RV solenoid	

II.K.3.13 Separation of HPCI and RCIC System Initiation Levels

In conjunction with the BWR Owners Group and General Electric, TAP Item II.K.3.13 was addressed as two separate issues which are: a) Separation of HPCI and RCIC initiation levels and b) Auto restart of the RCIC system on low water level.

In response to the separation of HPCI and RCIC initiation level, General Electric on behalf of the BWR Owners Group submitted a position defending the present initiation levels of the HPCI and RCIC system. This was transmitted to the NRC in G.E. letter, dated October 1, 1980. Boston Edison endorsed this position in BECo letter #80-246, dated October 1, 1980. Boston Edison believes the response provided by the Owners Group and endorsed by us adequately addresses the NRC concern and no further action is required.

General Electric on behalf of the BWR Owners Group evaluated the second NRC requirement to modify RCIC logic to have it restart on subsequent low water level signals. G.E. evaluation showed that this change would contribute to improved system reliability and that it could be accomplished without adverse effect on system function and plant safety. Boston Edison has reviewed G.E's proposed modification and generally concurs with G.E.'s solution for the addition of auto reset to the RCIC system, however, a detailed proposal based on plant specific requirements of PNPS-1 must be solicited from G.E. and examined by BECo before a commitment is made to incorporated any modification to the RCIC system. We anticipate to have a description of the proposed modification no later than January 31, 1981.

II.K.3.17 Report on Outages of ECC Systems and Proposed Technical Specification Changes

BECo contracted General Physics Corporation to prepare the required ECCS outage report. Work commenced in October and involved a significant amount of record searching, technical specification review, and review of post operation logs. The draft report submitted to BECo was confusing and hard to interpret, therefore, after further revision, the ECCS outage report will be submitted by January 15, 1981.

II.K.3.21 Restart of Core Spray and LPCI Systems

The BWR Owners Group and General Electric have reviewed this NRC requirement and do not believe the NRC suggestions will necessarily enhance the safety of the plant. This conclusion is based on the adequacy of the current ECCS logic design coupled with the potentially negative impact on overall safety of the proposed changes.

II.K.3.22 Automatic Switchover of RCIC System Suction--Verify Procedures and Modify Design

Boston Edison has reviewed station operating procedures to verify that procedures exist clearly describing manual switchover of the RCIC suction from the Condensate Storage Tank to the Suppression Pool. The necessary procedures will be revised by January 15, 1981.

II.K.3.44 Evaluation of Anticipated Transients with Single Failure to Verify no Fuel Failure

The BWR Owners Group and General Electric have reviewed this NRC requirement. It was shown that for BWR/2-6 plants 1 adequate core cooling is maintained for the worse case conditions evaluated. Boston Edison has reviewed the Owners Group position and believes it to be applicable to PNPS Unit 1. No further action is required.

11.K.3.45 Evaluation of Depressurization with other than ADS

The BWR Owners Group and General Electric have prepared a report in response to TAP Item 11.K.3.45. The report shows that depressurization rates other than full ADS:

1) Do not exceed vessel integrity limits for a full ADS blowdown.

2) For slower depressurization rates, there is little impact on vessel fatigue relative to full ADS blowdown.

3) Slower depressurization rates have an adverse impact on core cooling capability.

Boston Edison has reviewed the owners group report and endorses the position.

III.D.3.4 Control Room Habitability Requirements

Boston Edison has completed a control room habitability study in accordance with criteria established in NRC TAP Item III.D.3.4. The results of the study demonstrate that PNPS control room operators are adequately protected against the effects of accidental release of toxic and radioactive gases and that the Nuclear power plant can be safely operated or shutdown under design basis accident conditions. (GDC19). This study will be forwarded to your staff under separate cover by January 21, 1981.