Generic Letter 82-33

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Updated Summary of Compliance with
Regulatory Guide 1.97 and Response to NRC
Safety Evaluation Report (TAC 51119)

In response to your letter dated March 13, 1991, "NRC Safety Evaluation of the Pilgrim Nuclear Power Station Response to Conformance to Regulatory Guide 1.97," Boston Edison Company provides the attached revision to our summary of compliance. This revision updates Regulatory Guide 1.97 compliance information for the Pilgrim Nuclear Power Station and supersedes our previous submittals in their entirety. All new information in this revision of the summary of compliance is identified with revision bars to aid the reviewer.

The summary of compliance is revised to reflect the redundancy and separation recommended by Regulatory Guides 1.97 and 1.75 for those portions of instrumentation being upgraded to conform with the Category 1 criteria of Regulatory Guide 1.97. Also, the summary of compliance is clarified to indicate our review includes the design and qualification criteria of Regulatory Guide 1.97, Table 1, including display and recording; interfaces; direct measurement; and servicing, testing, and calibration. We will provide further information on the use of electrical isolation devices on Category 1 and 2 instrumentation under separate cover.

Our previous revisions to the summary of compliance included five deviations/clarifications which have not yet been reviewed by the NRC. These unreviewed items are repeated and discussed in the following sections of the summary of compliance:

- II.U Primary Containment Pressure - Suppression Pool
- II.V Effluent Radioactivity - Noble Gases, Turbine Building
- II.W Emergency Ventilation Damper Position
- II.Y Noble Gases and Vent Flow Rates (Common Plant Vent)
- II.Z Status of Standby Power and Other Energy Sources Important to Safety

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We request your review of these items for acceptability. The remaining open items identified in the summary of compliance will be completed in accordance with a schedule to be submitted in the next semi-annual report for the Long Term Program.

G. W. Davis

DMV/clc/3764

Attachment: Summary of Compliance with Regulatory Guide 1.97 for the Pilgrim Nuclear Power Station, Revision 3

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SUMMARY OF COMPLIANCE WITH REGULATORY GUIDE 1.97 FOR THE PILGRIM NUCLEAR POWER STATION

I. SUMMARY

A summary of compliance of the post-accident monitoring instrumentation at the Pilgrim Nuclear Power Station (PNPS) to the design and qualification criteria of Regulatory Guide 1.97, Revision 3 is provided in Table 1. Compliance information for individual primary containment isolation valves is provided in Table 2. Justifications are provided in Section II for all deviations identified on these tables. All open items requiring additional work are identified on these tables with an "O" and are described in more detail in Section III.

The remaining open items in the Regulatory Guide 1.97 Project will be completed in accordance with the completion schedule in the PNPS Long Term Program. In the most recent semi-annual report for the Long Term Program (Reference 8), the schedule for completion was resrinded pending receipt of an NRC safety evaluation of our summary of compliance. With the receipt of the NRC safety evaluation in Reference 10, a revised schedule for completion is planned to be submitted in the next semi-annual report for the Long Term Program.

References are provided for all compliance information previously submitted to the NRC. All new information provided in this revision of the summary of compliance is identified with revision bars.

II. JUSTIFICATIONS FOR DEVIATIONS

A. Drywell Atmosphere Temperature (Type A, Category 1 and Type D, Category 2)

The drywell atmosphere temperature instrumentation at PNPS deviates from the Regulatory Guide 1.97 recommended range of 40 to 440°F.

Although the drywell atmosphere temperature range of 0 to 400°F at PNPS does not correspond exactly with the Regulatory Guide 1.97 recommended range, it does provide sufficient range for monitoring the anticipated design temperature of 281°F, as described in the Final Safety Analysis Report (FSAR). The environmental qualification bounding drywell temperature for a steam line break inside containment of 330°F and the identified peak temperature of approximately 340°F described in the Emergency Operating Procedures are also adequately covered by the 0 to 400°F range. For this reason, the instrument range at PNPS is acceptable (Reference 3). The NRC accepted this deviation in Reference 10, Section 3.3.13.

B. Containment and Drywell Hydrogen Concentration (Type A, Category 1 and Type C, Category 1)

This variable deviates from the Regulatory Guide 1.97 recommended range of 0 to 30 percent hydrogen concentration.

The instrumentation provided at PNPS to measure the concentration of hydrogen in the containment has a range of 0 to 10 percent. This instrumentation was installed at PNPS to meet the requirements of NUREG-0737, Item II.F.1.6, Containment Hydrogen Monitor. As stated in Reference 2 and Reference 10, Section 3.3.9, the NRC concluded the instrumentation provided at PNPS was acceptable as part of their review of NUREG-0737, Item II.F.1.6. Accordingly, the provided instrument range is acceptable.

C. Coolant Level in Reactor Vessel (Type A, Category 1 and Type B, Category 1)

The instrumentation at PNPS to indicate the coolant level in the reactor vessel deviates from the Regulatory Guide 1.97 recommended range of the bottom of the core support plate to the lesser of the top of the vessel or the centerline of the main steamline. At PNPS, the Regulatory Guide 1.97 recommended range would be from 186 to 604 inches above the bottom of the vessel. However, the instrumentation provided at PNPS uses two overlapping sets of Category 1 instrumentation to cover the range of 205 to 532 inches.

The instrument range provided at PNPS gives the operator the reactor vessel level indication needed to perform safety functions under both accident and post-accident conditions. These safety functions include the automatic and manual actions that may be required to restore and maintain reactor vessel water level and to provide core cooling. Level indication below active fuel and greater than the high level trip setpoint of ECCS, as recommended by Regulatory Guide 1.97, does not contribute to information about the accomplishment of plant safety functions for following the course of an accident.

The PNPS reactor vessel water level range is sufficient to keep instruments on scale, utilizing overlapping ranges, at all times when information is required about the accomplishment of plant safety functions for following the course of an accident. The existing level indication range at PNPS meets the intent of the recommendations of Regulatory Guide 1.97 (Reference 3). The NRC accepted this range deviation for coolant level in the reactor vessel in Reference 10, Section 3.3.4.

D. Neutron Flux - APRM, SRM (Type B, Category 1)

Boston Edison has endorsed the BWR Owners' Group position that a fully-qualified, Class 1E post-accident neutron monitoring system is not required (Reference 5). The NRC completed its review of the BWR

Owners' Group position and found it to be unacceptable in the safety evaluation report, dated January 29, 1990 (Reference 6). The BWR Owners' Group submitted a request to the NRC (Reference 9) to appeal the requirement for a fully-qualified, Class 1E post-accident neutron monitoring system. The Regulatory Guide 1.97 project scope at PNPS will accommodate the results of the NRC's decision on this appeal. If the appeal is rejected, the addition of work to install a post-accident neutron monitoring system at PNPS will require a revised project completion schedule.

E. BWR Core Temperature (Type B, Category None and Type C, Category None)

BWR core temperature thermocouples are not provided at PNPS, which deviates from the Regulatory Guide 1.97 recommendation.

BWR core thermocouples would not provide an appropriate diverse indication of water level in the reactor vessel. Specifically, the thermocouples would not respond for at least 10 minutes following the uncovering of the core during a small break LOCA. During this period, the reactor operator would receive conflicting information from existing reactor vessel water level indication. Boston Edison concludes that in-core thermocouples would not provide the diverse indication of reactor vessel water level described by Regulatory Guide 1.97 and they will not be installed at PNPS. The NRC accepted this position in Reference 10, Section 3.3.24.

F. Drywell Sump Level (Type B, Category 3) and Drywell Drain Sumps Level (Type C, Category 3)

Regulatory Guide 1.97 recommends Category 1 instrumentation for these variables. The instrumentation provided at PNPS for these variables is Category 3.

The drywell sumps at PNPS are automatically isolated at the primary containment penetration should an accident signal occur. For small leaks to the drywell sump, the instrumentation is not expected to experience harsh environments during operation. For larger leaks, the drywell sumps fill promptly and the sump drain lines isolate due to the increase in drywell pressure, which negates the drywell sump level and drywell drain sumps level instrumentation. In addition, this instrumentation neither automatically initiates nor alerts the operator to initiate operation of a safety-related system in a post-accident situation. Boston Edison concludes that the Category 3 instrumentation provided at PNPS will provide appropriate monitoring of the parameters of concern. The NRC accepted this conclusion in Reference 2 and Reference 10, Section 3.3.6.

G. Primary Containment Isolation Valve Positions (Type B, Category 1)

1. Channel Redundancy Deviations

a. Check Valves

MO 1201-80, Reactor Water Cleanup (RWCU) Return
MO 1301-49, Reactor Core Isolation Cooling (RCIC) Pump Discharge
MO 2301-8, High Pressure Coolant Injection (HPCI) Pump Discharge
AO 5033A, Normal Nitrogen Makeup to Drywell
AO 5033C, Normal Nitrogen Makeup to Torus
AO 5040A, Torus Vacuum Breaker Isolation Valve
AO 5040B, Torus Vacuum Breaker Isolation Valve
MO 1001-28A and -29A, Low Pressure Coolant Injection (LPCI) Injection
MO 1001-28B and -29B, LPCI Injection

Each of these primary containment isolation valves are located on Class A or B lines which require two isolation valves in series. Check valves, which close on reverse flow, are used in conjunction with the above valves to isolate the lines. Because Regulatory Guide 1.97 specifically excludes check valves from any position indicating requirements, redundant valve position indication will not be provided for these lines. The NRC accepted this position in Reference 10, Section 3.3.7.

b. No Redundant Isolation Valve

MO 4002, Reactor Building Closed Cooling Water (RBCCW) Return

This primary containment isolation valve is located on a closed cooling water line penetrating the primary containment. It requires only one isolation valve. Position indication for the single primary containment isolation valve MO 4002 is provided in the control room. Boston Edison concludes that single control room indication of primary containment isolation valve position is acceptable for this line. The NRC accepted this position in Reference 10, Section 3.3.7.

c. Safety Systems

MO 1400-24A and -25A, Core Spray to Reactor
MO 1400-24B and -25B, Core Spray to Reactor
MO 1001-23A and -26A, Residual Heat Removal (RHR) to Drywell Spray
MO 1001-23B and -26B, RHR to Drywell Spray
MO 1001-34A and -37A, RHR to Suppression Pool Spray
MO 1001-34B and -37B, RHR to Suppression Pool Spray
MO 1001-28A and -29A, LPCI Injection
MO 1001-28B and -29B, LPCI Injection

Regulatory Guide 1.97 states in Table 1, Part 2, Redundancy, that "Within each redundant division of a safety system, redundant monitoring channels are not needed except for steam generator level instrumentation in two-loop plants." Redundant valve position indication is not provided on the redundant isolation valves for the safety systems listed above. Instead, redundancy criteria are applied to each redundant division of these safety systems, as required for reliable system operation. Accordingly, the isolation valves for these safety systems fully meet the redundancy criteria of Regulatory Guide 1.97 and no deviation is identified. The NRC accepted this position in Reference 10, Section 3.3.7.

d. System Design

SV 5081A and SV 5082A, Post-Accident Purge and Vent
SV 5081B and SV 5082B, Post-Accident Purge and Vent
SV 5083A and SV 5084A, Post-Accident Purge and Vent
SV 5083B and SV 5084B, Post-Accident Purge and Vent
SV 5085A and SV 5086A, Post-Accident Purge and Vent
SV 5085B and SV 5086B, Post-Accident Purge and Vent
SV 5087A and SV 5088A, Post-Accident Purge and Vent
SV 5087B and SV 5088B, Post-Accident Purge and Vent
SV 5065-31B and -35B, H_2/O_2 Analyzer and Post-Accident Sampling System (PASS) Supply
SV 5065-33A and -37A, H_2/O_2 Analyzer and PASS Supply
SV 5065-63 and -64, PASS Reactor Water Sample
SV 5065-85 and -86, PASS Reactor Water Sample
SV 5065-24A and -26A, H_2/O_2 Analyzer and PASS Gas Return
SV 5065-25B and -27B, H_2/O_2 Analyzer and PASS Gas Return
SV 5065-13B and -20B, H_2/O_2 Analyzer and PASS Supply
SV 5065-14A and -21A, H_2/O_2 Analyzer and PASS Supply
SV 5065-15B and -22B, H_2/O_2 Analyzer and PASS Supply
SV 5065-11A and -18A, H_2/O_2 Analyzer and PASS Supply
SV 5065-77 and -78, PASS Liquid Return
SV 5065-71 and -72, PASS Liquid Return

These primary containment isolation valves provide position indication from limit switches that are integral parts of the valves. Accordingly, both valve operation and position indication for each valve are powered from the same source. In addition, each of these systems is designed so that the isolation valves on redundant flowpaths are powered by redundant sources to ensure reliable system operation. For these reasons, redundant position indication for isolation valves within each flowpath is not needed for these systems and is not provided. Instead, the system design applies redundancy criteria to each redundant flowpath. Boston Edison concludes redundancy is not applicable for position indication of isolation valves within each flowpath because of the redundancy incorporated in the design of these systems. The NRC accepted this position in Reference 10, Section 3.3.7.

2. Valves Excluded from Regulatory Guide 1.97 Program

a. Disarmed Valves

MO 1001-60 and MO 1001-63, Residual Heat Removal (RHR) Head Spray

These valves have been electrically disarmed in the closed position and do not require valve position indication in the control room to verify primary containment isolation. For this reason, these valves are excluded from the Regulatory Guide 1.97 program. The NRC accepted this position in Reference 10, Section 3.3.7.

b. Control Rod Drive (CRD) Directional Control Valves

FCV 302-120 and -123, CRD Insert
SV 302-121 and -122, CRD Withdraw

These 580 directional control valves, when energized and opened in coordinated pairs, facilitate rod movement either in the insert or withdrawal modes. These valves are normally closed, except during rod movement in normal operation. No position indication is provided for these valves in the control room and they do not receive an automatic primary containment isolation signal (Reference 1).

Because these valves are not used to achieve a scram and are not used in a post-accident situation, no position indication is required. These valves are excluded from the Regulatory Guide 1.97 program. The NRC accepted this position in Reference 2 and Reference 10, Section 3.3.7.

c. Lines That Terminate Below Suppression Pool

MO 1001-36A and B, RHR Test Return
MO 1001-18A and B, RHR Minimum Flow
MO 1301-25, RCIC Pump Suction from Torus
MO 2301-36, HPCI Pump Suction from Torus
MO 1001-7A through -7D, RHR Pump Suction
MO 1400-3A and B, Core Spray Suction

These primary containment isolation valves are located on lines that terminate below the water level of the suppression pool during both normal and accident conditions. No path for gaseous leakage from the containment exists. The position indication of these valves provides no additional information to the operator on the accomplishment of containment isolation. Therefore, these valves are excluded from the Regulatory Guide 1.97 program. The NRC accepted this position in Reference 10, Section 3.3.7.

d. <u>Residual Heat Removal (RHR) Discharge to Radwaste</u>

MO 1001-21 and MO 1001-32

These valves are located upstream of the primary containment isolation valves on the RHR injection line and, therefore, are not relied upon to perform primary containment isolation. However, these valves do receive a primary containment isolation signal to ensure proper valve positioning. These valves are not containment isolation valves and they are excluded from the Regulatory Guide 1.97 program. The NRC accepted this position in Reference 10, Section 3.3.7.

e. <u>Air Supply to Drywell-to-Torus Vacuum Breakers</u>

CV 5046

This normally-closed remote manual valve controls the air supply to the drywell-to-torus vacuum breakers. It is used strictly for testing of the vacuum breakers during each refueling outage. Boston Edison will install administrative controls on this valve to ensure it cannot be inadvertently opened. Administrative controls include mechanical devices to seal or lock the valve closed or to prevent power from being supplied to the valve operator. With the installation of such administrative controls, redundant valve position indication in the control room will not be necessary to verify primary containment isolation. For this reason, this valve is excluded from the Regulatory Guide 1.97 program. The NRC accepted this position in Reference 10, Section 3.3.7.

3. Transversing Incore Probe (TIP) Shear and Ball Valves

736A, 736B, 736C, 736D
737A, 737B, 737C, 737D

Regulatory Guide 1.97 recommends Category 1 instrumentation for the position indication of these primary containment isolation valves. Category 3 position indication is provided for these valves at PNPS.

The TIP primary containment isolation design is commensurate with the importance to safety of isolating that system, and has been previously reviewed and accepted by the NRC on numerous dockets. The TIP guide tubes are normally closed by the TIP ball valves. A TIP scan requires insertion of the TIP probes into the reactor vessel for a period of approximately four hours per month. Over a one-year period, this amounts to less than 2% of the time the plant is operational. In the event of a LOCA, the TIP system design will reliably provide automatic isolation of any open TIP guide tubes by providing automatic retraction of the TIP cable followed by automatic closure of the TIP ball valves. Only in the case that the ball valve fails to automatically close, the shear valve is manually actuated by detonation squibs. However, because the TIP system electrical circuits are not safety grade and not separated, failure to isolate TIP guide tubes could be postulated.

The most likely sequence of events leading to fission product release through the TIP guide tubes has a probability of occurrence of about 5 X 10E-13 per reactor year. Using extremely conservative Regulatory Guide 1.3 source term assumptions and conservative PNPS-unique parameters, the offsite thyroid and whole body doses for this limiting event are below 10CFR100 limits. The extremely low probability of a fission product release, the minimal offsite radiological consequences of the TIP containment isolation failure, and the prohibitive costs involved in upgrading the position indicating circuits for the isolating TIP shear and ball valves support the Boston Edison decision not to upgrade the Category 3 equipment provided for this variable. The NRC accepted this deviation in Reference 10, Section 3.3.7. |

H. Radioactivity Concentration in Circulating Primary Coolant (Type C, Category 3)

The classification of this variable at PNPS as Category 3 deviates from the Regulatory Guide 1.97 recommendation of Category 1.

Instrumentation to monitor radioactivity concentration in circulating primary coolant is designated as Category 3 because no planned operator actions are identified and no operator actions are anticipated based on this variable. The existing Category 3 instrumentation provided by the post-accident sampling system (PASS) adequately measures radioactivity concentration in the coolant to indicate fuel cladding failure. In Reference 2 and Reference 10, | Section 3.3.8, the NRC concluded the alternative instrumentation | provided by PASS was acceptable to monitor this variable.

I. Suppression Chamber Spray Flow (Type D, Category 2) and Drywell Spray Flow (Type D, Category 2)

Regulatory Guide 1.97 recommends dedicated, Category 2 flow indication be provided on both the suppression chamber and drywell spray lines. At PNPS, Category 2 flow indication is provided on the residual heat removal (RHR) injection line which feeds the LPCI, suppression chamber spray, drywell spray, and the suppression chamber cooling lines. PNPS deviates from the Regulatory Guide 1.97 recommendation because dedicated flow indication is not provided on each spray line.

Operation of the suppression chamber and drywell sprays at PNPS requires the operator to manually open valves which divert RHR system flow to the sprays. These valves are normally closed and each is provided with Category 1 valve position indication in the control room. The knowledge of valve positions, coupled with RHR flow indication, assures the operator that flow is being diverted as desired to the suppression chamber spray and the drywell spray.

Additional verification that the suppression chamber and drywell sprays are operating as designed is indirectly provided by the Category 1 instrumentation indicating primary containment pressure. During accident conditions, the emergency operating procedures direct the control room operators to verify primary containment pressure to confirm the operation of the containment spray subsystems. Primary containment pressure indication tells the operator that the containment spray system spargers are operating within 3 to 5 minutes after system initiation. The containment spray system causes the primary containment pressure to decrease rapidly by approximately 16 psig, according to the calculated pressure responses of the containment.

The RHR flow and the injection valve position indications strictly provide the operator with the knowledge that there is flow and the spray path is open. The primary containment pressure indicators assure the operator that the subsystems are working as intended. Boston Edison concludes that the alternative instrumentation described above provides adequate indication of the suppression chamber and drywell spray flows. The NRC accepted these deviations in Reference 10, Sections 3.3.14 and 3.3.25.

J. Main Steamline Isolation Valve (MSIV) Leakage Control System Pressure (Type D, Category 2)

Regulatory Guide 1.97 recommends pressure indication be provided for the MSIV leakage control system. This Category 2, Type D variable is not applicable to PNPS because no designated leakage control system exists on the main steamline isolation valves (Reference 3). The NRC accepted this deviation in Reference 10, Section 3.3.15.

K. Isolation Condenser System Shell-Side Water Level (Type D, Category 2) and Valve Position (Type D, Category 2)

No isolation condenser system is provided in the Mark I containment design at PNPS; therefore, these variables are not applicable to PNPS. The NRC accepted this position in Reference 10, Section 3.3.24.

L. Low Pressure Coolant Injection (LPCI) System Flow (Type D, Category 2)

Regulatory Guide 1.97 recommends dedicated, Category 2 flow indication be provided for the LPCI system injection into the reactor vessel. At PNPS, Category 2 flow indication is provided on the residual heat removal (RHR) injection lines. PNPS deviates from the Regulatory Guide 1.97 recommendation because dedicated flow indication is not provided on the LPCI injection line.

Operation of the LPCI system is verified by RHR flow indication and LPCI injection valve position. The RHR flow indication has a range of 0 to 20,000 gpm. This is adequate to cover the required range of 0 to 110% of the PNPS LPCI system design flow, which is 0 to 15,840 gpm. The injection valves on the LPCI system flow path are provided with Category 1 valve position indication in the control room. The knowledge of valve positions, coupled with RHR flow indications, assures the operator that flow is being sent, as desired, to the LPCI system injection line.

Additional verification that the LPCI system injection is operating as designed is indirectly provided by the Category 1 instrumentation indicating reactor pressure vessel water level. During accident conditions, the emergency operating procedures (EOPs) direct the control room operators to verify reactor vessel water level to confirm the operation of the safety injection systems such as LPCI. Boston Edison concludes that the alternative instrumentation described above provides adequate indication of the LPCI system flow. The NRC accepted this position in Reference 10, Section 3.3.26.

M. Standby Liquid Control System (SLCS) Flow (Type D, Category 3)

Regulatory Guide 1.97 recommends that Category 2 flow indication be provided for the SLCS injection into the reactor vessel. At PNPS, proper operation of the SLCS is monitored by the Category 3 variables SLCS pump discharge header pressure and SLCS storage tank level.

The current design basis for the SLCS recognizes that the system has an importance to safety that is less than the importance to safety of the reactor protection system and the engineered safeguards systems. Accordingly, the instrumentation provided to monitor the operation of the SLCS is considered to be Category 3 (Reference 1).

The indication of SLCS pump discharge header pressure assures the operator that the SLCS pumps are operating as designed. The instrumentation has a range of 0 to 2,000 psig, which sufficiently encompasses the system design pressure of 1500 psig. All valves located between the SLCS storage tank and the reactor pressure vessel are normally locked open, with the exception of check valves and the highly reliable squib valves. A reduction in the SLCS storage tank level indication assures the operator that the SLCS is actually pumping fluid into the reactor vessel. Boston Edison concludes that the alternative instrumentation described above provides adequate indication to monitor the operation of the SLCS (Reference 3). The NRC accepted this position in Reference 10, Section 3.3.17.

N. Standby Liquid Control System (SLCS) Storage Tank Level (Type D, Category 3)

Regulatory Guide 1.97 recommends Category 2 indication be provided for the SLCS storage tank level with a recommended range of top to bottom of the tank. At PNPS, Category 3 instrumentation is provided to monitor this variable with an indicated range of 0 to 4,750 gallons.

The current design basis for the SLCS recognizes that the system has an importance to safety that is less than the importance to safety of the reactor protection system and the engineered safeguards systems. Accordingly, the instrumentation provided to monitor the operation of the SLCS is considered to be Category 3 (Reference 1).

The SLCS storage tank level instrumentation measures from 9 to 135 inches above the tank inside bottom. This range envelopes the General Electric specified range of 9 to 121.25 inches and allows the monitoring of SLCS tank level from Technical Specification required levels down to the effective bottom of the tank. Monitoring SLCS tank level above the upper range provides no additional useful information. In addition, Technical Specifications require SLCS tank level to be verified at least once each day which provides further assurance the correct level is maintained (Reference 3).

Recently, the scale on the SLCS storage tank level was replaced as a result of an enhancement identified by the Detailed Control Room Design Review (DCRDR) Project. The scale for this indication is now calibrated to read from 0 to 4,750 gallons. This new scale meets the intent of the Regulatory Guide 1.97 recommended range of top to bottom of the tank. The NRC accepted this instrumentation in Reference 10, Section 3.3.18.

O. Added Plant Variables (Type D, Category 3)

Regulatory Guide 1.97 provides a recommended minimum set of plant variables that should be monitored during and following an accident. At PNPS, this minimum set is supplemented by the following six plant variables. These variables provide important information to indicate the operation of individual safety systems and other systems important to safety. This Category 3 instrumentation provides indication in the control room for each plant variable. In the case of the additional drywell atmosphere temperature instrumentation, indication is provided in the control room on the EPIC computer.

- Bypass Valve Position
- Condenser Hotwell Level
- Condenser Vacuum
- Condensate Flow
- Recirculation Flow
- Drywell Atmosphere Temperature

P. Reactor Building or Secondary Containment Area Radiation (Type E, Category 2)

Boston Edison's position is that this Regulatory Guide 1.97 recommended variable is not required for the PNPS Mark I containment design.

The exposure rate in the secondary containment will be largely dependent on the radioactivity in the primary containment and the fluids flowing through the emergency core cooling system (ECCS) piping. Local radiation exposure rate monitors could only provide ambiguous indications because there are a large number of pipes in widely scattered locations. The noble gas effluent monitors will provide a more appropriate means of detecting any radioactivity release. For these reasons, area radiation indication in the secondary containment would not provide the operator with useful information and is not required at PNPS (Reference 3). The NRC accepted this position in Reference 10, Section 3.3.21.

Q. <u>Radiation Exposure Rate (Type E, Category 3)</u>

Regulatory Guide 1.97 recommends a range of 10E-1 to 10E4 R/hr for instrumentation to monitor the radiation exposure rate in areas where access is required to service equipment important to safety. The installed instrumentation at PNPS has a range of 10E-5 to 10E-1 R/hr.

Boston Edison will use the existing area radiation monitors and supplement them, on an as-needed basis, with portable radiation monitoring equipment that exists onsite. Because the portable radiation monitoring equipment is fully capable of covering the range of radiation exposure comparable to the emergency condition allowable exposure limits (25 R for health, safety, and property protection and 75 R for life saving), this alternative to hardware modifications meets the Regulatory Guide 1.97 recommendation to monitor access areas required to service equipment important to safety (Reference 3). The NRC accepted this position in Reference 10, Section 3.3.27.

R. <u>Particulates and Halogens (Type E, Category 3)</u>

Regulatory Guide 1.97 recommends that a range of 10^{-3} µCi/cc to 10^{2} µCi/cc be provided for instrumentation to monitor airborne radioactive materials (particulates and halogens) released from the plant. As described below, this is accomplished at PNPS through the combined use of existing instrumentation (multi-channel analyzer systems and radiation monitor survey meters), procedures, and analytical tools in the form of nomograms. The combined ranges provided at PNPS to measure airborne radionuclide concentrations of particulates and halogens released from the plant is from 1×10^{-12} µCi/cc to 3.5×10^{4} µCi/cc, which encompasses the recommended range.

In an accident condition, the identified release points at PNPS for particulates and halogens are the main stack, the reactor building vent, and the turbine building. Releases from the main stack and reactor building vent are sampled through the use of a particulate filter and a charcoal-based iodine collection chamber, installed ahead of the routine effluent monitoring sample lines. For turbine building releases under accident conditions, particulates and halogens are sampled through the use of a portable air sample pump and filter (Reference 5).

Station procedures specify how samples of effluent particulates and halogens will be collected and analyzed under accident conditions from the main stack, reactor building vent, and turbine building. When the sample dose rate is $\leq$ 25 mR/hr, the sample is measured in the onsite radiochemistry lab using a multi-channel analyzer. When the sample dose rate is > 25 mR/hr but $\leq$ 550 mR/hr, the sample may be measured using the multi-channel analyzer if it is first cut down to a section that has a dose rate $\leq$ 25 mR/hr. When the sample dose rate is > 550 mR/hr, the sample cannot be analyzed until it has decayed sufficiently (Reference 5).

The range of detection of the multi-channel analyzer is from 1×10^{-12} µCi/cc to 6.4×10^{-3} µCi/cc. The estimated upper limit of concentration can vary depending on the radionuclide species present and the elapsed time after reactor shutdown.

In addition to the multi-channel analyzer, station procedures require the use of nomograms to estimate sample activity from the sample dose rate. When the sample dose rate falls in the range 10^{-2} mR/hr to 10^{4} R/hr, the nomograms are capable of estimating Iodine-131 inventory on the sample in the range 10^{-1} µCi to 10^{6} mCi (Reference 5). The resultant range of Iodine-131 equivalent effluent plant release concentrations estimated from the nomograms is from 3.5×10^{-6} µCi/cc to 3.5×10^{4} µCi/cc.

High range radiation survey instruments (Teletector or equivalent) are available to measure dose rates up to 10^{3} R/hr. The radiation dose received by plant personnel in the collection, handling, transporting, and analyzing of effluent samples will not exceed the exposure limits of General Design Criterion 19 (Reference 5).

Boston Edison concludes that the overlapping ranges provided by the multi-channel analyzer and the nomograms sufficiently encompass the range recommended by Regulatory Guide 1.97. The NRC accepted this position in Reference 10, Section 3.3.22.

S. <u>Airborne Radiohalogens and Particulates (Type E, Category 3)</u>

Regulatory Guide 1.97 recommends that a range of 10^{-9} µCi/cc to 10^{-3} µCi/cc be provided for instrumentation to measure samples taken in the field for airborne radionuclide concentrations of particulates and halogens in the environs. As described below, this is accomplished at PNPS through the combined use of existing instrumentation (multi-channel analyzer in the onsite radiochemistry lab, SAM-2 sodium iodide detector with a dual-channel analyzer in the field, and radiation monitor survey meters both on and offsite), procedures, and analytical tools in the form of nomograms. The combined range provided at PNPS to measure field samples for airborne radionuclide concentrations of particulates and halogens in the environs is from 1×10^{-12} µCi/cc to 6.4×10^{-3} µCi/cc, which sufficiently encompasses the recommended range.

Field samples of airborne radionuclide concentrations of particulates and halogens in the environs surrounding PNPS can be measured using a SAM-2 detector in the field, a multi-channel analyzer in the onsite radiochemistry lab, or nomograms to estimate Iodine-131 equivalence until the samples can be brought to the onsite lab for analysis by the multi-channel analyzer.

The range of detection for the SAM-2 sodium iodide detectors in the field is from 8 X 10^{-9} µCi/cc to 8 X 10^{-5} µCi/cc. No quantitative measurement of particulate filter paper sample activity is made in the field. However, estimates of Iodine-131 concentrations in the environs can be made in an expeditious manner using a nomogram. When the sample count rate falls in the range of 1 cpm to 10^7 cpm, the nomogram is capable of estimating Iodine-131 inventory on the sample in the range 10^{-6} µCi to 10^2 µCi. The resultant range of Iodine-131 equivalent concentration in the environs estimated from the nomogram is from 10^{-12} µCi/cc to 10^{-4} µCi/cc.

The nomogram is used for quick Iodine-131 airborne concentration estimates by field teams, after which the field samples are brought back to the onsite radiochemistry lab for analysis using the multi-channel analyzer for accurate assessment. The range of detection of the multi-channel analyzer is from 1 X 10^{-12} µCi/cc to 6.4 X 10^{-3} µCi/cc. The multi-channel analyzer system is only used to analyze samples whose contact gamma dose rate is $\leq$ 25 mR/hr, in accordance with station procedures. The radiation dose received by plant personnel in the collection, handling, transporting, and analyzing of field samples will not exceed the exposure limits of General Design Criterion 19.

Boston Edison concludes that the overlapping ranges provided by the SAM-2 detector, the multi-channel analyzer, and the nomograms sufficiently encompass the range recommended by Regulatory Guide 1.97. The NRC accepted this position in Reference 10, Section 3.3.22.

T. Electrical Separation and Isolation

Regulatory Guide 1.97 requires that the redundant or diverse channels of Category 1 equipment be electrically independent and physically separated from each other and from equipment not classified important to safety up to, and including, any isolation device. Regulatory Guide 1.97 references Regulatory Guide 1.75, "Physical Independence of Electric Systems" as the standard for this requirement.

PNPS was designed and constructed to meet the proposed IEEE Standard "Criteria for Nuclear Power Plant Protection Systems," dated March 1968, which predates the issuance of Regulatory Guide 1.75.

The following separation criteria shall be used at PNPS, in accordance with Boston Edison Specification E-347, Section 5.4; Boston Edison Specification E-347A, Sections 5.2.3 and 5.2.4; and

PNPS FSAR Section 8.9.3. These criteria are considered minimum requirements and design guidelines for use in the absence of a confirming design review to support less stringent requirements.

- <u>Cable Tray</u>

  Cable Spreading Room Area:

  The minimum separation distance between redundant Class 1E cable trays shall be 1 foot between trays separated horizontally and 3 feet between trays separated vertically. Where plant arrangement precludes maintaining the minimum separation distance between trays, barriers shall be provided between redundant circuits.

  General Plant Areas:

  The minimum separation distance between redundant Class 1E cable trays shall be 3 feet between trays separated horizontally and 5 feet between trays separated vertically. Where plant arrangement precludes maintaining the minimum separation distance between trays, barriers shall be provided between redundant circuits.

- <u>Enclosed Raceway</u>

  Cable Spreading Room and General Plant Areas:

  The minimum separation distance between redundant Class 1E enclosed raceways shall be 1 inch.

- <u>Internal Wiring</u>

  The minimum separation distance between control panel internal wiring for redundant monitoring channels shall be 6 inches. Where this separation cannot be maintained, a qualified barrier shall be provided as described in IEEE Standard 384-1974.

BECo intends to use the guidance provided in Regulatory Guide 1.75, where applicable, with the following exceptions:

- The cable spreading room area contains instrumentation and control cables along with a 480V load center and a 480V motor control center. The 480V cables are routed in conduit and cable trays and separation shall be maintained in accordance with the requirements of the cable spreading room area, as stated above.

- Raceway markings are located at various intervals to provide adequate raceway identification. Conduits are labeled where they pass through walls and floors, at the conduit destination and origin points, and at other locations along the conduit. The interval between labels may exceed the 15-foot recommendation of Regulatory Guide 1.75.

- Associated cables and raceways are not uniquely identified. Unique identification is not required to ensure electrical separation of redundant systems.

- Electrical isolation shall be accomplished by use of coordinated Class 1E fuses or breakers, in accordance with the proposed IEEE Standard, "Criteria for Nuclear Power Plant Protection Systems," dated March 1968 (Reference 1).

In Reference 10, Section 3.3.2B, the NRC accepted the use of the redundancy and separation criteria for those Category 1 variables not otherwise upgraded to meet Regulatory Guide 1.97 recommendations. The redundancy and separation recommended by Regulatory Guides 1.97 and 1.75 are provided for those portions of instrumentation being upgraded to conform with the Category 1 criteria of Regulatory Guide 1.97. The use of electrical isolation devices is in review. Boston Edison will provide further information on the use of electrical isolation devices on Category 1 and 2 instrumentation at PNPS under separate cover.

In addition to the exceptions stated in our previous submittal, the following exception to the guidance provided in Regulatory Guide 1.75 will be taken:

Individual Class 1E wires and wire bundles within the control board will not be identified in a manner that readily distinguishes between redundant Class 1E wiring and between Class 1E and non Class 1E wiring.

This exception is consistent with the original design basis of the facility.

U. <u>Primary Containment Pressure - Suppression Pool (Type A, Category 1; Type B, Category 1; and Type C, Category 1)</u>

The instrumentation measuring primary containment pressure in the suppression pool at PNPS deviates from the Regulatory Guide 1.97 recommended range of -5 psig to 3 times design pressure for concrete, 4 times design pressure for steel. At PNPS, the suppression pool design pressure is 56 psig at 281 °F which means the Regulatory Guide 1.97 recommended range is -5 to 224 psig. The instrument range provided at PNPS for suppression pool pressure is 0 to 100 psig.

Primary containment pressure in the drywell at PNPS is monitored by instrumentation with ranges of -5 to +5 psig (narrow range) and 0 to 225 psig (wide range). This instrumentation fully meets the Regulatory Guide 1.97 recommended range for primary containment pressure in the drywell. The more limited range of 0 to 100 psig for primary containment pressure in the suppression pool adequately meets the information needs of the operators during any postulated design basis accident at PNPS. The PNPS Emergency Operating Procedures require the operators to take actions to prevent a breach of primary containment based on specific suppression pool pressures within the provided instrument range. Specifically, these procedures require emergency suppression pool venting prior to reaching the suppression pool design pressure. The 0 to 100 psig range for primary containment pressure in the suppression pool provides an adequate margin above this design pressure of 56 psig and the maximum internal pressure of 62 psig. For these reasons, the instrument range for suppression pool pressure at PNPS is acceptable.

V. Effluent Radioactivity - Noble Gases, Turbine Building (Type C, Category 3)

Regulatory Guide 1.97 recommends Category 2 instrumentation be installed to monitor gaseous effluent radioactivity "from buildings or areas where penetrations and hatches are located, e.g., secondary containment and auxiliary buildings and fuel handling buildings that are in direct contact with primary containment." A Category 2 instrument indicates system operating status and most directly indicates the accomplishment of a safety function. Type C instrumentation indicates the potential or actual breach of fission product barriers. Accordingly, Type C, Category 2 instrumentation is provided at PNPS to monitor gaseous effluent radioactivity from the main stack and the reactor building vent. However, instrumentation to monitor gaseous effluent radioactivity from the turbine building at PNPS is downgraded to Type C, Category 3.

Boston Edison uses normal operating range and high range instrumentation to monitor gaseous effluent radioactivity in the main stack and the reactor building vent at PNPS. By indicating the status of the fission product barriers, the main stack and reactor building vent monitors indirectly indicate the operating status of the standby gas treatment system, the augmented off-gas system, and various exhaust systems and the accomplishment of their safety functions. Therefore, these monitors are correctly designated as Type C, Category 2 equipment.

The turbine building operating floor has a low potential for radiological release. Any release from the turbine building basement area or the turbine building ground floor to the turbine building operating floor or adjacent area above Elevation 51 ft is precluded because the turbine building basement and ground floor are maintained at a slightly negative pressure relative to the turbine building operating floor. These releases below the operating floor are routed out the reactor building vent and are monitored by the reactor building vent instrumentation. The expected airborne activity on the turbine building operating floor will normally be below the levels allowed by 10CFR20, Appendix B, Table I. The releases from the turbine building operating floor and the reactor feedwater pump area are expected to be insignificant relative to the releases from the main stack and the reactor building vent. Therefore, no turbine building normal operating range monitoring devices are provided for compliance with Regulatory Guide 1.97 at PNPS.

In the case of a design basis accident coupled with a loss of offsite power, the differential pressure between the turbine building basement and operating floor will be lost. No effluent will flow out of the turbine building because of the loss of power to the turbine building roof exhaust fans. However, releases from the turbine building basement may potentially leak up to the turbine building operating floor and eventually leak out of the turbine building roof. Because of this concern, Boston Edison installed a high range radioactivity monitor on the turbine building operating floor to provide diagnostic information and to provide backup information for the information gathered by field survey teams in developing protective action recommendations.

As described above, the turbine building high range effluent monitor is appropriately designated as Category 3 instrumentation because the turbine building operating floor is not in direct contact with the primary containment or the exhaust from any systems containing potential fission product releases. This monitor does not indicate any system operating status and does not verify the accomplishment of a safety function. As a Category 3 instrument, the turbine building high range effluent monitor provides diagnostic and backup information to monitor potential leakage of radioactive effluent from the turbine building basement up through the turbine building roof exhaust.

Because the turbine building effluent monitor is designated to be Category 3 instrumentation, environmental qualification of this monitor is not required and will not be provided. This revises the commitment made by Boston Edison in Reference 7.

W. <u>Emergency Ventilation Damper Position (Type D, Category 2)</u>

Regulatory Guide 1.97 recommends damper position indication be provided to monitor the operation of emergency ventilation systems during post-accident conditions. At PNPS, the damper position instrumentation monitoring the operation of the reactor building isolation control (RBIC) and control room environmental control (CREC) systems are provided. However, operation of the standby gas treatment (SBGT) system is monitored by the instrumentation that indicates SBGT exhaust flow to the main stack. In conjunction with the RBIC system damper position indication, SBGT exhaust flow indication provides a more effective and reliable method of monitoring SBGT system operation. In accordance with Regulatory Guide 1.97 recommendations, the damper position indication on the RBIC and CREC systems and the SBGT exhaust flow indication will be qualified for their respective environments. For this reason, it is acceptable not to environmentally qualify the instrumentation that indicates SBGT system damper position.

X. Seismic Qualification

As stated in Reference 1, Boston Edison deferred the review of the seismic qualification of accident monitoring instrumentation for Regulatory Guide 1.97 pending the resolution of Unresolved Safety Issue (USI) A-46. In Generic Letter 87-02, dated February 19, 1987, the NRC issued their technical resolution of USI A-46. The generic letter stated equipment must either be qualified using seismic experience data in accordance with procedures developed by the Seismic Qualification Utility Group or by the analysis and testing methods of IEEE Standard 344-1975.

Subsequent to the generic letter, the Seismic Qualification Utility Group submitted a generic implementation procedure (GIP) for NRC approval. The GIP contains evaluation procedures and acceptance criteria for the use of seismic experience data in the resolution of USI A-46. The NRC issued its safety evaluation of the GIP in July, 1988. The GIP applies to plants with construction permits issued prior to 1972 (i.e., plants not originally licensed to IEEE Standard 344-1975 at startup). PNPS is in the group of plants covered by the GIP.

Related to this, IEEE Standard 344 was revised in 1987 to include provisions for the use of seismic experience data to qualify electrical equipment. The standard applies to all plants regardless of age. The NRC endorsed this standard in the latest revision to Regulatory Guide 1.100, "Seismic Qualification of Electric Equipment for Nuclear Power Plants."

In view of these developments, Boston Edison completed a program to verify the seismic qualification of Regulatory Guide 1.97 Category 1 equipment. Equipment purchased and installed to the requirements of IEEE Standard 344-1975 was deemed to be acceptable as is, provided the qualification documentation was readily available and auditable. For other equipment, the program verified the seismic qualification per IEEE Standard 344-1987, Section 9, "Experience," and the Seismic Qualification Utility Group Generic Implementation Procedure, Revision 1, dated November 1988. As indicated in Tables 1 and 2, the seismic qualification of Regulatory Guide 1.97 Category 1 equipment was either found to be acceptable or remains open pending the completion of the additional work described in Section III.B.

Although this program also reviewed the seismic qualification of Regulatory Guide 1.97 Category 2 equipment of safety-related systems, seismic qualification of Category 2 equipment is not required by Regulatory Guide 1.97. Accordingly, Table 1 has been revised to indicate that the seismic qualification of Category 2 equipment is not required. The NRC accepted this seismic qualification program in Reference 10, Section 3.3.1

Y. Noble Gases and Vent Flow Rates (Common Plant Vent) (Type E, Category 2)

Regulatory Guide 1.97 recommends monitoring the noble gases and vent flow rates through the common plant vent (main stack) to allow the continuous assessment of the magnitude of post-accident radiological releases to the environment. Radiological releases from the main stack at PNPS are monitored by overlapping normal operating range and high range radiation monitors. The flow past the high range radiation monitor is monitored at PNPS by Category 2 instrumentation in accordance with the recommendations of Regulatory Guide 1.97. However, the flow instrumentation for the normal operating range radiation monitor is not included in the Regulatory Guide 1.97 program at PNPS.

The flow instrumentation for the normal operating range radiation monitor is located near the top of the main stack and is indicated locally at the main stack. Because no indication in the control room is provided, the flow rate past the normal operating range radiation monitor may not be immediately available post-accident. In this case, the Emergency Dose Assessment Program at PNPS conservatively estimates the flow rate based on the number of stack dilution and standby gas treatment system fans operating. For this reason, indication of flow past the normal operating range radiation monitor is not required to assess the magnitude of post-accident radiological releases at PNPS.

Z. Status of Standby Power and Other Energy Sources Important to Safety (Type D, Category 2)

The standby power systems and other energy sources important to safety at PNPS include the 4160 V distribution system, the emergency diesel generators, the 125 VDC system, and the 250 VDC system. The following instrumentation is used to monitor the status of these standby power systems.

Instrumentation	Range
4160 V Distribution System:	
Ammeters measuring current from startup transformer to Buses A5 and A6	0-2,000 amps
Voltmeters for Buses A5 and A6	0-6,000 volts
Emergency Diesel Generators:	
Diesel Generator 1,	
Voltmeter	0-6,000 volts
Ammeter	0-600 amps

Diesel Generator 2,	
Voltmeter	0-6,000 volts
Ammeter	0-600 amps

<u>125 VDC System</u>:

Battery chargers A & B	
Voltmeters (charger output)	0-150 VDC
Ammeters (charger output)	0-250 amps
Batteries A & B	
Voltmeters (voltage on battery)	0-150 VDC
Ammeters (current in or out of battery)	-150 to 0, 0 to 500 amps

<u>250 VDC System</u>:

Battery charger	
Voltmeters (charger output)	0-300 VDC
Ammeters (charger output)	0-250 amps
Battery	
Voltmeter (voltage on battery)	0-300 VDC
Ammeter (current in or out of battery)	-50 to 0, 0 to 500 amps

This information was initially submitted to the NRC in Reference 5. It has been revised as described below to list the instrumentation used at PNPS to monitor these variables.

- Instrumentation monitoring Buses A1 through A4 on the 4160 V distribution system was removed from this list because Buses A1 through A4 do not power any equipment important to safety.

- The ammeters measuring current from the unit auxiliary transformer and the shutdown transformer were removed from this list because neither transformer would be in use during or while recovering from a design basis accident. The unit auxiliary transformer feeds safety buses from the main generator during normal operation. However, the reactor is shutdown in an accident condition and the main generator would not be available to supply the unit auxiliary transformer. The shutdown transformer would be used as a source of offsite power in an accident condition only upon the loss of both emergency diesel generators. As such, it would not be used during or while recovering from a design basis accident.

- The backup battery chargers for the 125 and 250 VDC systems are not normally in service and would require operator action to place them in operation. Accordingly, the backup battery chargers were removed from this list because they do not constitute automatic standby power sources and are not required to be used during or while recovering from a design basis accident.

- The instrument air system instrumentation was removed from this list because the instrument air supply system is not required to be operable during or while recovering from a design basis accident. Where air or nitrogen is required for safety-related equipment to perform its safety function, accumulators are provided as a backup to ensure the supply of pneumatic energy important to safety during or while recovering from a design basis accident. Some air-operated safety-related equipment at PNPS is not provided with a backup accumulator because it does not require air to perform its safety function.

AA. Effluent Radioactivity - Noble Gases (Main Stack, Reactor Building Vent, and Turbine Building) (Type C, Category 2 and 3)

Regulatory Guide 1.97 recommends instrumentation for this variable with a range of 10^{-6} μCi/cc to 10^{3} μCi/cc. Although the instrument ranges at PNPS nearly match the Regulatory Guide 1.97 recommended range, they do not in all cases meet it. The following table presents the calculated ranges of the effluent radioactivity instrumentation at the three release points at PNPS: the main stack, the reactor building vent, and the turbine building vent. Because no normal effluents pass through the turbine building vent, the high range monitor does not overlap any other effluent radioactivity monitor. The ranges were calculated assuming sampling at a time one-hour post-LOCA and are given both in terms of the actual composition of isotopic mix in the expected effluents at PNPS and in terms of the Xe-133 equivalent to yield the same offsite whole body gamma dose rate.

Effluent Radioactivity Monitor Response Ranges, Halogen and Noble Gas Mix, One-Hour Post-LOCA

Location	Monitor	Actual Mix (μCi/cc)	Xe-133 Equivalent to Yield Same Offsite Whole Body Gamma Dose Rate (μCi/cc)
Main Stack	Low Range	1.2×10^{-5} to 1.2×10^{0}	1.3×10^{-4} to 1.3×10^{1}
	High Range	9.9×10^{-2} to 9.9×10^{3}	1.0×10^{0} to 1.0×10^{5}
Reactor Building Vent	Low Range	1.5×10^{-6} to 1.5×10^{-1}	2.5×10^{-5} to 2.5×10^{0}
	High Range	2.1×10^{-2} to 2.1×10^{3}	3.6×10^{-1} to 3.6×10^{4}
Turbine Building Vent	High Range	1.7×10^{-3} to 1.7×10^{2}	3.1×10^{-2} to 3.1×10^{2}

This range information was originally submitted to the NRC in Reference 3. The NRC accepted this range deviation in Reference 10, Section 3.3.11.

BB. Plant and Environs Radiation (Portable) (Type E, Category 3)

Regulatory Guide 1.97 recommends high range radiation survey instruments be available to measure dose rates up to 10^4 R/hr. However, the high range radiation survey instruments at PNPS only measure dose rates up to 10^3 R/hr. Administrative restraints would prevent entry into areas where radiation levels could cause excessive personnel radiation exposure. Boston Edison concludes the high range of the provided radiation survey instruments at PNPS is acceptable. The NRC accepted this position in Reference 10, Section 3.3.22.

CC. Primary Coolant and Sump (Type E, Category 3)

Regulatory Guide 1.97 recommends providing the capability of obtaining representative liquid and gas samples from within the primary containment for radiological and chemical analysis post-accident. For dissolved gas, the recommended range is zero to 2000 cc/kg. At PNPS, the post-accident sampling system provides the capability of analyzing dissolved gas in the range zero to 400 cc/kg. This conforms with the range recommended by the BWR Owners' Group to meet the post-TMI requirements of NUREG-0737, Item II.B.3 (Reference 1). The NRC accepted this range deviation in Reference 10, Section 3.3.23.

III. OPEN ITEMS

The open items indicated on Tables 1 and 2 require additional work to verify compliance with Regulatory Guide 1.97, Revision 3. Open items related to specific issues are discussed below.

A. Equipment Qualification

Additional work is required on the environmental qualification of the instrumentation and associated equipment listed as open on Tables 1 and 2. Additional information on the work needed to environmentally qualify instrumentation monitoring effluent radioactivity and status of standby power was submitted to the NRC by Reference 7. A revision to the commitment on the environmental qualification of the turbine building effluent monitor is provided in Section II.V.

B. Seismic Qualification

The seismic verification program for Regulatory Guide 1.97 Category 1 equipment at PNPS identified additional work required for the open seismic qualification items listed in Tables 1 and 2. These open items either require minor modifications to the equipment supports or replacement with seismically-qualified equipment. The seismic verification program for Regulatory Guide 1.97 did not review the seismic qualification of some equipment that was already planned to be replaced for other reasons. Instead, seismically-qualified replacement equipment will be provided.

C. Neutron Flux Monitoring

As discussed in Section II.D, the Regulatory Guide 1.97 project scope will accommodate the results of the NRC's decision on the BWR Owners' Group appeal. If the appeal is rejected, the addition of work to install a post-accident neutron monitoring system at PNPS will require a revised project completion schedule.

D. Equipment Identification and Human Factors

In conjunction with the Boston Edison Detailed Control Room Design Review (DCRDR) Project, a human engineering review of Regulatory Guide 1.97-related devices on the main control room panels will be performed in accordance with NUREG-0700. The Regulatory Guide 1.97-related devices on the main control room panels will be marked or identified as such as part of the ongoing control room enhancements activity. These activities are further described in Boston Edison letters to the NRC, dated May 2, 1989 and July 24, 1989. The remaining Regulatory Guide 1.97-related devices outside the main control room panels will also be reviewed and marked in a similar manner. The DCRDR Project is included in the Boston Edison Long Term Program.

E. Channel Redundancy; Channel Availability; Quality Assurance; Display and Recording; and Servicing, Testing, and Calibration

Boston Edison is currently reviewing the compliance of the post-accident monitoring instrumentation at PNPS with the Regulatory Guide 1.97 recommendations for these design criteria. A schedule for completion of this review will be included in our next semi-annual submittal of the Long Term Program.

F. Administrative Controls

As described in Section II.G.2.e, Boston Edison will install administrative controls on CV 5046, Air Supply to Drywell to Torus Vacuum Breakers, to ensure that it cannot be inadvertently opened.

G. Electrical Separation/Isolation and Interfaces

Boston Edison is currently reviewing the compliance of the post-accident monitoring instrumentation at PNPS with the electrical separation and isolation design criteria described in Section II.T.

Boston Edison will provide further information on the use of electrical isolation devices on Category 1 and 2 instrumentation under separate cover.

REFERENCES

1. Letter from W. D. Harrington (BECo) to D. B. Vassallo (NRC), dated November 1, 1984 (BECo 84-187), "Generic Letter 82-33: Regulatory Guide 1.97"

2. Letter from J. A. Zwolinski (NRC) to W. D. Harrington (BECo), dated December 12, 1985 (BECo 1.85.372), "Generic Letter 82-33; Regulatory Guide 1.97 Request for Additional Information"

3. Letter from J. M. Lydon (BECo) to NRC, dated February 10, 1987 (BECo 87-021), "Additional Information Concerning Regulatory Guide 1.97"

4. Letter from D. G. McDonald (NRC) to R. G. Bird (BECo), dated January 24, 1989 (BECo 1.89.044), "Emergency Response Capability, Conformance to Regulatory Guide 1.97, Revision 3, Request for Additional Information"

5. Letter from R. G. Bird (BECo) to NRC, dated April 11, 1989 (BECo 89-053), "Response to Request for Additional Information, Emergency Response Capability, Regulatory Guide 1.97, Revision 3 (TAC 51119)"

6. Letter from F. J. Miraglia (NRC) to S. D. Floyd (BWR Owners' Group), dated January 29, 1990, "BWR Owners' Group Licensing Topical Report 'Position on NRC Regulatory Guide 1.97, Revision 3 Requirements for Post-Accident Neutron Monitoring System' (General Electric Report NEDO-31558)"

7. Letter from R. G. Bird (BECo) to NRC, dated January 11, 1990 (BECo 90-005), "Environmental Qualification of Instrumentation Monitoring Effluent Radioactivity and Status of Standby Power for Regulatory Guide 1.97, Revision 3 (TAC 51119)"

8. Letter from G. W. Davis (BECo) to NRC, dated February 28, 1991 (BECo 91-025), "Long Term Program: Semi-Annual Report" |

9. Letter from G. J. Beck (BWR Owners' Group) to W. T. Russell (NRC), dated August 20, 1990 (BWROG-90107), "Appeal of Staff Position on Upgraded Neutron Flux Monitoring Systems"

10. Letter from S. F. Shankman (NRC) to G. W. Davis (BECo), dated March 13, 1991 (BECo 1.91.070), "Safety Evaluation of the Pilgrim Nuclear Power Station Response to Conformance to Regulatory Guide 1.97 (TAC No. 51119)" |

TABLE 1 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX [3]

Variable	Deviations	Envir Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
TYPE A CAT 1 DRYWELL ATMOSPHERE TEMPERATURE	RANGE deviation. See Section II.A	A	A	A	A	A	A	A	A	AWJ	O	**O**	A	O	**A**
TYPE A CAT 1 CONTAINMENT AND DRYWELL HYDROGEN CONCENTRATION	RANGE deviation. See Section II.B	A	A	O	A	A	A	A	A	AWJ	O	**O**	A	O	**A**
TYPE A CAT 1 CONTAINMENT AND DRYWELL OXYGEN CONCENTRATION		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**
TYPE A CAT 1 PRIMARY CONTAINMENT PRESSURE - DRYWELL		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**
TYPE A CAT 1 PRIMARY CONTAINMENT PRESSURE - SUPPRESSION POOL	RANGE deviation. See Section II.U	A	A	O	O	A	A	A	O	AWJ	O	**O**	A	O	**A**
TYPE A CAT 1 RCS PRESSURE		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**
TYPE A CAT 1 COOLANT LEVEL IN REACTOR VESSEL	RANGE deviation. See Section II.C	A	A	O	A	A	A	A	O	AWJ	O	**O**	O	O	**A**

A - Acceptable, meets the RG1.97 design and qualification criteria.
O - Open, see descriptions in Section III.
AWJ - Acceptable with justification.
NR - Not required, no specific provision required in RG1.97 Table 1.
NA - Not applicable for this variable.

1 See Section II.T for Boston Edison's position on compliance with electrical separation and isolation design criteria.

2 See Section II.X for details of the seismic verification program at PNPS.

3 Recent revisions are shown in **Boldface.**

TABLE 1 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX [3]

Variable	Deviations	Envir Qual	Seismic Qual 2	Separation/ Isolation 1	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
TYPE A **CAT** 1 SUPPRESSION POOL WATER LEVEL		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**
TYPE A **CAT** 1 SUPPRESSION POOL WATER TEMPERATURE		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**
TYPE B **CAT** 1 NEUTRON FLUX - APRM	Awaitng NRC action on BWROG appeal. See Section II.D	O	O	O	A	O	A	O	A	A	O	**O**	O	O	**A**
TYPE B **CAT** 1 NEUTRON FLUX - SRM	Awaitng NRC action on BWROG appeal. See Section II.D	O	O	O	A	O	A	O	A	A	O	**O**	O	O	**A**
TYPE B **CAT** 3 CONTROL ROD POSITION		NR	NR	NR	NR	NR	NR	A	A	A	NR	**NR**	O	O	**A**
TYPE B **CAT** 3 RCS SOLUBLE BORON CONCENTRATION		NR	NR	NR	NR	NR	NR	A	**NA**	A	NR	**NR**	**A**	**NA**	**NA**
TYPE B **CAT** 1 COOLANT LEVEL IN REACTOR VESSEL	RANGE deviation. See Section II.C	A	A	O	A	A	A	A	O	AWJ	O	**O**	O	O	**A**
TYPE B **CAT** None BWR CORE TEMPERATURE	Not included in PNPS RG1.97 program. See Section II.E														
TYPE B **CAT** 1 RCS PRESSURE		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**
TYPE B **CAT** 1 DRYWELL PRESSURE		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**

TABLE 1 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX [3]

Variable	Deviations	Envir Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
TYPE B **CAT** 3 DRYWELL SUMP LEVEL	Downgraded variable from Cat 1 to Cat 3. See Section II.F	NR	NR	NR	NR	NR	NR	A	A	A	NR	**NR**	O	O	**A**
TYPE B **CAT** 1 PRIMARY CONTAINMENT PRESSURE - DRYWELL		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**
TYPE B **CAT** 1 PRIMARY CONTAINMENT PRESSURE - SUPPRESSION POOL	RANGE deviation. See Section II.U	A	A	O	O	A	A	A	O	AWJ	O	**O**	A	O	**A**
TYPE B **CAT** 1 PRIMARY CONTAINMENT ISOLATION VALVE POSITION	See Table 2 for PCIV compliance matrix and Section II.G for deviations.														
TYPE C **CAT** 3 RADIOACTIVITY CONCENTRATION IN CIRCULATING PRIMARY COOLANT	Downgraded variable from Cat 1 to Cat 3. See Section II.H	NR	NR	NR	NR	NR	NR	A	**NA**	A	NR	**NR**	**A**	**NA**	**NA**
TYPE C **CAT** 3 ANALYSIS OF PRIMARY COOLANT		NR	NR	NR	NR	NR	NR	A	**NA**	A	NR	**NR**	**A**	**NA**	**NA**
TYPE C **CAT** None BWR CORE TEMPERATURE	Not included in PNPS RG1.97 program. See Section II.E														
TYPE C **CAT** 1 RCS PRESSURE		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**

TABLE 1 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX[3]

Variable	Deviations	Envir Qual	Seismic Qual 2	Separation/ Isolation 1	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
TYPE C **CAT** 3 PRIMARY CONTAINMENT AREA RADIATION		NR	NR	NR	NR	NR	NR	A	A	A	NR	NR	O	O	A
TYPE C **CAT** 3 DRYWELL DRAIN SUMPS LEVEL	Downgraded variable from Cat 1 to Cat 3. See Section II.F	NR	NR	NR	NR	NR	NR	A	A	A	NR	NR	O	O	A
TYPE C **CAT** 1 SUPPRESSION POOL WATER LEVEL		A	A	O	A	A	A	A	A	A	O	O	A	O	A
TYPE C **CAT** 1 DRYWELL PRESSURE		A	A	O	A	A	A	A	A	A	O	O	A	O	A
TYPE C **CAT** 1 RCS PRESSURE		A	A	O	A	A	A	A	A	A	O	O	A	O	A
TYPE C **CAT** 1 PRIMARY CONTAINMENT PRESSURE - DRYWELL		A	A	O	A	A	A	A	A	A	O	O	A	O	A
TYPE C **CAT** 1 PRIMARY CONTAINMENT PRESSURE - SUPPRESSION POOL	RANGE deviation. See Section II.U	A	A	O	O	A	A	A	O	AWJ	O	O	A	O	A
TYPE C **CAT** 1 CONTAINMENT AND DRYWELL HYDROGEN CONCENTRATION	RANGE deviation. See Section II.B	A	A	O	A	A	A	A	A	AWJ	O	O	A	O	A
TYPE C **CAT** 1 CONTAINMENT AND DRYWELL OXYGEN CONCENTRATION		A	A	O	A	A	A	A	A	A	O	O	A	O	A

TABLE 1 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX [3]

Variable	Deviations	Envir Qual	Seismic Qual 2	Separation/ Isolation 1	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
TYPE C CAT 3 CONTAINMENT EFFLUENT RADIOACTIVITY - NOBLE GASES		NR	NR	NR	NR	NR	NR	A	A	A	NR	**NR**	O	O	**A**
TYPE C CAT 2 EFFLUENT RADIOACTIVITY - NOBLE GASES (MAIN STACK AND REACTOR BLDG VENT)	**RANGE deviation. See Section II.AA**	O	NR	NR	NR	A	A	O	A	**AW,J**	O	**O**	A	O	**A**
TYPE C CAT 3 EFFLUENT RADIOACTIVITY - NOBLE GASES (TURBINE BLDG)	Downgraded variable from Cat 2 to Cat 3. See Section II.V **RANGE deviation. See Section II.AA**	NR	NR	NR	NR	NR	NR	A	A	**AW,J**	NR	**NR**	O	O	**A**
TYPE D CAT 3 MAIN FEEDWATER FLOW		NR	NR	NR	NR	NR	NR	A	A	A	NR	**NR**	O	O	**A**
TYPE D CAT 3 CONDENSATE STORAGE TANK LEVEL		NR	NR	NR	NR	NR	NR	A	A	A	NR	**NR**	O	O	**A**
TYPE D CAT 2 SUPPRESSION CHAMBER SPRAY FLOW	RHR system flow and RHR to suppression pool spray valve position used. See Section II.I														
TYPE D CAT 2 DRYWELL PRESSURE		A	NR	NR	NR	A	A	A	A	A	NR	**O**	**A**	O	**A**
TYPE D CAT 2 SUPPRESSION POOL WATER LEVEL		A	NR	NR	NR	A	A	A	A	A	NR	**O**	**A**	O	**A**

TABLE 1 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX [3]

Variable	Deviations	Envir Qual	Seismic Qual 2	Separation/ Isolation 1	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
TYPE D **CAT** 2 SUPPRESSION POOL WATER TEMPERATURE		A	NR	NR	NR	A	A	A	A	A	NR	O	A	O	A
TYPE D **CAT** 2 DRYWELL ATMOSPHERE TEMPERATURE	RANGE deviation. See Section II.A	A	NR	NR	NR	A	A	A	A	AWJ	NR	O	A	O	A
TYPE D **CAT** 2 DRYWELL SPRAY FLOW	RHR system flow and RHR to drywell spray valve position used. See Section II.I														
TYPE D **CAT** 2 MSIV'S LEAKAGE CONTROL SYSTEM PRESSURE	Not included in the PNPS Mark I design. See Section II.J														
TYPE D **CAT** 2 PRIMARY SYSTEM SAFETY RELIEF VALVE POSITIONS		O	NR	NR	NR	A	A	O	A	A	NR	O	O	O	A
TYPE D **CAT** 2 ISOLATION CONDENSER SYSTEM SHELL-SIDE WATER LEVEL	Not included in the PNPS Mark I design. See Section II.K														
TYPE D **CAT** 2 ISOLATION CONDENSER SYSTEM VALVE POSITION	Not included in the PNPS Mark I design. See Section II.K														
TYPE D **CAT** 2 RCIC FLOW		O	NR	NR	NR	A	A	O	A	A	NR	O	O	O	A
TYPE D **CAT** 2 HPCI FLOW		A	NR	NR	NR	A	A	O	A	A	NR	O	O	O	A

TABLE 1 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX [3]

Variable	Deviations	Envir Qual	Seismic Qual 2	Separation/ Isolation 1	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter- faces	Service Testing & Calib	Human Factors	Direct Meas
TYPE D **CAT** 2 CORE SPRAY SYSTEM FLOW		O	NR	NR	NR	A	A	O	A	A	NR	**O**	O	O	**A**
TYPE D **CAT** 2 LPCI SYSTEM FLOW	RHR system flow and LPCI injection valve position used. See Section II.L														
TYPE D **CAT** 3 SLCS FLOW	Downgraded variable from Cat 2 to Cat 3. **DIRECT MEASUREMENT deviation.** See Section II.M	NR	NR	NR	NR	NR	NR	A	A	A	NR	**NR**	O	O	**AWJ**
TYPE D **CAT** 3 SLCS STORAGE TANK LEVEL	Downgraded variable from Cat 2 to Cat 3. **RANGE deviation.** See Section II.N	NR	NR	NR	NR	NR	NR	A	A	**AWJ**	NR	**NR**	O	O	**A**
TYPE D **CAT** 2 RHR SYSTEM FLOW		O	NR	NR	NR	A	A	O	A	A	NR	**O**	O	O	**A**
TYPE D **CAT** 2 RHR HEAT EXCHANGER OUTLET TEMPERATURE		O	NR	NR	NR	A	A	O	A	A	NR	**O**	O	O	**A**
TYPE D **CAT** 2 COOLING WATER TEMPERATURE TO ESF SYSTEM COMPONENTS		O	NR	NR	NR	A	A	O	A	A	NR	**O**	O	O	**A**
TYPE D **CAT** 2 COOLING WATER FLOW TO ESF SYSTEM COMPONENTS		O	NR	NR	NR	A	A	O	A	A	NR	**O**	O	O	**A**

TABLE 1 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX [3]

Variable	Deviations	Envir Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
TYPE D **CAT** 3 HIGH RADIOACTIVITY LIQUID TANK LEVEL		NR	NR	NR	NR	NR	NR	A	A	A	NR	**NR**	O	O	**A**
TYPE D **CAT** 2 EMERGENCY VENTILATION DAMPER POSITION	**DIRECT MEASUREMENT** deviation. See Section II.W	O	NR	NR	NR	A	A	O	A	A	NR	**O**	O	O	**AWJ**
TYPE D **CAT** 2 STATUS OF STANDBY POWER AND OTHER SOURCES OF ENERGY IMPORTANT TO SAFETY	See Section II.Z for additional information.	O	NR	NR	NR	A	A	O	A	A	NR	**O**	O	O	**A**
TYPE D **CAT** 3 BYPASS VALVE POSITION	Added plant specific variable for more information. See Section II.O	NR	NR	NR	NR	NR	NR	A	A	NR	NR	**NR**	O	O	**A**
TYPE D **CAT** 3 CONDENSER HOTWELL LEVEL	Added plant specific variable for more information. See Section II.O	NR	NR	NR	NR	NR	NR	A	A	NR	NR	**NR**	O	O	**A**
TYPE D **CAT** 3 CONDENSER VACUUM	Added plant specific variable for more information. See Section II.O	NR	NR	NR	NR	NR	NR	A	A	NR	NR	**NR**	O	O	**A**
TYPE D **CAT** 3 CONDENSATE FLOW	Added plant specific variable for more information. See Section II.O	NR	NR	NR	NR	NR	NR	A	A	NR	NR	**NR**	O	O	**A**
TYPE D **CAT** 3 RECIRCULATION FLOW	Added plant specific variable for more information. See Section II.O	NR	NR	NR	NR	NR	NR	A	A	NR	NR	**NR**	O	O	**A**

TABLE 1 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX [3]

Variable	Deviations	Envir Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
TYPE D **CAT** 3 DRYWELL ATMOSPHERE TEMPERATURE	Added plant specific variable for more information. See Section II.O	NR	NR	NR	NR	NR	NR	A	A	NR	NR	**NR**	O	O	**A**
TYPE E **CAT** 1 PRIMARY CONTAINMENT AREA RADIATION - HIGH RANGE		A	A	O	A	A	A	A	A	A	**A**	**O**	A	O	**A**
TYPE E **CAT** 2 REACTOR BUILDING OR SECONDARY CONTAINMENT AREA RADIATION	Not required at PNPS. See Section II.P														
TYPE E **CAT** 3 RADIATION EXPOSURE RATE	RANGE deviation. See Section II.Q	NR	NR	NR	NR	NR	NR	A	A	AWJ	NR	**NR**	O	O	**A**
TYPE E **CAT** 2 NOBLE GASES AND VENT FLOW RATES (COMMON PLANT VENT)	See Section II.Y for additional information.	O	NR	NR	NR	A	A	O	**A**	A	NR	**O**	**A**	O	**A**
TYPE E **CAT** 3 PARTICULATES AND HALOGENS	See Section II.R for additional information on range.	NR	NR	NR	NR	NR	NR	A	**NA**	A	NR	**NR**	**A**	**NA**	**NA**
TYPE E **CAT** 3 AIRBORNE RADIOHALOGENS AND PARTICULATES	See Section II.S for additional information on range.	NR	NR	NR	NR	NR	NR	A	**NA**	A	NR	**NR**	**A**	**NA**	**NA**

TABLE 1 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX [3]

Variable	Deviations	Envir Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
TYPE E **CAT** 3 PLANT AND ENVIRONS RADIATION (PORTABLE)	RANGE deviation. See Section II.BB	NR	NR	NR	NR	NR	NR	A	NA	AWJ	NR	NR	A	NA	NA
TYPE E **CAT** 3 PLANT AND ENVIRONS RADIOACTIVITY (PORTABLE)		NR	NR	NR	NR	NR	NR	A	NA	A	NR	NR	A	NA	NA
TYPE E **CAT** 3 METEOROLOGY		NR	NR	NR	NR	NR	NR	A	A	A	NR	NR	O	O	A
TYPE E **CAT** 3 PRIMARY COOLANT AND SUMP	RANGE deviation. See Section II.CC	NR	NR	NR	NR	NR	NR	A	A	AWJ	NR	NR	A	NA	NA
TYPE E **CAT** 3 CONTAINMENT AIR		NR	NR	NR	NR	NR	NR	A	A	A	NR	NR	A	NA	NA

TABLE 2 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX FOR PRIMARY CONTAINMENT ISOLATION VALVES [3]

Valves	Deviations	Envir Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
SV5081A SV5082A POST-ACCIDENT PURGE AND VENT	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
SV5081B SV5082B POST-ACCIDENT PURGE AND VENT	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
AO5033B AO5035A DRYWELL/ TORUS PURGE		O	O	O	A	A	A	O	A	A	O	O	A	O	A
AO5035A AO5035B DW/ PURGE MAKEUP		O	O	O	A	A	A	A	A	A	O	O	A	O	A
SV5085A SV5086A POST-ACCIDENT PURGE AND VENT	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
SV5085B SV5086B POST-ACCIDENT PURGE AND VENT	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
AO5033A (Check valve 9-CK-340) NORMAL N2 MAKEUP TO DRYWELL	REDUNDANCY deviation. See Section II.G.1.a	O	O	O	AWJ	A	A	O	A	A	O	O	A	O	A
SV5065-33A SV5065-37A H2/O2 ANALYZER AND PASS SUPPLY	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
CV5065-91 CV5065-92 C-19 O2 ANALYZER RETURN		O	O	O	A	A	A	A	A	A	O	O	A	O	A

TABLE 2 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX FOR PRIMARY CONTAINMENT ISOLATION VALVES

Valves	Deviations	Envir Qual	Seismic Qual 2	Separation/ Isolation 1	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Interfaces	Service Testing & Calib	Human Factors	Direct Meas
MO1201-2 MO1201-5 RWCU SUCTION		O	O	O	A	A	A	A	A	A	O	O	A	O	A
SV5065-31B SV5065-35B H2 /O2 ANALYZER AND PASS SUPPLY	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	O	A	A	O	O	A	O	A
MO1400-24A MO1400-25A CORE SPRAY TO REACTOR	REDUNDANCY deviation. See Section II.G 1.c	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
MO1400-24B MO1400-25B CORE SPRAY TO REACTOR	REDUNDANCY deviation. See Section II.G.1.c	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
MO1001-60 MO1001-63 RHR HEAD SPRAY	Disarmed valves. Not part of RG1.97 program. See Section II.G.2.a														
AO7017A AO7017B R/W COLLECTION AND D/W FLOOR SUMP		O	O	O	A	O	A	O	A	A	O	O	A	O	A
AO7011A AO7011B R/W COLLECTION AND D/W FLOOR SUMP		O	O	O	A	O	A	O	A	A	O	O	A	O	A
MO4002 RBCCW RETURN	REDUNDANCY deviation. See Section II.G.1.b	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
AO5043A AO5043B DRYWELL 2" EXHAUST BYPASS		O	O	O	A	A	A	O	A	A	O	O	A	O	A
AO5044A AO5044B DRYWELL PURGE EXHAUST		O	A	O	A	A	A	O	A	A	O	O	A	O	A

TABLE 2 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX FOR PRIMARY CONTAINMENT ISOLATION VALVES [3]

Valves	Deviations	Envir Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
AO203-1A AO203-2A MSIV LINE "A"		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**
AO203-1B AO203-2B MSIV LINE "B"		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**
AO203-1C AO203-2C MSIV LINE "C"		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**
AO203-1D AO203-2D MSIV LINE "D"		A	A	O	A	A	A	A	A	A	O	**O**	A	O	**A**
MO220-1 MO220-2 MAIN STEAM DRAIN		O	O	O	A	A	A	A	A	A	O	**O**	A	O	**A**
MO1201-80 (Check valve 6-58A) RWCU RETURN	REDUNDANCY deviation. See Section II.G.1.a	O	A	O	AWJ	A	A	A	A	A	O	**O**	A	O	**A**
MO1301-49 (Check valve 1301-50) RCIC PUMP DISCHARGE	REDUNDANCY deviation. See Section II.G.1.a	O	A	O	AWJ	A	A	O	A	A	O	**O**	A	O	**A**
MO2301-8 (Check valve 2301-7) HPCI PUMP DISCHARGE	REDUNDANCY deviation. See Section II.G.1.a	O	A	O	AWJ	A	A	A	A	A	O	**O**	A	O	**A**
MO1001-47 MO1001-50 RHR S/D COOLING		O	O	O	A	A	A	A	A	A	O	**O**	A	O	**A**

A - Acceptable, meets the RG1.97 design and qualification criteria.
O - Open, see descriptions in Section III.
AWJ - Acceptable with justification.
NR - Not required, no specific provision required in RG1.97 Table 1.
NA - Not applicable for this variable.

[1] See Section II.T for Boston Edison's position on compliance with electrical separation and isolation design criteria.

[2] See Section II.X for details of the seismic verification program at PNPS.

[3] Recent revisions are shown in **Boldface.**

TABLE 2 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX FOR PRIMARY CONTAINMENT ISOLATION VALVES [3]

Valves	Deviations	Envir Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
737A, 737B, 737C, 737D TIP BALL VALVES	Downgraded variable from Cat 1 to Cat 3. See Section II.G.3.	NR	NR	NR	NR	NR	NR	A	A	A	NR	O	O	O	A
736A,736B, 736C,736D TIP SHEAR VALVES	Downgraded variable from Cat 1 to Cat 3. See Section II.G.3.	NR	NR	NR	NR	NR	NR	A	A	A	NR	O	O	O	A
FCV302-120 FCV302-123 CRD WITHDRAW	Not part of RG1.97 program. See Section II.G.2.b														
SV302-121 SV302-122 CRD WITHDRAW	Not part of RG1.97 program. See Section II.G.2.b														
MO1001-23A MO1001-26A RHR TO DRYWELL SPRAY	REDUNDANCY deviation. See Section II.G.1.c	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
MO1001-23B MO1001-26B RHR TO DRYWELL SPRAY	REDUNDANCY deviation. See Section II.G.1.c	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
SV5065-63 SV5065-64 PASS REACTOR WATER SAMPLE	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
SV5065-85 SV5065-86 PASS REACTOR WATER SAMPLE	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
AO220-44 AO220-45 REACTOR SAMPLE LINE		O	O	O	A	A	A	A	A	A	O	O	A	O	A
SV5065-24A SV5065-26A H2/O2 ANALYZER AND PASS GAS RETURN	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A

TABLE 2 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX FOR PRIMARY CONTAINMENT ISOLATION VALVES [3]

Valves	Deviations	Envir Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
SV5065-13B SV5065-20B H2/O2 ANALYZER AND PASS SUPPLY	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
MO1001-28A MO1001-29A (Check valve 1001-68A) LPCI INJECTION	REDUNDANCY deviation. See Section II.G.1.a and II.G.1.c	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
MO1001-28B MO1001-29B (Check valve 1001-68B) LPCI INJECTION	REDUNDANCY deviation. See Section II.G.1.a and II.G.1.c	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
MO2301-4 MO2301-5 HPCI TURBINE STEAM SUPPLY		O	O	O	A	A	A	A	A	A	O	O	A	O	A
MO1301-16 MO1301-17 RCIC STEAM TO TURBINE		O	O	O	A	A	A	A	A	A	O	O	A	O	A
SV5065-14A SV5065-21A H2/O2 ANALYZER AND PASS SUPPLY	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
AO5033B AO5036A DRYWELL/ TORUS PURGE		O	O	O	A	A	A	O	A	A	O	O	A	O	A
AO5036A AO5036B TORUS PURGE INLET		O	A	O	A	A	A	A	A	A	O	O	A	O	A
SV5087A SV5088A POST-ACCIDENT PURGE AND VENT	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A

TABLE 2 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX FOR PRIMARY CONTAINMENT ISOLATION VALVES [3]

Valves	Deviations	Envir Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
SV5087B SV5088B POST-ACCIDENT PURGE AND VENT	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
AO5033C (Check valve 9-CK-341) NORMAL N2 MAKEUP TO SUPPRESSION POOL	REDUNDANCY deviation. See Section II.G.1.a	O	O	O	AWJ	A	A	O	A	A	O	O	A	O	A
MO1001-36A MO1001-36B RHR TEST RETURNS	Terminate below suppression pool. Not part of RG1.97 program. See Section II.G.2.c														
MO1001-18A MO1001-18B RHR MINIMUM FLOW	Terminate below suppression pool. Not part of RG1.97 program. See Section II.G.2.c														
MO1001-34A MO1001-37A RHR TO SUPPRESSION POOL SPRAY	REDUNDANCY deviation. See Section II.G.1.c	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
MO1001-34B MO1001-37B RHR TO SUPPRESSION POOL SPRAY	REDUNDANCY deviation. See Section II.G.1.c	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
MO2301-33 MO2301-34 HPCI TURBINE EX VAC BRKR		O	O	O	A	A	A	A	A	A	O	O	A	O	A
CV9068A CV9068B HPCI GLAND SEAL CONDENSER		O	O	O	O	O	O	O	O	O	O	O	O	O	O

TABLE 2 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX FOR PRIMARY CONTAINMENT ISOLATION VALVES [3]

Valves	Deviations	Envir Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
MO1301-25 RCIC PUMP SUCTION FROM TORUS	Terminate below suppression pool. Not part of RG1.97 program. See Section II.G.2.c														
MO2301-36 HPCI PUMP SUCTION FROM TORUS	Terminate below suppression pool. Not part of RG1.97 program. See Section II.G.2.c														
MO1001-7A,7B,7C,7D RHR PUMP SUCTION	Terminate below suppression pool. Not part of RG1.97 program. See Section II.G.2.c														
AO5040A (Check valve X-212A) TORUS VACUUM BREAKERS ISOLATION	REDUNDANCY deviation. See Section II.G.1.a	O	O	O	AWJ	A	A	O	A	A	O	O	A	O	A
AO5041A AO5041B TORUS EXHAUST BYPASS		O	O	O	A	A	A	A	A	A	O	O	A	O	A
AO5042A AO5042B TORUS MAIN EXHAUST		O	A	O	A	A	A	A	A	A	O	O	A	O	A
AO5025 AO5042B DIRECT TORUS VENT ISOLATION		O	A	O	A	A	A	A	A	A	O	O	A	O	A
SV5083A SV5084A POST-ACCIDENT PURGE AND VENT	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
SV5083B SV5084B POST-ACCIDENT PURGE AND VENT	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A

TABLE 2 - PNPS REGULATORY GUIDE 1.97 COMPLIANCE MATRIX FOR PRIMARY CONTAINMENT ISOLATION VALVES [3]

Valves	Deviations	Env'r Qual	Seismic Qual [2]	Separation/ Isolation [1]	Channel Redund	Power Source	Channel Avail	QA	Display/ Record	Range	Equip ID	Inter-faces	Service Testing & Calib	Human Factors	Direct Meas
AO5040B (Check valve X-212B) TORUS VACUUM BREAKERS ISOLATION	REDUNDANCY deviation. See Section II.G.1.a	O	O	O	AWJ	A	A	O	A	A	O	O	A	O	A
SV5065-15B SV5065-22B H2/O2 ANALYZER AND PASS SUPPLY	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
CV5046 (Check valve 31-CK-434) AIR TO DW TO TORUS VACUUM BREAKERS	Not part of RG1.97 program. See Section II.G.2.e and Sectio: III.F.														
SV5065-77 SV5065-78 PASS LIQUID RETURN	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
SV5065-71 SV5065-72 PASS LIQUID RETURN	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
SV5065-11A SV5065-18A H2/O2 ANALYZER AND PASS SUPPLY	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
SV5065-25B SV5065-27B H2/O2 ANALYZER AND PASS GAS RETURN	REDUNDANCY deviation. See Section II.G.1.d	O	A	O	AWJ	A	A	A	A	A	O	O	A	O	A
MO1400-3A MO1400-3B CORE SPRAY SUCTION	Terminate below suppression pool. Not part of RG1.97 program. See Section II.G.2.c														
MO1001-21 MO1001-32 RHR DISCHARGE TO RADWASTE	Not part of RG1.97 program. See Section II.G.2.d														