FINAL REPORT

SAFETY EVALUATION REPORT FOR THE DRYWELL EVENT

BOSTON EDISON COMPANY

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SUMMARY

On September 26, 1981, during a routine shutdown and cool-down, Yarway level instrumentation experienced oscillations followed by high level isolation and low level scram signals as reported in LER 81-55/)1T-0 by Boston Edison Company Pilgrim Nuclear Power Station, Docket No. 50-293. The cause of this event has been determined to be higher than normal ambient drywell temperature levels due to inadequate drywell cooling. Degraded drywell cooling system performance was the result of several deficiencies in maintenance-related actions since 1974 which led to progressive system deterioration.

The following conclusions are presented:

1. Elevated drywell temperatures did not adversely affect FSAR Chapter 14 or Amendment 20 analyses.

2. Neither the steel liner nor concrete structure of the drywell was significantly affected by the elevated drywell temperatures.

3. Safety functions of drywell components required for plant shutdown, accident mitigation, and transient response were not jeopardized.

4. A detailed analysis of temperature effects on drywell components is presented on a component-by-component basis.

This report is presented in three parts:

I. Event Description and Boston Edison Response.

The drywell cooling system is briefly described for background information, followed by a summary of maintenance-related deficiencies which caused system deterioration. The conservative estimate of drywell temperature history that formed the basis of component material degradation analysis is presented, and actions taken to reduce ambient drywell temperature are listed.

II. Consequences of the Drywell Temperature Event

The effects of elevated temperatures on reactor water level instrumentation are discussed and a study performed by General Electric of elevated temperature effects upon safety analyses is summarized. The drywell structure is described and the effects of the event upon the steel liner and surrounding concrete are briefly discussed. Results of analyses addressing temperature affects upon drywell components required for plant shutdown, accident mitigation, and transient response are summarized.

III. Equipment Analysis

The technical approach followed in analyzing the effects of elevated temperature upon individual components is discussed. Results of these studies, together with the approach to deficiency resolution and justification for continued use are presented on a component by component basis.

It should be noted that this SER addresses only the drywell event and the consequences/corrective actions associated with this event. Qualification deficiencies for drywell equipment which have been identified in the NRC Safety Evaluation Report on BECo's 79-01B submittal are addressed in a separate BECo response to the NRC. However, BECo recognizes that the drywell event and the 79-01B issues are interrelated since safety-related electrical equipment must be addressed in each case. BECo's approach is to address the drywell high temperature (DHT) as a separate deficiency in much the same way that aging (A) is addressed as a 79-01B deficiency. In BECo's response to the NRC on 79-01B deficiencies, DHT has been added to the list of other deficiencies for drywell equipment and addressed accordingly-drawing upon the evaluations, corrective actions and justifications for continued operation presented in this report.

Safety Evaluation Report
for the Drywell Event

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Section I - Event Description and BECo Response

I-1 Background

PNPS consists of one 687 MWe GE Boiling Water Reactor (BWR 3) with a Mark I containment. It was built by Bechtel, with NSS systems supplied by General Electric. Commercial operation began in June 1972.

The Drywell Cooling System is described in the following excerpt from section 5.2.3.7 of the FSAR:

> "The primary containment (drywell) cooling system utilizes eight fan-coil units distributed inside the drywell. Each fan-coil unit consists of two cooling coils and two direct connected motor-driven vaneaxial fans. Each cooling coil is connected to a cooling water supply and return piping system inside the drywell. One or both cooling coils may be utilized for temperature control. Each unit recirculates the drywell atmosphere through the cooling coils to control the drywell space temperature. Cooling water is supplied from the reactor building closed cooling water system.
>
> "Fan coil units circulate cooled air around the recirculating pumps and motors, the control rod drive area and the annular space between the reactor pressure vessel and the biological shield. The personnel access and control rod drive removal openings are sealed to ensure positive flow of cool air from the control rod drive area into the annular space between the reactor vessel and the biological shield through pipe openings in the reactor vessel support located primarily at the upper level of the control rod drive space.
>
> "Cooled air will also be circulated through the reactor vessel head area, the space immediately below the refueling seal plate and the relief valve area.
>
> "Each fan-coil unit has provisions for installing dust filters. Filters are to be employed only during construction and will be removed prior to normal station operation.
>
> "Each fan is started from a local panel by using run-off-auto type switches. One fan is started by switching to run and the other fan switch is placed in the auto position. If the normal operating fan fails, a flow switch will sense a reduced pressure and automatically start the standby fan and light an amber light at a local panel and annunciate in the control room. Cooling unit discharge air temperature is sensed by a temperature element and indicated in the control room. All fan-coil units can be operated from the emergency power supplies.

Local ambient drywell temperature measurements are available from fifteen RTD's installed in various azimuthal locations and ranging from the +12 foot to the +90 foot elevations. These temperature elements were installed to support the required Integrated Leakrate Tests. 5050-series TE outputs at present go to the process computer and temperature recorders.

I-2 Event Description

Normal operating temperatures prior to the 1974 outage at the upper drywell elevations (90') remained reasonably steady with peak temperatures approximately 180°F, and temperatures at the 38' elevation typically near 120°. During and subsequent to the 1974 outage, several deficiencies in maintenance-related actions lead to a gradual increase in temperature at the 38 foot level (as read by TE-9044) and a sharp increase in drywell temperature. By the end of cycle five, temperatures reached approximately 160°F at the 38 foot elevation and exceeded 240° at upper elevations. These deficiencies, and the direct consequences attributable to them in retrospect, are:

a. Removal of the undervessel access closure in 1974 (which reduced cooling airflow to the 81 foot elevation through the annular space between reactor vessel and biological shield).

b. Damage sustained by some flexible connections in ductwork during reinstallation, primarily in the 1977 and 1978 outages (leading to degraded ventilation airflow).

c. Incomplete reinstallation of ductwork removed to support outage maintenance, primarily in 1977 and 1978 outages (leading to degraded airflow due to distorted and missing ducts).

d. Plant operation with roughing filters installed (observed at the start of the 1981 outage). (This reduced the airflow through the cooler.)

e. Cancellation of filter unit cleaning in 1976, 1977, and 1980 (leading to reduced airflow and heat transfer capability due to air-side fouling of the coolers).

f. Damaged insulation, primarily from the 1977 and 1980 outages (which increased the ambient heat load in the drywell).

g. Mussel fouling of the Salt Service Water System during cycle 5 (effectively reducing the heat sink).

The above deficiencies may all be classified as inadequate control of a non-safety system (drywell ventilation) during maintenance of the drywell ventilation system itself as well as equipment in proximity to ventilation system components.

I-3 Temperature Environment

A conservatively estimated drywell temperature history was developed from the limited information available. This information consists of:

a) A record of temperatures at approximately 38 feet for the entire plant operating history from TE-9044. Gaps in this record are minimal.

b) A record of daily temperatures at ten different drywell locations from March 15, 1981 through September 24, 1981. the specific locations monitored were:

TE	Elevation	Azimuth
TE-5050-A	86'	000^{o}
TE-5050-D	90'	330^{o}
TE-5050-E	60'	270^{o}
TE-5050-F	60'	090^{o}
TE-5050-G	40'	270^{o}
TE-5050-H	40'	090^{o}
TE-5050-J	35'	000^{o}
TE-5050-K	35'	180^{o}
TE-5050-L	22'	205^{o}
TE-5050-M	22'	045^{o}

A time-temperature profile at the 38 foot elevation was developed for the history of the plant, based upon readings from TE-9044. This profile is presented in Figure I-1 as a series of plateaus where the plateau value is the maximum record temperature for the particular interval. The numbers above the plateaus indicate the respective interval durations. The plateau approach was selected because it simplifies the subsequent degradation analyses while ensuring that any error injected through this approximation method is conservative.

Drywell temperatures for other elevations were estimated by determining the worst-case temperature difference with respect to 38 foot readings for the ten 5050-series elements listed above. "Worst-case" for elevations above 38 was the largest observed difference; below 38 feet, the smallest difference was used.

The resulting temperature differentials as shown in Figure I-2 were applied to the "plateaued" history of temperature at 38 feet to obtain a conservative estimate of temperature at other elevations for the entire operating history of the plant.

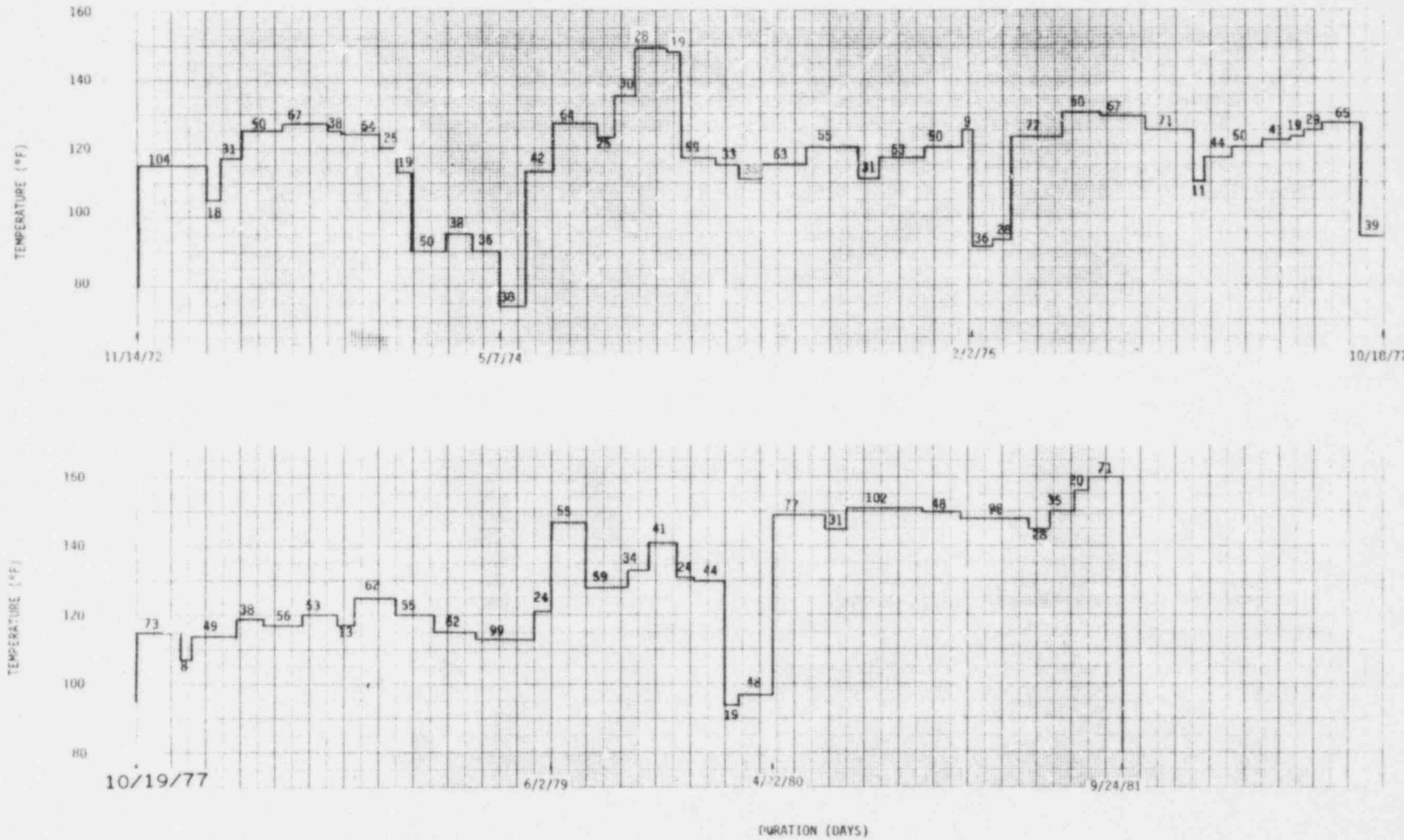
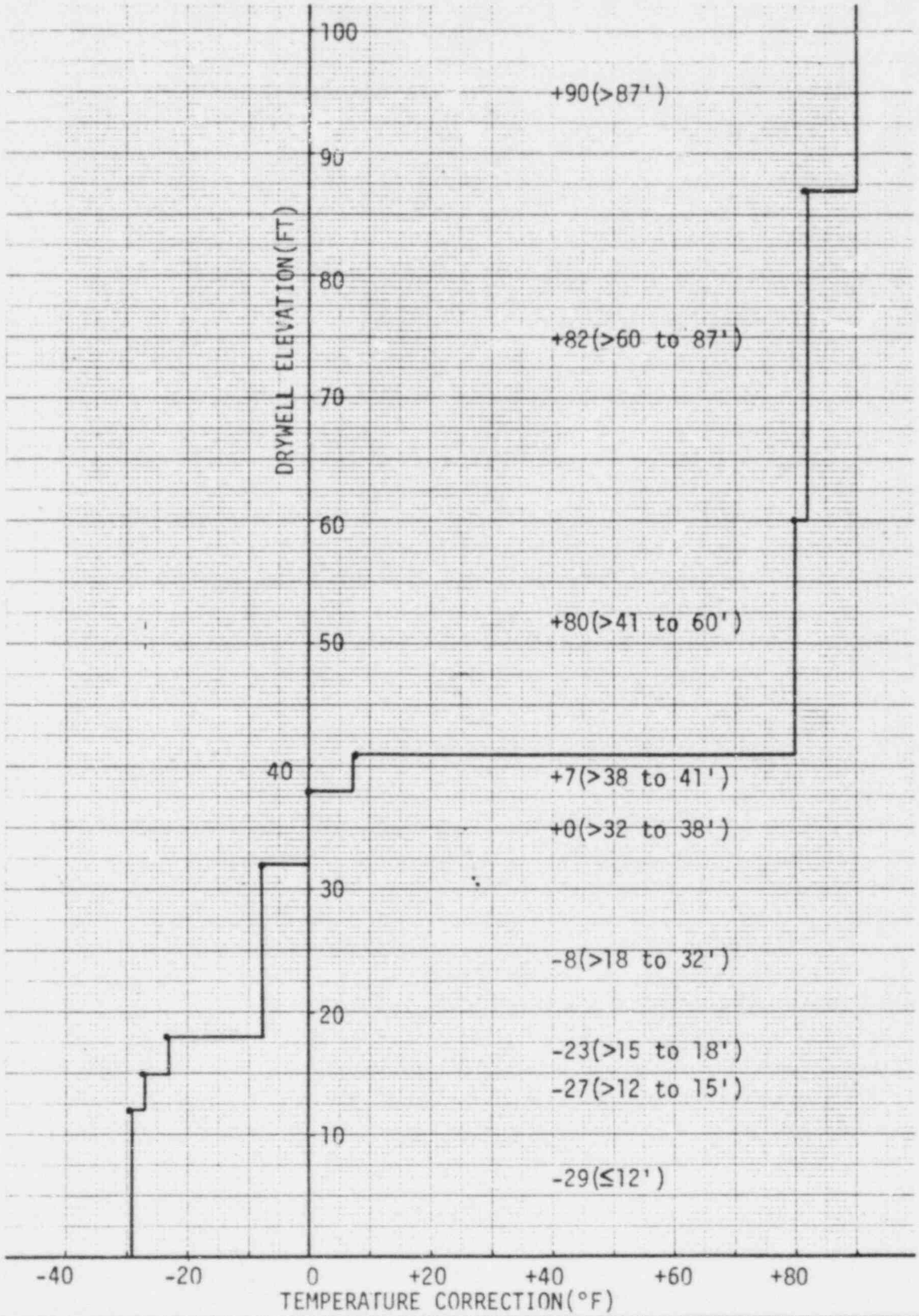


FIGURE I-1

PILGRIM NUCLEAR POWER STATION, UNIT 1
TEMPERATURE PROFILE (38' ELEVATION)

FIGURE I-2

DRYWELL TEMPERATURE CORRECTION CHART



I-4 Actions Taken to Reduce Ambient Drywell Temperature

Prior to the 1981 outage, Boston Edison scheduled several tasks associated with improving the performance of the drywell cooling equipment and monitoring system performance to identify potential areas system upgrades. These tasks included:

a. Replacement of all drywell cooling coils

b. Repair/replacement of existing ventilation ducting

c. Ventilation fan inspection and test

d. Drywell ventilation system balancing

e. Mechanical cleaning of the Salt Service Water System piping

f. RBCCW Heat Exchanger baffle plate modification

g. Installation of continuous chlorination system (1 year EPA approved test for mussel control)

h. Drywell temperature instrumentation refurbishment

i. Insulation inspection and subsequent repair

j. Installation of additional instrumentation to monitor the performance of drywell coolers and RBCCW Heat Exchangers

k. Start up monitoring program to monitor drywell cooler and RBCCW performance

As a result of cable and component inspections performed early in the outage, the following tasks were initiated:

a) Detailed component analyses to identify the extent of potential temperature damage to constituent non-metallic materials

b) Analysis/test/replacement of selected cables

c) Installation of undervessel access closures

Additional tasks to be performed prior to plant operation include:

a. Submittal of proposed tech spec limits for drywell temperature

b. Operational testing of components subjected to elevated temperatures

Section II - Consequences of the Drywell Temperature Event

II-1 Impact of High Drywell Temperatures on Reactor Pressure Vessel (RPV) Water Level Instrumentation

II-1.1 Introduction

Errors on the RPV Water Level detection system will arise as a result of operation at off-calibration conditions or under certain design basis accident conditions.

At Pilgrim Station, the RPV water level instrumentation is provided with both heated (YARWAY) and cold (GEMAC) reference leg designs. The instruments connected to these reference legs are typically calibrated to indicate the correct reactor vessel water level with the drywell at its original design operating temperature. At Pilgrim Station that temperature was assumed to be 135°F in the vicinity of the reference legs. Increases in drywell temperature would cause the reference leg of the instruments to heat up. As the reference leg water temperature increases, the reference leg water density will decrease. This will cause the sensed differential pressure to decrease even if the RPV water level and temperature remain unchanged. This process would introduce error in the water level readings, such the indicated RPV water level is higher than the actual RPV level.

A detailed discussion of the high drywell temperature effect (under accident conditions) on RPV water level instrumentation and specific recommendations were identified by GE in Service Information Letter (SIL) No. 299, dated July 25, 1979. GE's SIL No. 299 findings and recommendations are:

Findings

- A conservative maximum error of 12.7% of the heated (Yarway) reference leg's total length is predicted for drywell temperatures of 340°F (14 inches conservative maximum error for Pilgrim Station Yarway legs).

- Cold reference leg instruments are subjected to the same type of error, the magnitude of the error depending on the difference in vertical drop of the reference leg and the variable leg inside the drywell.

- Even with the potential RPV level instrumentation inaccuracies, acceptable ECCS performance is expected.

Recommendations

- Provide the Operators with specific guidance regarding the effects of high drywell temperature on the various reactor vessel level measurement systems.

Recommendations (continued)

- Modify control room level indicator scale readings to make the Operator consciously aware of the drywell temperature effect on level instrumentation and to avoid the possible misleading inference that the water level is stable when actual water levels are "off scale".

- Revise the RPV water level Safeguard trip setpoints (for Pilgrim Station Yarway Safeguard Instruments, consider raising trip setpoints by 14 inches).

II-1.2 Impact of GE SIL No. 299 on Pilgrim Station

Boston Edison reviewed the contents of GE SIL No. 299 in October and November 1979 and concluded that:

a) RPV water level instrumentation was experiencing an off-calibration error due to higher than normal drywell temperatures in the vicinity of the reference legs.

b) An engineering evaluation was needed to:

- determine the magnitude of the existing error;

- calculate new calibration conditions;

- Determine the necessity to raise ECCS trip actuation based on the GE recommendations.

The actions taken during start-up from the 1980 refueling outage included:

- Installation of thermocouples in the Yarway heated reference legs;

- Determination of drywell temperature in the area of the reference legs;

- Calculation of the probable off-calibration RPV water level error;

- Calculation and recommendation to recalibrate all RPV water level instruments (GEMAC and Yarway) based on the determination of drywell temperature.

- Recommendation not to raise the existing ECCS trip set points based on the concerns of SIL No. 299 related to unnecessary ECCS initiations during normal Plant transients.

- Revise the existing Operating Procedures

As a result of these actions the Pilgrim Station RPV water level instrumentation was found to be inaccurate by approximately three to four inches of indicated level (in the nonconservative direction). To correct this situation, the RPV water level instrumentation was recalibrated by 5 inches of indicated level at the end of July 1980, conincident with initiation of operating cycle #5. This recalibration was referenced to a measured drywell temperature of 205°F in the vicinity of the reference legs, but ultimately based on a conservative value of 215°F.

The bases for the recalibration was that it would ensure ECCS actuation, would result in more accurate level indications, and would not affect the perceived level response to the operators.

On September 22, 1980, the above actions were reported to the NRC via LEF 80-0 32/01X-0, "Drywell Temperature".

II-1.3 Safety Implementations of the PNPS 1 RPV Water Level Actions Taken During 1980.

By performing an instrumentation recalibration at the beginning of Cycle-5 (July 1980), the station assured that the Reactor Water Level Instrumentation Safeguard Level trips and indications were conservative with respect to the drywell operating temperature at the location of the reference legs (205°F). This was accomplished by recalibrating all level instrumentation to an assumed slightly higher drywell temperature (215°F) than that measured. Furthermore, this action was deemed appropriate to resolve the potential issues raised by GE - SIL 299, without requiring additional ECCS trip set point changes because:

(1) By recalibrating the safeguard instrumentation (Yarway) equivalent to 5 inches of indicated level (in the conservative direction), the worst steady-state water level error of 14 inches (G.E. SIL-299) developed as a result of an increase in drywell temperature from 125°F to 340°F will effectively be reduced to a potential 9 inches (14 inches minus 5 inches).

(2) Actual level indication changes due to increasing drywell temperature under accident conditions will occur rather slowly since the normal time constant of the Yarway reference leg is calculated by G.E. to be twenty to thirty minutes. This time constant is the characteristic response of the reference leg due to a sudden (step) increase in drywell temperature from 135°F to 340°F. A step increase in temperature of this magnitude is very conservative.

(3) G.E.'s analysis in SIL-299 showed that, even with potential vessel level instrumentation inaccuracies, acceptable ECCS performance would occur.

(4) Although the Technical Specifications low-low level setpoint is -49 inches indicated, the nominal trip setpoint at Pilgrim Station is set at -47 inches (2 inches conservatism).

(5) For a steady-state worst error of nine (9) inches in indicated level (drywell at 340°F) as a result of the most limiting small break LOCA, enough differential pressure will be available to reasonably assure ECCS low level actuation.

(6) The most limiting small break LOCA analyzed for PNPS 1 (.05 ft^2) shows that ECCS low-low level is reached in a time period considerably less than the time constant of the Yarway reference legs. The level measurement error at the time of ADS actuation is therefore considerably less than the bounding error of 9 inches discussed in item 5 above, thus further assuring ADS actuation.

(7) Raising the ECCS trip setting increases the risk of inadvertently actuating the ECCS systems during non-accident events. If mishandled, this could lead to adverse consequences rather than mitigate them. It could also increase the number of fatigue cycles due to cold RCIC and HPCI water being injected into the still warm feedwater system.

(8) Operating procedures were revised per SIL-299 recommendation to caution the operator of level instrument inaccuracy for large increases in drywell temperature. For this situation, the accident procedures caution the operators as follows:

> During rapid reactor depressurization and particularly less than 500 psig, the operator should utilize the cold reference leg type of level indicators (GEMAC) to give backup information on vessel water level. The operator should not turn off any ECCS unless there is sufficient confirming information from cold reference leg level instruments that water level has been restored. The operator should not rely on the "Yarways" if erratic behavior indicative of reference leg flashing has occured until the Yarway readings are on scale and in reasonable agreement with cold reference leg level instruments. the operator should verify that automatic ECCS actuations occur when the levels are at the trip points. The operator should be prepared to manually actuate ECCS during a suspected LOCA if automatic actuation is not achieved.

II-1.4 High Drywell Temperature Effects on RPV Instrumentation at the End of Operating Cycle #5.

Throughout Cycle 5 no deviations from normal operations were experienced with respect to RPV level instrumentation until the plant was being shut down for refueling on September 26, 1981. On that date, the Yarway reference legs experienced flashing which led to successive RPV high water level isolations followed by RPV low levelk scrams. This event was due to an existing RPV pressure essentially at atmospheric coupled with drywell temperatures of 240°F at the elevation of the reference legs.

The RPV water level instrumentation was at that time experiencing an off-calibration error, caused by a 25^{o}F higher drywell temperature than the one corresponding to the latest 1980 recalibration (215^{o}F).

This off-calibration situation accounts for approximately 1.7 inches error in indicated RPV level (the actual level being lower than indicated). With the normal low-low level setpoints at -47 inches, ECCS actuation would still have occurred within the Technical Specifications value of -49 inches.

It is concluded that the effect of high drywell temperature on reactor vessel level instrumentation did not compromise nuclear safety during cycle 5. The high drywell temperature event of September 26, 1981 posed no threat to the public health of safety. It did result, however in some off-normal level indications (oscillations on control room instruments. But, station procedures and operator training were sufficient to handle the event in a routine and correct manner.

In order to quantify the impace that a lower than required ECCS water level initiation has on existing G.E. applicable safety analysis, a bounding error value of 10 inches was assumed and used by General Electric in Section II-2.

II-2 The Impact of the Drywell Event on Safety Analyses

This section is based upon verified calculations performed by General Electric, to be issued as part of a report to Boston Edison at a future date.

II-2.1 Containment Response

In order to evaluate the effect of higher initial drywell temperature, the design basis loss-of-coolant accident, which results in the most severe drywell pressurization rate and peak pressure loading, was analyzed. Containment pressure/temperature response results for several higher initial drywell temperatures were compared with those for the standard 135°F initial drywell temperature assumption. As expected, the results indicate less severe response with higher initial temperature.

Sensitivity studies were done for 135, 160, 180, and 200°F initial drywell temperature values. The comparison results, presented below, show lower peak pressures and lower drywell pressurization rate for higher initial drywell temperature, which is expected because of reduced air density at higher temperature. There was negligible effect on peak temperatures. This comparison demonstrates that existing Mark I LDR Loads (based on 135°F) bound the loading at higher initial drywell temperature.

Initial Drywell Temp (°F)	135	160	180	200
Peak Drywell Pressure (psia)	55.77	55.52	55.30	55.07
Peak Wetwell Pressure (psia)	36.16	34.96	33.85	32.56
Drywell Pressurization rate (psi/sec)	68.5	63.6	60.0	58.8

II-2.2 Pool Swell Loads

Higher initial drywell temperatures will affect pressurization rate and enthalpy flow properties during discharge from the drywell to the Torus after a DBA. Using regression analysis results obtained from the Mark I Containment Program 1/4 Scale Test Program, GE has shown that pool swell loads (both downforce and upforce) will be lower for higher initial drywell temperatures.

II-2.3 Peak Cladding Temperature (PCT) for LOCA

The effect of lower than technical specification ECCS water level initiations water level instrument error on the peak cladding temperature (PCT) for loss of coolant accidents (LOCAs) was determined using the approved 10CFR50 Appendix K models. An error of 10 inches was assumed as a bounding case. The actual error at PNPS was less than this value. (Previously discussed in section II-1.3).

Reactor Scram and emergency core cooling system actuation instrumentation use vessel low water level (LWL) and/or high drywell pressure (HDP) signals for system initiation. LOCA's give rise to one or both of these conditions due to escaping vessel fluid inventory. Generally larger (0.2 ft^2 break flow area) pipe breaks cause a decreasing vessel pressure after scram, and an early HDP. The automatic depressurization system (ADS), which requires both LWL and HDP to acutate, is not needed, since the vessel fully depressurizes by the high break flow.

The limiting LOCA in terms of PCT is the Design Basis Accident (DBA), which is a complete double-ended recirculation suction line break. During this event, most of the vessel fluid inventory is lost from the break, due to the large break flow area, causing a very quick reactor depressurization. The effect of reduced initial water level is to cause the reactor water level to reach the level of the jet pump suction faster, resulting in a quicker core uncovery. During the time period from reactor scram to jet pump uncovery, the fuel rods are in a state of nucleate boiling, and hence the decreased initial water level means less nucleate boiling time. Nucleate boiling is associated with high heat transfer coefficients so that the length of time of nucleate boiling directly affects the PCT.

Based on Pilgrim vessel geometry and DBA break size, it is found that a 10 inch reduction in initial vessel water level results in a 0.2 second earlier jet pump uncovery time. Using earlier sensitivity studies, it can be determined that the 0.2 second reduction in jet pump uncovery time results in a 5^{o} increase in PCT in Pilgrim. This increase is deemed insignificant. Since the HDP signal occurs earlier than the LWL signal for a DBA, the reduced vessel water level has no effect on scram or ECCS initiation, since both systems require either HDP or LWL signals for initiation.

However, during a small break accident (less than approximately 0.2 ft^2 break flow area), the ADS system is required to actuate if the high pressure coolant injection (HPCI) system is assumed to be failed. This is because the vessel pressure tends to remain high due to steam generation in the core by decay heat and the break area provides insufficient flow to depressurize the reactor vessel. Also the escaping vessel inventory causes a HDP signal and the absence of high pressure coolant injection leads to a LWL signal.

By analyzing various break sizes, the most limiting small break in terms of PCT was found to be a 0.05 ft^2 recirculation suction line break, with an assumed failure of the HPCI system.

Next, this most limiting case was reanalyzed taking into account the assumed 10 inch instrument error. This had the effect of delaying the ADS actuation by 10.9 seconds, and produced a $44^{o}F$ increase in PCT, though the PCT was well below the 2200^{o} limit. Therefore, there is no significant impact of the water level instrumentation error on PCTs for LOCAs.

II-2.4 Abnormal Transient Event

Among abnormal transients, the instrument error will impact the Feedwater Controller Failure (FWCF) most because the water level indicator plays the key role in mitigating this event. The impact can be two ways:

- FWCF High Water Level Trip (maximum demand)

- FWCF Low Water Level Trip (minimum demand)

Other transient events, such as Load Rejection, Turbine Trip, MSIV Flux/Pressure Scram, etc., will not be impacted significantly because in these events the reactor will scram within less than one second (independent of the water level instrumentation) and the events are terminated shortly after.

Therefore, in order to evaluate the effect of water level instrument error, only the FWCF transient was investigated, and the evaluation results show no impact of the 10 inch assumed instrument error.

II-2.4.1 FWCF High Water Level Trip (maximum demand)

Since the instrument error results in a water level reading higher than the actual value, the reactor high water level trip would occur at water level lower than the standard level. Therefore, the FWCF transient with instrument error will be less severe than the standard transient with zero error.

Thus, the instrument error has no adverse impact on the FWCF high level trip case.

II-2.4.2 FWCF Low Water Level Trip (minimum demand)

For the FWCF low water level trip event, the instrument error would cause the reactor to scram later than normal.

Again a 10 inch bounding error was assumed. This error during a loss of feedwater flow will delay the reactor scram time by about 5 seconds. With reference to the FSAR Figure 14.0-12 it is seen that after losing all feedwater sources, all parameters, such as neutron flux, vessel pressure, surface heat flux..., show a continuous downward trend until the low water level scrams the reactor at 7 seconds into the transient. Adding an extra 5 seconds due to instrument error will allow the these parameters to decrease even more before the low level scram occurs.

During normal operation, i.e., with no instrument error, about 46 seconds after the feedwater controller failure, complete drive motor trip, main steam isolation valve closure and HPCI and RCIC initiation will occur when the water level drops to the low-low level set point. At this time, greater than 4.5 feet of water still remains above the active core. With a 10 inch instrument error, the active core would still be covered by more than 3.5 ft of coolant. Once HPCI/RCIC initiate, rapid water level restoration will occur. Therefore, the instrument error has no significant impact on the FWCF low water level trip event.

II-2.5 Conclusions

GE has concluded from the considerations delineated above that the elevated drywell temperatures experienced by the Pilgrim Station had negligible effect on the safety analyses.

II-3 Drywell Structure

The drywell structure is a cylindrical bulb type of containment vessel whose outer skin takes the form of approximately a 6'-0" thick reinforced concrete shell. There is a 2" air space separating the inside of concrete from the steel containment liner. The drywell containment liner was designed and installed in accordance with the ASME Boiler and Pressure Vessel Code Section III, Subsection b, Article 12. The welded plates comprising the liner vary in thickness from 13/16" thick (3" course) to 1-7/16" thick (8" course). The plates are carbon steel conforming to ASME SA 516, Grade 70 Fire Box quality made to SA 300. The code allowable tensile stress to 650°F is 17.5 ksi. Significant reduction in critical properties - yield strength, tensile strength and modulus of elasticity for high grade pressure plate steels used as drywell containment liners does not occur until a temperature of 800°F is reached. Therefore, the effect of the recorded high drywell temperature (244°F) is not a matter of concern with respect to structural integrity of the drywell during plant operation.

The liner vessel is not mechanically attached to the outer concrete shell. The lower bulb of the liner vessel (bottom 7'-0") is seated into the containment building concrete mat. Stability is additionally gained from the reactor vessel pedestal within the liner. The pedestal is composed of a reinforced concrete base mat poured within the steel liner bulb along with a cylindrical reinforced concrete pedestal (4'-0" thick) anchored to the base mat.

The functions of the drywell structure-outer concrete shell and inner steel containment liner are designed for the following considerations: Containment of normal operating pressures and temperatures (internal & external); containment of accident pressure and temperatures (internal and external); containment of radiation; jet forces-steam and/or water @ 300°F; gravity loads; wind loads-including tornado wind and missiles; earthquake loads; flooding of containment vessel loads; stress concentration from piping penetrations and vent thrust loads - suppression chamber.

Evaluations have been performed by Stone & Webster Engineering Corporation using published test data for residual concrete strength after 5 and 10 cycles of heating to 200°C (392°F) and 23°C (73°F) respectively. The indications at this time during this 1981-192 outage are that the concrete nearest to the drywell liner probably has a residual strength well above the specified concrete design strength of 4000 psi; interior concrete can be assumed to be closer to or possibly greater than the original strength of the concrete when it was placed due to its exposure to lower temperatures.

Extrapolation of published data for a thermal cycle in the drywell from 117.7°C to 23°C (244°F to 73°F) for the first 6 cycles, and an additional 21 cycles ranging between 93.3°C (200°F) and 23°C (73°F) estimates the residual strength of concrete closest to the liner 90.3% and 73.5%, respectively, of the original strength. Statistical records of the concrete as placed indicate an average strength of approximately 5000 psi. This indicates that the residual strength of the concrete closest to the liner is estimated to average approximately 4500 psi at this time in the life of the plant, and approximately 3700 psi after 21 additional cycles at the end of the plant life; also, the concrete strength increases as the distance from the liner increases.

These residual strength valves (of 4500 psi and 3700 psi) are considered for the purposes of this analysis to be nearly equivalent to the original design strength of 400 psi. The original calculations used an allowable stress of 0.75 of the original design strength or 3000 psi. Although the residual strength may differ somewhat from the original design strength, this is considered to be conservative when compared to the computed actual stresses of less than 2000 psi. (FSAR amendment 20).

II-4 Equipment Operability

The evaluation of the effects of elevated temperature upon drywell components began with a definition of those components requiring examination. The 79-01B equipment list was selected as a basis from which to work. To this list were added those items which, in the opinion of the Nuclear Operations and Nuclear Engineering Departments, were needed for safe and economical plant operation. Those safety-related items which were not subjected to excessive temperatures and the testable check bypass solenoid valves (SV 1001-95 A, B; SV 1400-41 A, B) were removed from the list. Finally, a page by page review of the technical specifications was performed to ensure that the effects of elevated temperature on systems required to be available/operable by technical specifications were considered. No additional list entries were required as a result of this review.

The net result was the list that was considered.

Next, the Nuclear Engineering Department defined those drywell components required to operate for plant shutdown, accident mitigation, and transient response. A detailed analysis revealed the following:

a) Safety functions of drywell components required for mitigation of a LOCA or small break inside containment were not jeopardized by the drywell event.

b) Safety functions of drywell components required for safe shutdown were not jeopardized by the drywell event.

c) The safety functions of drywell components required to mitigate abnormal transients, accidents, or events outside the drywell were not jeopardized by the drywell event.

Section III - Equipment Analysis

III-1 Technical Approach to Equipment Evaluation

The technical approach to the evaluation of possible detrimental effects of the event on equipment contained within the drywell is illustrated by the block diagram presented in Figure III-1. Initial steps in the evaluation process addressed three areas:

1) The drywell environment prior to and during the event--particularly temperature time histories for the life of the plant.

2) Identification of equipment located within the drywell with specific classification according to:

    a) Safety-related electrical equipment

    b) Non-safety but essential electrical equipment

    c) Mechanical equipment

3) Development of an evaluation methodology for the assessment of potential damage to the equipment due to the occurrence of the event.

Subsequent steps in the evaluation process addressed the actual assessment of potential damage and the determination of equipment status from the standpoint of both qualification integrity (for safety-related equipment) and reliability to perform its intended function. The flow of information which facilitated this evaluation, as illustrated in Figure III-1, identifies other inputs which were addressed during the course of performing the evaluation.

III-1.1 Methodology

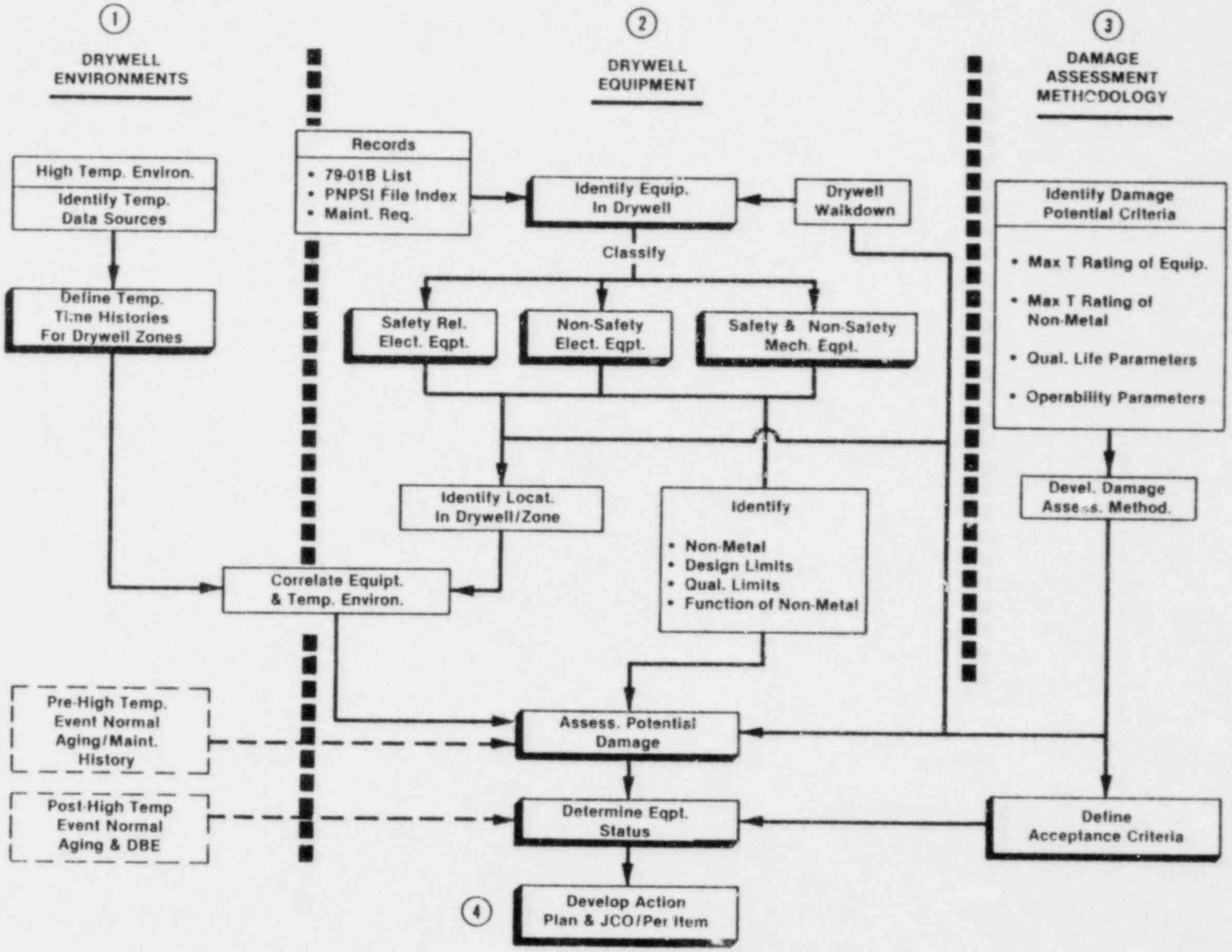
Key elements of the evaluation for all drywell equipment in general are discussed in the following paragraphs. The numbers preceding the paragraph titles correspond to annotation of the major elements of the Action Plan flow-chart presented in Figure III-1. Evaluation criteria unique to specific classes of drywell equipment, i.e., safety-related electrical equipment, non-safety electrical equipment essential to plant operation, and mechanical equipment are discussed in Subsections below which follow.

1 <u>Definition of Drywell Temperature Environment</u>

A temperature-time profile for the drywell was developed for the entire history of plant operation based upon temperature measurements recorded at the 38-foot elevation (Temperature element TE 9044). The drywell temperatures at other elevations were estimated by 1) determining the variation in temperature with elevation in the drywell for selective times and 2) applying the appropriate differential temperature correction factors to the baseline time-temperature profile corresponding to the 38-foot elevation. A correlation was then made between the equipment at various locations in the drywell, and the corresponding temperature to arrive at the temperature of exposure for individual items of equipment. A detailed discussion of the drywell temperature environment and the procedure followed in its development was previously discussed in Section I-3.

FIGURE III-1

ACTION PLAN
FOR ASSESSMENT OF DRYWELL
EVENT DETRIMENTAL EFFECT



2 Identification of Equipment in the Drywell

Two basic sources of data provided identification of equipment in the drywell.

a) File Records

1) 79-01B Safety-Related Equipment List

2) PNPS-1 Equipment File Index

3) Maintenance Reports

4) P & ID's

b) Drywell Walkdown Verification

A drywell walkdown was performed and information was recorded for:

i) Tag No. Verification

ii) Manufacturer Verification

iii) Model Number and Serial Number

iv) Elevation in Drywell

v) Visual check of the equipment with observation comments made for equipment discoloration, physical damage, leakage, melted components, etc.

3 Development of Damage Assessment Methodology

The Damage Assessment Methodology consisted of defining several criteria for equipment evaluation. Although there were some variations in the evaluation criteria from one class of equipment to the other (safety-related electrical equipment, non-safety electrical equipment essential to plant operation, and mechanial equipment) the evaluation was essentially made on the basis of the following data:

1) A drywell walkdown was made to verify equipment in the drywell and to make a visual inspection to determine any obvious degradation of the equipment.

2) Manufacturer's rated temperatures for the equipment were identified.

3) Materials comprising the equipment were identified and the rated temperatures of the materials, particularly nonmetallic components and lubricants, were determined.

4) Material degradation (aging) analyses were performed to determine materials susceptible to thermal aging.

5) Potential damage mechanisms and failure modes, other than just material degradation, were identified through discussions with equipment manufacturers.

6) Manufacturer's recommended maintenance of the equipment were identified.

These data, both individually and collectively, led to the specification of acceptance criteria for three classes of equipment contained in the drywell:

1) Safety-related electrical equipment

2) Non-safety electrical equipment essential to plant operation

3) Mechanical equipment.

4 Drywell Equipment Summaries

The evaluation of possible detrimental effects of the drywell event led to the definition of equipment status. An action plan was developed to resolve any deficiencies, and on the basis of the resolution, justifications for continued operation were established.

Section III-2 contains the completed summaries.

III-1.2 Acceptance Criteria

III-1.2.1 Safety-Related Electrical Equipment

The activities which were performed to evaluate possible detrimental effects of the event on safety-related electrical equipment are summarized as follows:

1) A drywell walkdown was performed for:

    a) Field verification of equipment tag number, manufacturer, model number and serial number as well as location within the drywell and,

    b) Visual inspection of the equipment to identify any evidence of physical damage, discoloration, leakage, etc.

2) Functional tests of the equipment have been (or will be) performed to demonstrate operability prior to plant operation.

3) Manufacturer's rated temperature for the equipment as well as materials and their rated temperature for continuous exposure were idenfitied to determine if design and end-use temperature limits were exceeded by the event.

4) Degradation analysis of nonmetallic materials in the equipment were performed using the Arrhenius methodology in order to determine material degradation resulting from aging for the temperatures of exposure prior to and during the event.

5) Potential damage mechanisms and failure modes, other than just material degradation, were identified through discussions with equipment manufacturers.

6) A thorough review of equipment maintenance and replacement records was made.

The following acceptance criteria were established for determining equipment serviceability and qualification status as well as further actions which should be taken to resolve potential qualification deficiencies arising from the event.

Acceptance Criteria

Equipment integrity was considered to be within acceptable limits for either 1) the period of interim operation identified for the equipment, or 2) the remaining life of the plant, as applicable, if the evaluation of possible detrimental effects due to the event met the following case-by-case criteria.

CASE I - EQUIPMENT DESIGN LIMITS NOT EXCEEDED

1) Visual inspection of the equipment indicated no evidence of physical damage to the equipment.

2) Functional tests are required to be performed prior to plant operation to demonstrate current operability of the equipment

3) The manufacturer's rated temperature for continuous exposure was not exceeded by the event.

4) An equipment material evaluation was conducted and no known materials susceptible to excessive degradation because of aging prior to or during the event were used.

5) Maintenance/replacement records were reviewed to ascertain that appropriate maintenance/replacement activities have been performed which may support the justification for continued use.

CASE II - MATERIALS DESIGN LIMITS NOT EXCEEDED

1) Visual inspection of the equipment indicated no evidence of physical damage to the equipment.

2) Functional tests are required to be performed prior to plant operation to demonstrate current operability of the equipment.

3) An equipment materials evaluation was conducted and it was established that the maximum rated temperature for continuous exposure of the materials was not exceeded and no known materials susceptible to excessive degradation because of aging prior to or during the event were used.

4) Other potential detrimental effects of the event have been resolved through failure modes and effects analysis.

5) Maintenance/replacement records were reviewed to ascertain that appropriate maintenance/replacement activities have been performed which may support the justification for continued use.

CASE III - MATERIAL DEGRADATION WITHIN ACCEPTABLE LIMITS

1) Visual inspection of the equipment indicated no evidence of physical damage to the equipment.

2) Functional tests are required to be performed prior to plant operation to demonstrate current operability of the equipment.

3) An equipment materials evaluation was conducted and the degradation of nonmetallic materials in the equipment for the drywell environment prior to and during the event was determined, by the Arrhenius technique, to be

below the qualification limits established for the equipment and that adequate qualification margin presently exists to account for additional aging degradation for the period of continued operation as well as an end-of-life design basis event (DBE).

4) Other potential detrimental effects have been resolved through failure modes and effects analyses.

5) Maintenance/replacement records were reviewed to ascertain that the maintenance/replacement activities have been performed as required in order to maintain qualification integrity.

Damage assessment reports have been prepared for the safety-related equipment located in the drywell. These individual reports are on file at the BECo offices. The damage assessment reports provided information necessary to determine possible detrimental effects of the drywell event on safety-related equipment. For cases where the foregoing acceptance criteria were met, the damage assessment reports provided the basis for Justification for Continued Operation (JCO). For cases where the equipment failed to meet the acceptance criteria, an action plan was developed to resolve the deficiencies. Although the specific action taken is unique to individual items of equipment, generally, the action consisted of one of the following:

1) Replacement of the affected equipment with new equipment of like kind or with new equipment which meets IEEE Standard 323-1974 requirements. In some cases, equipment of like kind was adequate, whereas, in other cases, the equipment qualification status was upgraded.

2) Replacement of the degraded materials with like kind in order to restore the equipment to its original qualification status, or using replacement materials which upgrade the qualification status of the equipment, in both cases taking into consideration any other potential detrimental effects of the event.

3) Removal of the equipment from the drywell for test evaluation to demonstrate that the integrity of the equipment is adequate for continued use for the life and design basis events (DBE's) demonstrated by the test. For this action, the equipment removed for testing would be the item in a generic group in the drywell most adversely affected by the event. This item would be replaced by new equipment of like kind or equipment with upgraded qualification status. A successful test evaluation on removed equipment would thus establish qualification for remaining generic equipment in the drywell.

Section III-2 presents the results of the evaluation of possible detrimental effects of the drywell event for each item of safety-related equipment, as well as the resolution of deficiencies arising from the event and justifications for continued operation.

III-1.2.2 Non-Safety Electrical Equipment Essential to Plant Operation

The activities which were performed to evaluate possible detrimental effects of the event on non-safety electrical equipment was basically the same as that employed for safety-related electrical equipment discussed in Section III-1.2.1. The primary differences in the evaluations between the two classes of equipment dealt with specifications of acceptance criteria. For safety-related electrical equipment, qualification integrity to meet IE Bulletin 79-01B requirements was a primary acceptance criterion, whereas for non-safety electrical equipment, the same stringent damage assessment methodology was employed; however, it was not necessary to demonstrate qualification integrity for an end-of-life design basis event. On the other hand, the acceptance criteria fully addressed possible detrimental effects of the drywell event and, where these effects were determined to be significant, the appropriate actions were taken to resolve deficiencies and to ensure the serviceability of the equipment for continued plant operation.

Acceptance Criteria

Equipment integrity was considered to be within acceptable limits for either 1) the period of interim operation, or 2) the remaining life of the plant, as applicable, if the evaluation of possible detrimental effects met the following case-by-case criteria:

CASE I - EQUIPMENT DESIGN LIMITS NOT EXCEEDED

(Same criteria used for Safety-Related Electrical Equipment, Section III-1.2.1)

CASE II MATERIALS DESIGN LIMITS NOT EXCEEDED

(Same criteria used for Safety-Related Electrical Equipment, Section III-1.2.1)

Case III - MATERIAL DEGRADATION WITHIN ACCEPTABLE LIMITS

(Same criteria used for Safety-Related Electrical Equipment, except for Item 3. For non-safety electrical equipment, adequate qualification margin for an end-of life design basis event was not considered essential for continued operation.)

Damage assessment reports have been prepared for non-safety electrical equipment essential to plant operation located in the drywell. These individual reports are on file at the BECo offices. The damage assessment reports provided information necessary to determine possible detrimental effects of the drywell event on non-safety electrical equipment. For cases where the foregoing acceptance criteria were met, the damage assessment reports provide the basis for justification for continued operation. For cases where the equipment failed to meet the acceptance criteria, an action plan was developed to resolve the deficiencies. The actions taken for non-safety electrical equipment placed emphasis on detailed inspections and checkout, using the damage assessment reports as a guide to determine what materials may have been degraded by the drywell event. For cases where inspections/checkout revealed equipment degradation, one of the following actions was taken:

1) Replacement of the affected equipment with new equipment. Where possible, equipment qualified as safety-related electrical equipment was used to replace non-safety related electrical equipment in order to achieve added reliability and increased service life.

2) Replacement of the degraded materials with either like kind or upgraded materials, in both cases taking into consideration any other detrimental effects of the event.

Section III-2 presents the results of the evaluation of possible detrimental effects of the drywell event for each item of non-safety electrical equipment, as well as the resolution of deficiencies arising from the event and the justifications for continued operation.

III-1.2.3 Mechanical Equipment

The evaluation of possible detrimental effects of the drywell event on mechanical equipment followed two parallel paths:

1) Visual inspection and, where necessary, tests were or will be performed to assess unacceptable deterioration of equipment components, and

2) Material evaluations were performed to determine, by analytical means, the possible degradation of material properties.

Mechanical equipment considered in this evaluation were snubbers, operators for air-operated valves, fans and blowers, and ventilation ductwork and coolers. The component parts of this equipment directly vulnerable to thermal degradation are elastomeric seals, diaphragms, and flexible connections, lubricants, and hydraulic fluids. Seals and flexible connections in the ductwork were inspected to determine if these may have been subjected to mechanical abuse, as well as being exposed to debilitating temperature. In general, the evaluation of mechanical equipment placed emphasis on inspections and tests as a means of determining physical damage, with analytical assessments being used as a backup to ensure that components that may have been significantly degraded were not overlooked.

Acceptance Criteria

Acceptance criteria for mechanical equipment located in the drywell was based largely on inspections and tests which were (or will be) performed on the equipment. Because of the diverse nature of the mechanical equipment in the drywell, the method of assessing potential damage, as well as the acceptance criterial used, varies for each type. The following procedures were generally followed:

Snubbers

1) All snubbers were visually inspected for indication of leakage, physical distortion, discoloration, or obvious physical damage. None of these indications were observed.

2) Sixteen snubbers distributed from the +11 to +52 foot elevation were tested in the snubber test rig. All snubbers passed all tests including bleed rate, lockup, leakage, and piston movement. One snubber from elevation +90 remains to be tested.

3) Seals were verified to be comprised of ethylene propylene rubber (EPR) and the hydraulic fluid was verified to be SF 1154.

Valve Air Operators

1) A total of five air operators may have been affected by the drywell high temperature event (see detailed list in section III-2).

2) Air operators will be disassembled, inspected and rebuilt starting with the operator at the highest elevation (A0-220-52) and proceeding through operators at lower elevations until air operators are found that have no evidence of non-metallic materials degradation.

3) Diaphragms will be inspected for damage such as cuts, tears, cracks, or spalls and will be replaced as necessary.

4) All air regulators will be replaced on listed air operators (section III-2).

5) A materials evaluation was performed by identifying the materials in the air operators and, on the basis of rated temperatures and rated properties, making a determination of possible materials degradation. Where age-sensitive materials were identified, corrective action will be taken to resolve the apparent deficiencies.

Ventilation System Fans, Blowers, Ductwork, and Coolers

1) Bearings in the three fan motors at the highest elevation (45 ft) have been examined to verify that seals are intact and that no gross leakage of lubricant has taken place. No deficiencies were noted by this inspection.

2) The ductwork was checked and corrections made for any dents that may restrict flow and broken or loose seams, seals, and flexible connections. Doors and panels were repaired and replaced as necessary.

3) All cooling coils were replaced, thus returning them to their original design cooling capacity.

III-2 Results of Equipment Evaluation

III-2.1 Safety-Related Electrical Equipment

Safety related electrical equipment located in the PNPS-1 Drywell is identified in Boston Edison's 79-01B submittal. The evaluation of possible detrimental effects of the event considered each item of equipment on this list. The evaluation results are presented in individual damage assessment reports on file at Boston Edison. The information presented in this section is an item-by-item description of the results of these evaluations and identifies specific actions which either have been performed or which are planned for completion prior to continued operation of PNPS. The format of the presentation for each item consists of:

a. Identification of the Equipment, including:

   1) Manufacturer

   2) Type of device

   3) Model number

   4) Equipment tag numbers (same as equipment numbers presented in the 79-01B equipment list)

   5) Installation date

   6) Maximum temperature of exposure during the Drywell Event

   7) Manufacturer rated temperature for continuous exposure

b. Summary of Equipment Evaluation - This is a summary of the damage assessment reports mentioned above.

c. Resolution of Deficiencies - This is a statement of specific action which must be completed to resolve deficiencies arising from the event.

d. Justification for Continued Operation - This is a statement that possible detrimental effects have been addressed, based upon sound engineering analyses, and where they have been found to exist, an action plan has been implemented to resolve the attendant deficiencies.

e. References - References which support the justification for continued operation are cited and generally include:

   1) Wyle's Evaluation Report on possible detrimental effects of the drywell event.

   2) BECo's response to the NRC Safety Evaluation Report on 79-01B deficiencies.

1) ASCO SOLENOID VALVES MODEL NO. NP8320A184E

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
SV-220-44	5-8-80*	70'	242°F	250°F(deenergized)

*PNPS-1 Maintenance Request Log No. 080-246

Summary of Equipment Evaluation[1]

ASCO rates NP Series valves at 140°F while energized. The subject PNPS-1 valve is energized only when sampling (normally deenergized). The manufacturer has indicated that these valves are rated at 250°F when deenergized, based on the limit for the EPR seal and disc material. The difference is to allow for the temperature rise of the coil when energized and the loss of power due to increased coil resistance on DC valves at elevated temperatures. Therefore, the manufacturer's deenergized rating was not exceeded for this valve. Additionally, NP Series valves have been tested successfully following thermal aging at 268°F for 288 hours while energized. The temperature rise is 144°F, which gives an aging temperature equivalent to 268 + 144 = 412°F for a deenergized valve. This results in a qualified life of 3800 years for EPR, based on 50% loss of initial elongation and a temperature of 190°F for a deenergized valve. From this analysis, continued use of the ASCO solenoid valve is justified.

Resolution of Deficiencies

The evaluation indicated that the drywell event has no deterimental effects on either the serviceability or the qualification integrity of this equipment.

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1 Superscript denotes reference number. References are presented at the end of the discussion on each item equipment.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments;

1. The equipment was evaluated for possible detrimental effects of the drywell event with the result that:

    a) Equipment design limits were not exceeded

    b) Materials design limits were not exceeded

    c) Material degradation was within acceptable limits such that the remaining qualified life is considerably greater than the manufacturer's recommended maintenance interval.

    Having met the above acceptance criteria, it is concluded that the drywell event had no detrimental effects on either the serviceability or the qualification integrity of this equipment.

2. Qualification deficiencies identified in the NRC's Safety Evaluation Report are addressed in BECO's response[2] to the NRC. It suffices to note here that this equipment meets the requirements of the DOR Guidelines.

3. This equipment will be subjected to operational checkout in accordance with BECO's startup management program.

References

1. Evaluation of Possible Detrimental Effects of the Drywell Event on ASCO Solenoid Valves, NP8320A184, Wyle Report No. 17536-11 December 1981.

2. Boston Edison Company, Response to NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293, January, 1982.

3. Final Report on the Evaluation of the Qualification of Three-Way Solenoid Valve, ASCO Model NP8320A184E for use in Boston Edison's Pilgrim Nuclear Power Station Unit 1, Wyle Report No. 17446-11, October 1980.

4. ASCO Qualification Report No. AQS-21678/TR, March 1978.

2) AVCO SOLENOID VALVES, MODEL NO. C5159

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
AO-203-1A,B,C,D	7-77*	27	152°F	167°F (energized)

*PDCR No. 77-39.

Summary of Equipment Evaluation[1]

The subject valves are used to open the main steam isolation valve and are normally energized during operating (Reference Instruction Manual, Atwood-Morrill MSIV, Section 1, page 2, G.E., Purchase Order No. 205H1504, BECo/G.E. file I.D. M1-R). The manufacturer's rating for continuously energized valves has not been exceeded; however, the valve package does contain epoxy which encapsulates the three solenoid coils. An aging analysis performed on the epoxy indicates that its expected life based on 50% loss of flexural strength is 24 years at its maximum rated temperature of 320°F (160°C) which includes heat rise in the coil. These solenoid operated valve assemblies were installed in 1977 per PNPS-1 PDCR No. 77-39. Because the expected life of epoxy is greater than three times its installed life, it is concluded from the evaluation that the solenoid coils suffered no detrimental effects due to the drywell temperature of 152°F (67°C).

All other non-metallic components in these valves except the Parker lubricant were analyzed and found to be insensitive to thermal aging effects at 152°F (67°C).

Because the maximum drywell temperature experienced did not exceed Parker's rating of 180°F (82°C) for the O-Lube lubricant, it is concluded that this lubricant suffered no adverse effects.

Contact with the valve manufacturer, as referenced in the Wyle report, indicates a maintenance interval of 12 to 18 months. The manufacturer suggests replacement of all valve component parts marked with a small triangle on the assembly drawings. The overall assembly drawing is AVCO No. 5-5140-4H, dated 9-30-75; C-5140-8H, dated 10-3-75; C5577 dated 9-27-74 and; C-3988-15, dated 9-19-74. The suggested component part replacement is quite extensive and appears to be very conservative. Material degradation analysis indicates that continued use of these valves is justified, without component replacement.

Resolution of Deficiencies

The evaluation indicated that the drywell event had no detrimental effects on either the serviceability or the qualification integrity of this equipment. Furthermore, the materials aging analysis supports continued use of this equipment.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments;

1) The equipment was evaluated for possible detrimental effects of the drywell event with the result that:

   a) Equipment design limits were not exceeded

   b) Material design limits were not exceeded

   c) Material degradation is within acceptable limits such that the remaining qualified life justifies continued use of this equipment

   Having met the above acceptance criteria, it is concluded the the drywell event had no detrimental effects on either the serviceability or the qualification integrity of this equipment.

2) These valves are functionally tested as part of BECo's ongoing surveillance program and closed properly when tested during this outage.

3) Qualification deficiencies identified in the NRC's Safety Evaluation Report are addressed in BECo's response[2] to the NRC.

4) This equipment will be subjected to operational check-out in accordance with BECo's startup management program.

References

1) Evaluation of Possible Detrimental Effects of the Drywell Event on AVCO Solenoid Valves, C5159, Wyle Report No. 17536-3A, dated December, 1981.

2) Boston Edison Company, Response to NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Environmental docket No. 50-293, dated January, 1982.

3) NAMCO LIMIT SWITCHES, MODEL NO. EA740-50100

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
AO-220-44	Assumed to be 1972	70'	242°F	194°F (90°C)
AO-203-1A,B,C,&D	5-16-80*	27'	152°F	194°F (90°C)

*PNPS-1 Maintenance Request Log No. 080-17, 18, 19 & 20.

Summary of Equipment Evaluation[1]

The manufacturer's rating has not been exceeded on switches AO-203-1A, B, C and D. However, these switches contained Buna-N, which is a known age-sensitive material, that could have experienced degradation due to the drywell event. Also, it should be noted that the manufacturer's maintenance instructions require "Scheduled Maintenance" after the first 1-1/2 years of operation and at 4-1/2 to 5-year intervals thereafter (reference NAMCO Maintenance Procedures No. EA749-20010). These switches have been installed for approximately 18 months and maintenance is due in order to maintain qualification integrity. This requirement is independent of any effects of the event.

The manufacturer's rating has been exceeded on switch AO-220-44. Based on the analysis contained in Wyle Report No. 17536-3B, the limiting nonmetallic materials are Buna-N and EPR, which compose the gaskets and seals. Both of these materials have a design limit of 250°F (121°C) which were approached for switch AO-220-44. Also aging analyses indicated that both of these materials are subject to thermal degradation for the 242° temperature to which this switch may have been exposed.

Performance of the manufacturer's 18-month Scheduled Maintenance, which calls for replacing the Buna-N gaskets and EPR seals with new silicone rubber materials (NAMCO Maintenance Procedure No. EA749-20010), will resolve the deficiencies for this equipment, as well as extend the maintenance interval to 4-1/2 to 5 years.

Resolution of Deficiencies

The evaluation indicated that the drywell event had possible detrimental effects on the limit switches in the drywell due to the age sensitivity of Buna-N gaskets and EPR seals. One of the following alternative actions will be taken to resolve the deficiencies prior to resuming plant operation:

1. Perform the manufacturer's recommended 18-month schedule maintenance, using the recommended replacement kit with silicone rubber gaskets and seals, or

2. Replace all limit switches with either like kind or, preferably, with switches of upgraded qualification status

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1. Possible detrimental effects of the drywell event on the equipment have been thoroughly evaluated. Based upon a materials degradation analysis, age-sensitive materials have been identified and corrective action shall be taken to resolve the deficiencies prior to resuming plant operation. The listed NAMCO switches will be rebuilt.

   It is concluded that, once this corrective action is performed, the serviceability and qualification integrity of this equipment will be ensured.

2. This equipment is functionally tested as part of BECo's ongoing surveillance program.

3. Qualification deficiencies identified in the NRC's Safety Evaluation Report are addressed in BECo's response[2] to the NRC. It suffices to note here that this equipment satisfies the requirements of the DOR Guidelines. Corrective actions identified in 1) above will reestablish the qualification integrity of this equipment.

4. This equipment will be subjected to operational checkout in accordance with BECo's startup management program.

References

1. The Evaluation of Possible Detrimental Effects of the Drywell Event on NAMCO Limit Switches, EA740-50100, Wyle Report No. 17536-3B, dated December 1980.

2. Boston Edison Company, Response to NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293, dated January, 1982.

3. Final Report on the Evaluation of the Qualification of NAMCO EA740 MSIV Limit Switches, Containment Isolation Valve Control System, Boston Edison Pilgrim Nuclear Power Station, Unit 1, Wyle Report No. 17446-3B, October 1980.

4. Qualification of NAMCO Controls Limit Switch Model EA740 to IEEE Standards 344-1975, 323-1974, and 382-1972, Rev. 1, dated February 22, 1979, ACME Cleveland Development Company (no report numbers indicated).

4) TARGET ROCK SOLENOID VALVES, MODEL NO. 1/2SMS-A-01

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
SV203-3A,B,C&D	5/13/80*	47'	240°F	350°F

*PDCR 80-04

Summary of Equipment Evaluation[1]

The manufacturer's rating was not exceeded by the drywell event. These valves are mounted to the actuators of the relief valves. Conductive heating of the relief valve assembly occurs due to the hard coupling of the relief valve to the main steamline. During certification testing of the relief valve, it is preheated to 500 to 540°F. Conceivably, the solenoid valve could have experienced excessive temperatures if the surrounding air is such that normal convective and radiactive cooling does not occur. This potential problem has been discussed with the manufacturer. It is their recommendation that the solenoid valves be reworked or replaced. It should be noted also that the manufacturer requires replacement of solenoid internals and silicone o-rings every six years, per Target Rock Technical Manual No. 7567F-000, dated October 1980. Due to potential detrimental effects of the drywell event, this maintenance should be performed during this outage.

Resolution of Deficiencies

The evaluation indicated possible detrimental effects of the event on this equipment. The manufacturer's recommended six-year maintenance will be performed during this outage, and it will consist of replacing the valve internals including nonmetallic materials.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1. The equipment was evaluated for possible detrimental effects of the drywell event and, even though the temperature rating of the equipment was not exceeded, possible detrimental effects could have been experienced due to the effects of the drywell high temperature on the natural cooling of the valves. These solenoid valves will be replaced prior to start-up.

2. This equipment is functionally tested as part of BECo's ongoing surveillance program.

3. Qualification deficiencies identified in the NRC's Safety Evaluation Report are addressed in BECo's response[2] to the NRC.

4. This equipment will be subjected to operational checkout in accordance with BECo's startup management program.

References

1. Evaluation of Possible Detrimental Effects of the Drywell Event on Target Rock Solenoid Valves, 1/2SMS-A-01, Wyle Report No. 17536-13, dated December 1981.

2. Boston Edison Company, Resonse to NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293 dated January 1982.

5) LIMITORQUE VALVE ACTUATORS MODEL SMB-(Various)

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
MO1001-63	Assumed 1972	84'	242°F	150°F
MO1201-2	" "	48'	240°F	150°F
MO1301-16*	March 1981	41'	167°F	150°F
MO2301-4*	Assumed 1972	40'	167°F	150°F
MO1001-50	" "	50'	240°F	150°F
MO202-5A, B	1980	23'	152°F	150°F
MO261-1* (MO220-1)	Assumed 1972	23'	152°F	150°F

*Limitorque motor operators with tag numbers M01301-16, MO2301-4 and MO261-1 (MO220-1) employ Peerless electric motors, whereas the others employ Reliance electric motors.

Summary of Equipment Evaluation[1]

No data was found to indicate that the Peerless motor has been environmentally tested on the Limitorque valve operator. Limitorque has indicated that the Peerless motor insulation system is similar to that of the motor used in the operator which was environmentally tested (Test Report Number 600376A). Since this test was conducted with a Reliance motor, the Peerless motor design was not tested. Qualification requires evaluation of both materials and design under accident conditions, therefore, this test does not demonstrate full qualification of the Peerless motor. The Class H insulation system is rated at 356°F (180°C) during operation (including coil temperature rise). At the 167°F ambient temperature of M01301-16 and MO2301-4 (with Peerless motors) this allows for 356 -167 = 189°F (105°C) temperature rise of the coil before exceeding the motor rating. This would be an excessive temperature rise for a Class H system operated within design load and is therefore unlikely to have occurred. Similarly for MO 261-1 (MO220-1), which is equipped with a Peerless motor, at 152°F ambient temperature, the allowed temperature rise of the coil is 356°F-152°F = 204°F (113°C) without exceeding the motor rating. Based on this analysis continued use of the Peerless motor (actuator is addressed later) is justified; however, qualification of the motor is incomplete pending resolution.

Actuators MO1001-63, MO1001-5O, and MO1201-2 have Reliance Class H motors and have possibly been exposed to a temperature of 240°F to 242°F. This allows for 356 - 242 = 114°F (63° C T) temperature rise of the coil before exceeding the motor rating. This temperature rise is not out of line for a continuous duty Class H motor. Since these motor operators are operated relatively infrequently, however, rather than continuously, continued use of these motors is justified. Actuators MO202-5A and MO202-5B have the Reliance Class H, type RH qualified motors which were installed approximately two years ago. At 160°F the expected life is 1.12×10^5 years and the demonstrated life remaining is 42.0 years based on Arrehenius type evaluation of actual motorette failure test data. The Reliance Class H, type RH qualified motor has been successfully tested on a Limitorque actuator. Additional assurance of motor integrity is provided by the materials evaluation for the Reliance Class H, type RH insulation motor. At 242°F the expected life is 2036 years.

The lubricants used in Limitorque operators are rated above the maximum temperature experienced. The drive lubricant, Exxon Nebula EP-1, is rated at 300°F by the manufacturer. Limitorque is presently using EP-O in their containment service actuators which is rated at 250°F. Exxon considered Limitorque's catalog rating of 150°F as conservative for the Nebula EPO and EP1. The geared limit switch grease is Beacon 325 which has a rating of 250°F. Limitorque is still using the Beacon 325 grease with Mobil 28 grease as an acceptable substitute. The Mobil 28 grease is rated at 350°F. The Nebula EP1 and the Beacon 325 lubricants have been qualified by testing at Limitorque. Nebula EP-1 and Beacon 325 lubricants are used at PNPS. It is concluded that the drywell event had no adverse affect upon lubricants.

The melamine or fibrite limit and torque switch materials have been qualified by Limitorque testing. The melamine is white in color and the fibrite is brown. Melamine has an expected life of 146 years at 242°F and an expected life of 1.32×10^6 years at 160°F. Fibrite expected life is 20.45 years at 242°F and is 1,265 years at 160°F. Both materials' lives are based on 50% loss of original flexural strength. It is recommended that the limit and torque switch materials be inspected to determine whether or not they are melamine (white) or fibrite (brown). If they are not either melamine or fibrite, Limitorque recommends replacement with fibrite components.

Raychem "Flamtrol" and Rockbestos "Firewall III" wire are qualified by Limitorque testing. Both wire insulations have minimum expected lives in excess of two years at 242°F. At 240°F, the expected lives are in excess of 2.5 years. Raychem "Flamtrol" exhibited expected lives of 95 years at 167°F, of 192 years at 152°F, and of 131 years at 160°F. The Rockbestos "Firewall III" exhibited expected lives of 263 years at 167°F., of 784 years at 152°F, and of 4[illegible] years at 160°F. It is recommended that Raychem "Flamtrol" and Rockbestos "Firewall III" wire be replaced M01001-63, M01001-50, and M01201-2.

Viton seals and anchorite gaskets are qualified by Limitorque testing. Limitorque uses both Buna-N and Viton seals. Buna-N has an expected life of 39 days at 152°F, based on 20% loss of initial elongation. Viton has an expected life of 46 years at 242°F and 3372 years at 160°F based on 60% compression set. Limitorque recommends Viton type seals for their in-containment actuators. Limitorque operators at PNPS have been verified to contain Viton seals. The anchorite gasket material serves as cover seals and is not related to qualification. Limitorque stated their actuators are designed to be independent of absolute sealing for survival under normal and accident conditions, for their containment units include "T" drains to permit them to "Breathe." This independence is evidenced by accidental submergence of an operator during testing (Limtorque Report No. 600376A) with no ill effects.

Resolution of Deficiencies

1) Continued use of the Peerless motor and Reliance Class H motor is justified from a standpoint of the effects of the drywell temperature event; however, a final decision on these motors will be addressed in the 79-01B submittal.

2) All listed MOVs will be inspected and repacked as required prior to outage completion. Lubricants have been verified to be Nebula EP1 and Beacon 325.

3) Limit Switches will be inspected to verify that only melamine or fibrite switches are installed prior to outage completion.

4) Motor leads on MO 1001-63, MO 1201-2, and MO 1001-50 will be inspected and replaced as required prior to outage completion. Jumper wires on these valves will be replaced.

5) Limitorque Corporation has verified that Buna-N seals have not been used in the fabrication of the listed MOV's.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1. Possible detrimental effects of the drywell event on the equipment have been thoroughly evaluated. Based upon a materials degradation analysis, age - sensitive materials have been identified and corrective action shall be taken to resolve the deficiencies prior to resuming plant operation. This corrective action shall include:

   a) Lubricant repacking as required

   b) Limit switch case verification

   c) Motor lead inspection and, if required, replacement

   d) Jumper wire replacement on MO 1001-50, MO 1001-63, MO 1201-2.

It is concluded that once the above corrective actions are performed, the serviceability of the equipment will be ensured

2. This equipment will be subject to operational checkout in accordance with BECo's startup management program.

3. Qualification deficiencies identified in the NRC's Safety Evaluation Report are addressed in BECo's response[2] to the NRC.

References

1. Evaluation of Possible Detrimental Effects of the Drywell Event on Limitorque Valve Actuators Model SMB-(Various), Wyle Report No. 17536-5, December, 1981.

2. Boston Edison Company, Response to NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293, dated January, 1982.

6) TEC VALVE FLOW MONITOR SYSTEM, MODEL NO. 1414

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
ZT203-1 through 6	1981 outage (except accelerometer)	47 ft	240°F (ambient) 525°F (accel.)	600°F

Summary of Equipment Evaluation[1]

This equipment is a safety/relief valve monitoring system which is designed to detect flow through the valves. Major components of this system are:

1. Accelerometer Sensor - BBN Model 424-1SO-TEC mounted to discharge piping
2. Charge Converter TEC Model 504 Transient Shield TEC Model 160 — In-Containment
3. Cable Assembly TEC Model 424-C2
4. Monitor Module and Alarm TEC Model 914 — Out-of-Containment

The 504 charge converter with transient shield and the cable assemblies are new installations which replace older equipment. The BBN accelerometers have been installed since the 1980 outage and will be retained in service.

The evaluation indicated that the drywell event had no detrimental effects on the accelerometer. Also, BBN certifies these accelerometers for maximum continuous operation at 600°F and they have been successfully aged at 125°C for 600 hours. One consideration addressed during the evaluation was the temperature of the piping to which the accelerometers are mounted. The temperature of the relief valve discharge piping has not been established; however, based on relief valve temperature, the piping could conceivably reach 500°F. BBN advised that 550°F would not present any problem to these sensors based on in-service experience. Consequently, it is concluded that the drywell event had no debilitating affects on the accelerometers and their continued use is justified.

Resolution of Deficiencies

The evaluation indicated that the drywell event had no detrimental effects on either the serviceability or the qualification integrity of the accelerometers. The charge converters with transient shields and the cable assemblies are new equipment which have been qualified for the drywell environment.

Justification for Continued Operation

Continued operation of the accelerometers is justified on the basis of the following arguments:

1. The accelerometers were evaluated for possible detrimental effects of the drywell event as well as potential heating due to the accelerometer mounting on the relief valve discharge piping with the result that:

   a) equipment design limits were not exceeded

   b) material design limits were not exceeded

   c) the equipment does not contain any known age-sensitive materials.

   It is concluded that neither the drywell event nor the piping temperature had detrimental effects on the equipment which would affect equipment serviceability or qualification integrity.

2. The charge converters with transient shield and the cable assemblies are new equipment which have been qualified for the drywell environment.

3. Qualification deficiencies identified in the NRC's Safety Evaluation Report are addressed in BECo's response[2] to the NRC. It suffices to note here that this equipment is qualified[3] to IEEE Standard 323-74 and meets the requirements of NUREG-0588-Category 1.

4. This equipment will be subjected to operational checkout in accordance with BECo's startup management program.

References

1. Evaluation of Possible Detrimental Effects of the Drywell Event on TEC Valve Flow Monitor System, 1414, Wyle Report No. 17536- December, 1981.

2. Boston Edison Company, Response to the NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293, January, 1982.

3. Qualification Test Report for Environmental and Seismic Testing of the TEC Valve Flow Monitor System, TEC REport No. 517-TR-03, December 1980; Addendum on Qualification Test of Charge Converter 540 with Transient Shield 160 in preparation.

7) JUNCTION BOXES (BUCHANAN TERMINAL BLOCKS, SERIES 525 and HD 222 LOCATED IN HOFFMAN NEMA 4 ENCLOSURES)

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
J 208 through 216	Assumed	Various	$250^{\circ}F$	$350^{\circ}F$ (Term. Block)
J 43,44,55,56	1972			$150^{\circ}F$ (Encl. Seal)

Summary of Equipment Evaluation[1]

The materials evaluation identified the terminals blocks as being composed of G.E. Genal 4000, filled, flame retardant phenolic with a temperature rating of $350^{\circ}F$ and the Hoffman enclosures as being fabricated of steel with neoprene seal having a temperature rating of $150^{\circ}F$. The temperature rating for the terminal blocks was not exceeded by the drywell event, whereas the rating for the neoprene seal was exceeded. Also, neoprene is a known age sensitive material which probably experienced debilitating effects due to the drywell event. However, loss of enclosure seal integrity would not have had a detrimental effect on the terminal blocks since the Genal 4000 phenolic may be considered as age insensitive. This material has an expected life of 170 years at the maximum drywell temperature of $250^{\circ}F$.

Enclosure seal integrity is necessary to maintain the qualification status of the terminal blocks since these blocks have not been qualified for a LOCA or MSLB outside of an enclosure. The enclosure seal integrity must be verified by inspection prior to resuming plant operation.

Resolution of Deficiencies

The evaluation indicated that the drywell event had possible detrimental effects on the qualification integrity of this equipment. This deficiency will be resolved prior to resuming plant operation by inspecting the junction boxes in the drywell (for safety-related electrical equipment) and replacing seals which have experienced degradation.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1. Possible detrimental effects of the drywell event on this equipment have been thoroughly evaluated. Based upon a materials evaluation, it was established that:

   a) The terminal blocks experienced no debilitating effects since the rated temperature was not exceeded and the phenolic composing the blocks is insensitive to aging for the range of temperatures to which it was exposed.

b) The enclosure seals may have experienced debilitating effects which require that these seals be inspected and replaced where degradation is evident. This corrective action will be taken prior to resuming plant operation.

It is concluded that once the above corrective actions are performed, the serviceability and qualification integrity of this equipment will be reestablished.

2. The qualification integrity of the junction box enclosures will be reestablished by the inspection/replacement action. Qualification deficiencies for this equipment as identified in the NRC's Safety Evaluation Report are addressed in BECo's response[2] to the NRC on 79-01B issues.

References

1. Evaluation of Possible Detrimental Effects of the Drywell Event on Junction Boxes (Buchanan Terminal Blocks 525 and HD 222 in Hoffman NEMA 4 Enclosures), Wyle Report No. 17536-4, December 1981.

2. Boston Edison Company, Response to the NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293, January, 1982.

8) VARIOUS TERMINATIONS, RING-TONGUE TYPE

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
Terminations (Less Than 4 kV)	Assumed 1972	All	250°F	Unknown

Summary of Equipment Evaluation[1]

Since no manufacturer or catalog numbers have been identified, a specific assessment of drywell high-temperature degradation effects cannot be performed. Therefore, some general comments concerning terminations and analysis of a common termination insulation material are provided. Generally speaking, if sound installation practices are used to attach terminations to a qualified barrier-type terminal block, then the lug shank insulation can be considered as not safety related; that is, failure of the insulation would not degrade the serviceability of the circuit. If, however, inadequate clearance are provided between the wires, the degradation of the lug shank insulation must be considered.

For drywell terminations, the following actions are presented.

1. Inspect representative samples of termination for visible evidence of deterioration or inadequate clearance between wires, excessive stripping, etc.

2. Correct all clearance problems. Remake terminations on all defective lugs.

Resolution of Deficiencies

The evaluation indicated that the drywell event should not have caused any detrimental effects on the terminations, provided good installation practices were followed when the terminations were originally installed. As a precautionary measure, representative terminations will be inspected to identify any deficiencies. Corrective action will be taken, as necessary.

Justification for Continued Operation

Justification for Continued operation is based upon the following:

1. Representative sampling should indicate the general quality of terminations. If all terminations inspected are satisfactory, BECo. is justified in assuming termination practices followed in the drywell are adequate, and no further justification is required.

2. In the event a significant number of poorly made terminations are discovered, then all termination of systems essential to plant operation will be inspected and re-worked as required. Completion of this action justifies continued use of drywell terminations.

References

1. Evaluation of Possible Detrimental Effects of the Drywell Event on Terminations (less than 4Kv), Ring-Tongue Type, Wyle Report No. 17536-12, December 1981.

9) KERITE 600 V POWER AND CONTROL CABLE, TYPE FR/FR

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
Kerite 112,212, 312,512,712,912, B7,B8	1972	Various	250°F	194°F (90°C)

Summary of Equipment Evaluation[1]

The Kerite Company performed electrical and physical tests on two separate cable samples which were removed from the drywell for this purpose. The sample cables were 9/C, #12 FR/FR control cable which have been in service for approximately 10 years. Sample #1 was removed from the 74-ft elevation in the drywell and it was exposed to a maximum in-service temperature of approximately 230°F. Sample #2 was removed from the 41-ft elevation and it was exposed to a maximum in-service temperature of approximately 165°F. The results of the Kerite evaluations are summarized as follows:

1. The initial examination revealed no visible evidence of damage and both samples were flexible and appeared to be in very good condition.

2. Electrical tests for insulation resistance revealed that the values for both samples were substantially higher than the guaranteed value for this cable, ranging from 7.4×10^4 to 2.1×10^5 Megohms for Sample #1 and form 1.7×10^4 to 4.2×10^4 Megohms for Sample #2. The samples, both as completed cables and the individual insulated conductors, passed the mandrel-bend voltage withstand test specified in IEEE 383-74. The minimum quickrise breakdown voltage was 21.5 kV, about 36 times the rated voltage.

3. Physical tests were performed to measure the percent elongation of the FR insulation (HI-70) and the FR jacket (HI-711). Percent elongation is the material property used to determine the cable degradation due to aging. The end-of-life criteria for elongation used by Kerite and recommended for use in determining when cable replacement is required is 50% absolute elongation for the FR insulation and 60% absolute elongation for the FR jacket. These values of elongation represent the condition of prototype (test) samples, after thermal and radiation aging, prior to exposure to a LOCA profile which enveloped the PNPS-1 DBE. The following table is the tabulation of elongation test results for the two samples (1 & 2). Also presented for comparison are elongation results for materials aged to 40 years at 90°C operating temperature (C), materials aged to 40 years at 90°C operating temperature, and then exposed to 200 megarads of gamma radiation (pre-LOCA) conditioning) (D), and the end-of-life due to physical cracking of the materials(E).

Conditioning	Elongation (%) HI-70 Insulation	HI-711 Jacket
(A) Sample 1 (160°F)	235	250
(B) Sample 2 (230°F)	110	160
(C) 40 Years	150	190
(D) 40 Years + 200 Megarads	50-75	60-100
(E) End-of-Life	less than 20	less than 20

4. Based on a life comparison analysis between the elongation of the PNPS-1 samples and the end-of-life criteria, Kerite estimates that 10% to 16% of Sample #1 and 50% to 55% of Sample #2 life has been expended, resulting in an estimated minimum of 18 years (100%-55%)x40 years remaining life at 194°F.

5. Based on their analysis, Kerite recommends that the remaining Kerite cables at PNPS-1 be continued in service and that cables exposed to similar conditions be tested at subsequent refuelings for quantifying degradation. When the value of elongation for a cable approaches the recommended end-of-life criteria, it should be replaced.

Resolution of Deficiencies

The Kerite test evaluation quantified the degradation experienced by this equipment for its total service history. On the basis of this evaluation, an estimated minimum of 18 years remaining life has been established. Per Kerite's recommendations, an ongoing program for periodic testing of cable samples will be instituted to quantify future degradation and to establish cable serviceability and qualification integrity.

Since the worst-case condition evaluated by Kerite corresponded to the 74-ft elevation, BECo has elected to replace cables serving safety system components required for accident mitigation above this level with new Anaconda cable which has been qualified for the drywell environment.

Justification of Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1. Possible detrimental effects of the drywell event have been thoroughly evaluated through inspection and tests to quantify the electrical and physical properties of test samples removed from the PNPS-1 drywell. On the basis of this evaluation, an estimated minimum of 18 years remaining life has been established based on the physical properties of the cable. Electrical tests confirmed the present serviceability of the cable. Kerite cable above the 74-ft elevation powering components whose operation is required for accident mitigation will be replaced with new Anaconda cable qualified for the drywell environment.

2. An ongoing program for periodic testing of Kerite cable samples will be incorporated into BECo's ongoing surveillance program.

3. Qualification deficiencies identified in the NRC's Safety Evaluation Report are addressed in BECo's response[2] to the NRC. It suffices to note here that this equipment meets the requirements of the DOR Guidelines.

4. This equipment will be subjected to operational checkout in accordance with BECo's startup management program.

References

1. Report to Boston Edison on Returned Pilgrim-1 Cables, The Kerite Company Letter, dated November 20, 1981.

2. Boston Edison Company, Response to the NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293, January, 1982.

3. Final Report on the Evaluation of the Qualification of Kerite Type FR/FR Power and Control Cable for Use in Boston Edison's Pilgrim Nuclear Power Station, Unit 1. Wyle Report No. 17446-2, October 1980.

4. Franklin Institute Research Laboratories, Report No. F-C4020-1, March 1975 (Kerite Proprietary).

10) OKONITE POWER AND CONTROL CABLE, OKONITE INSULATION, OKOPRENE JACKET

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
Okonite 112,212, 312,412,512,712, B8	1972	Various	$250^{o}F$	$194^{o}F$ ($90^{o}C$)

Summary of Equipment Evaluation[1]

The manufacturer's temperature rating was exceeded on all cable above the 41-ft elevation due to the drywell event. The Okonite electrical cable used in the PNPS-1 drywell is composed of Okonite insulation, which is EPR, with Okoprene flame retardant jacket, which is neoprene. A materials evaluation was performed which indicated that possible detrimental effects could occur for the Okoprene jacket at all elevations in the drywell. This, in fact, was verified at the higher elevations by inspection of the cable. The jacket material had separated and crumbled from cable at the higher elevations. The expected life of neoprene at 250^{o} is only 0.01 year and, in fact, less than one year for all elevations in the drywell. However, this is not considered to be consequential since 1) the drywell is inerted to 5% oxygen or less which will preclude any fire hazard and 2) the Okoprene is not required for electrical insulation. The EPR insulator (30 mils) has been qualified for $302^{o}F$ for 504 hours. The Okonite cable used in PNPS-1 has a minimum of 30 mils EPR insulation. Based on a degradation equivalency analysis, using the PNPS-1 drywell temperature-time profile, a demonstrated qualified life of 6.2 years at rated temperature ($194^{o}F$) remains for the insulation at or below the 41-ft elevation, based on the $90^{o}C$ cable rating. In actuality, the safety-related electrical equipments provide virtually no load to these cables (only the AVCO solenoid valves are normally energized). Based on a drywell maximum temperature of $160^{o}F$ for future operation, the remaining qualified life of these cables below the 41-ft elevation is 127 years.

Based on this evaluation, it is concluded that Okonite cables serving equipment which might be required for accident mitigation should be replaced above the 41-ft elevation with a more suitable cable. Okonite cable below the 41-ft elevation may be continued in service since a) the flame retardant jacket, though possibly degraded, is not required by virtue of the drywell inert atmosphere, and b) the EPR insulator has a remaining qualified life in excess of 120 years.

Resolution of Deficiencies

The Okonite cable serving equipment which might be required for accident mitigation, or which has been determined to be essential for plant operation is being replaced above the 41-ft elevation during the present outage with Anaconda cable which has been qualified to IEEE Standard 323-1974.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1. Possible detrimental effects of the drywell event on this equipment have been identified based upon a materials degradation analysis. It was determined that Okonite cable above the 41-ft elevation serving essential equipment should be replaced. Anaconda cable is presently being installed as a replacement. This cable has been qualified to IEEE 323-1974 for inside containment use.

2. The material degradation analysis indicated that the insulation for the Okonite cable located at or below the 41-ft elevation has a remaining qualified life in excess of 6 years at rated temperature and in excess of 120 years at 160°F ambient temperatures. The flame retardant jacket may have experienced degradation at these elevations; however, this is considered to be inconsequential due to the inert atmosphere of the drywell during plant operation. Also, the jacket material is not required as an insulator.

3. Qualification deficiencies for this equipment as identified in the NRC's Safety Evaluation Report are addressed in BECo's response[2] to the NRC.

4. The Anaconda cable used to replace selected Okonite above the 41-ft elevation has been qualified to IEEE Standard 323-74 as addressed in BECo's response[2] to the NRC's Safety Evaluation Report.

5. The new Anaconda cable will be subjected to electrical checkout prior to resuming plant operation in accordance with BECo's startup management program.

References

1. Evaluation of Possible Detrimental Effects for the Drywell Event on Okonite #12 600 V Power and Control Cable. Wyle Report No. 17536-1, December 1981.

2. Boston Edison Company, Response to the NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293, January 1982.

11) GENERAL ELECTRIC SWITCHBOARD WIRE, CAT. NO. S157275 (TYPE SIS)

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
S157275	1972	37 ft	160°F	194°F (90°C)

Summary of Equipment Evaluation[1]

The manufacturer's rating has not been exceeded on the wire; however, the crosslinked polyethylene insulation is checked age-sensitive in the 40 year column of the table in the DOR Guidelines. Further evaluation indicates that crosslinked polyethylene has a design limit of 194°F (90°C). Based on analysis at 160°F for 25% loss of initial elongation, the expected life is 82.1 years. Also, crosslinked polyethylene wire has been tested successfully following thermal aging at 268°F for 1,296 hours. This corresponds to a qualified life of 45.5 years at 160°F based on time to reach 25% loss of initial elongation. Remaining qualified life is 43.8 years at 160° F and 4.2 years at the rated temperature of 194°F. Based on this analysis, continued use of the SIS wire is justified.

Resolution of Deficiencies

The evaluation indicated that the drywell event had no detrimental effect on either the serviceability or the qualification integrity of this equipment.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1. The equipment was evaluated for possible detrimental effects of the drywell event with the result that:

   a) Equipment design limits were not exceeded

   b) Material design limits were not exceeded

   c) Material degradation due to aging for the period of installation prior to and during the event was found to be within acceptable limits. Based on 160°F and 25% loss of initial elongation, this equipment has an expected life in excess of 80 years and a qualified life, based on test demonstration, in excess of 40 years.

   Having met the preceding acceptance criteria, it is concluded that the drywell event had no detrimental effects on either the serviceability for the qualification integrity of this equipment.

2. Qualification deficiencies identified in the NRC's Safety Evaluation Report are addressed in BECo's response[2] to the NRC. It suffices to note here that this equipment has been qualified[3,4] to the requirements of the DOR Guidelines.

References

1. Evaluation of Possible Detrimental Effects of the Drywell Event on G.E. Switchboard Wire, Type SIS, Wyle Report No. 17536-8, dated December 1981.

2. Boston Edison Company, Response to the NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293, dated January 1982.

3. Final Report on the Evaluation of the Qualification of G.E. SI-57275 Type SIS Switchboard Wire for Use in Boston Edison's Pilgrim Nuclear Power Station, Unit 1, Wyle Report No. 17446-8, dated October 1980.

4. Qualification Tests of Electrical Cables Under Simulated Reactor Containment Service Conditions, Including LOCA, Franklin Institute Research Laboratories Report No. F-C4497-2, dated March 1977.

12) RAYCHEM CABLE SPLICES, MODEL WCSF-N

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
Splice (600 V	4/77	37 ft	160°F	196°F (191°C)
Penetration)	and Later			
Splice (SOV)		27-70 ft	242°F	196°F (191°C)

Summary of Equipment Evaluation[1]

The manufacturer's rating has not been exceeded on these splices for SOVs at the higher elevations. The material comprising these splices is polyolifin. Qualification tests have been performed on these splices which included accelerated aging at 302°F (150°C) for 1500 hours under full rated load (1000 volts, 25 amps). The remaining demonstrated qualified life at 160°F for the worst case (corresponding to SOV splices exposed to 242°F maximum temperature) is in excess of 200 years, and at 190°F, 32.3 years.

Resolution of Deficiencies

The evaluation indicated that the drywell event had no detrimental effects on either the serviceability or the qualification integrity of this equipment.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1. The equipment was evaluated for possible detrimental effects of the drywell event with the result that, although the manufacturer's rating was exceeded, only one material comprises this equipment and it was determined to be insensitive to thermal degradation for the range of temperatures to which it was exposed. Furthermore, the equipment has a remaining demonstrated qualified life in excess of 200 years at 160°F and in excess of 32 years at 190°F.

   Having met the preceding acceptance criteria, it was concluded that the drywell event had no detrimental effects on either the serviceability or the qualification integrity of this equipment.

2. Cables for which this equipment is used as splices have been subjected to an extensive inspection, test, and, for certain cases, replacement. This resulted in a comprehensive inspection of cable splices and, where cables were replaced, new splices were installed (PDCR 78-07).

3. Qualification deficiencies identified in the NRC's Safety Evaluation Report are addressed in BECo's response[2] to the NRC. It suffices to note here that this equipment meets the requirements of the DOR Guidelines.

References

1. Evaluation of Possible Detrimental Effects of the Drywell Event on Raychem Cable Splices, WCSF-N Wyle Report No. 17536-10, dated December 1981.

2. Boston Edison Company, Response to the NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293, dated January 1982.

3. Environmental Qualification Test Report of Raychem WCSF-N Nuclear In-Line Cable Splice Assemblies for Raychem Corporation, Menlo Park, California, Wyle Report No. 58442-1, May 1980

13) G.E. ELECTRICAL PENETRATIONS, CANNISTER TYPE

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
Q100 A-E	1972	37 ft	160F	150°F (Bechtel Spec.)
Q101 B	1972	37 ft	160°F	150°F (Bechtel Spec.)
Q102 A,B	1972	37 ft	160°F	150°F (Bechtel Spec.)
Q103 A,B	1972	37 ft	160°F	150°F (Bechtel Spec.)
Q104 A-H,J	1972	37 ft	160°F	150°F (Bechtel Spec.)
Q105 A,B	1972	37 ft	160°F	150°F (Bechtel Spec.)
Q106 B	1972	37 ft	160°F	150°F (Bechtel Spec.)

Summary of Equipment Evaluation[1]

The safety-related functions of these penetrations are identified as follows:

1. Drywell Pressure Boundary Integrity (Only)

Equipment Tag No.	Electrical Distribution
Q 100 A Q 100 B Q 100 C Q 100 D	Neutron Monitoring
Q 100 E Q 103 B	Thermocouple
Q 104 A Q 104 B Q 104 C Q 104 D Q 104 E Q 104 F Q 104 G Q 104 H	Control Rod Position Indication
Q 104 J	Control Rod Position Indication and TIP System

2. Drywell Pressure Boundary Integrity and Electrical Distribution of Safety Related Equipment

Equipment Tag No.	Electrical Distribution
Q 101 B Q 105 A Q 105 B	Low Voltage Power
Q 102 A Q 102 B Q 103 A Q 103 B	Low Voltage Control

It should be noted here that all electrical connectors originally installed on low voltage power and control penetrations have been removed and Raychem WCSF-N splices installed per BECo's PDCRs 79-04, 07, 09, 10, and 11.

It should be noted that the manufacturer's rating identified is stated on Summary of Proposal dated 5/8/69 submitted by G.E. with their quotation for the penetrations. This equipment was subjected to aging at 281°F/63 psig for 10 days. Similar equipment with the same epoxy primary seal material was exposed to the same aging conditions followed by a steam test at 320°F for 2 hours.

A materials evaluation to identify materials susceptible to aging has been performed. A number of nonmetallic materials have been identified; however, the end-uses of the materials have not been established in every case. It is anticipated that many of these materials compose the electrical connectors which have been removed from service. Materials which are considered to be critical to the penetrations internal pressure integrity and electrical integrity are the primary sealant which is an epoxy with an overlay of potting compound and the wire insulation. The epoxy sealant has a remaining qualified life of 72.5 years at 160° F base based on 50 percent loss of flexual strength. Only one type of wire has been identified for these penetrations; G.E. switchboard wire, type SIS with Vulkene SI-57275 insulation (cross-linked polyethylene) which has a remaining qualified life of 47.8 years at 160°F based on 25% loss of elongation.

In addition to the materials evaluation, in-situ pressure tests are recommended. Periodic pressure tests are recommended by the manufacturer to check seal integrity between the penetration cannister and the drywell nozzle. These tests should be performed on all electrical penetrations.

Resolution of Deficiencies

An action plan has been defined to further evaluate possible detrimental effects of the drywell event on this equipment and to establish equipment serviceability. This plan consists of performing pressure tests on the penetration in-situ to establish drywell pressure boundary integrity.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1. Tests have been performed during the 1981-82 outage to establish pressure integrity of this equipment. No deficiencies were noted.

2. Materials evaluations provide added assurance that the pressure and electrical integrity of the penetrations have not been compromised by the drywell event.

3. Qualification deficiencies identified[1] in the NRC Safety Evaluation Report are addressed in BECo's response[2] to the NRC.

4. BECo is a participant in the EPRI-UEQ BWR Owners' Group which has a Subgroup[3] pursuing 79-01B deficiencies for GE penetrations. The results of their effort will provide additional insight to the qualification integrity of this equipment.

5. This equipment is subject to BECo's ongoing surveillance program.

References

1. Evaluation of Possible Detrimental Effects of the Drywell Event on G.E. Electrical Penetrations, Cannister Type, Wyle Report No. 17536-9, dated December 1981.

2. Boston Edison Company, Response to the NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293, dated January 1982.

3. Action Plan for the Environmental Qualification of BWR Utilities Common Items of Class 1E Electrical Equipment, Prepared for UEQ-BWR Owners' Group, Wyle Report No. 17478-13, dated August 1981.

14) PHYSICAL SCIENCE ELECTRICAL PENETRATIONS

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
Q 101 A	1972	37 ft	160°F	Not specified
Q 101 C	Assumed	37 ft	160°F	Not specified

Summary of Equipment Evaluation[1]

These penetrations are used for high-voltage power and their safety-related function is drywell pressure boundary integrity only. A materials evaluation has been performed on this equipment to identify possible detrimental effects arising from the drywell event. It has been determined that only one non-metallic material is involved insofar as pressure integrity is concerned - Durock T-39 silica ceramic potting compound. This material is not susceptible to thermal degradation at the conditions to which it was exposed (160°F maximum temperature). This material has a maximum continuous operating temperature of 750°F to 800°F and its softening temperature is 1200°F.

However, in view of the fact that a periodic pressure check is a manufacturer's recommended maintenance/surveillance activity, these penetrations will be subjected to pressure tests to establish their pressure boundary integrity.

Resolution of Deficiencies

An action plan has been defined to further evaluate possible detrimental effects of the drywell event on this equipment and to establish equipment serviceability. This plan consists of performing in-situ pressure tests on the penetrations to establish drywell pressure boundary integrity.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1. Tests have been performed during the 1981-82 outage to establish the pressure integrity of this equipment. No deficiencies were noted.

2. A materials evaluation indicated that this equipment contains no age-sensitive materials which would compromise pressure boundary integrity.

3. Qualification deficiencies identified in the NRC Safety Evaluation Report are addressed in BECo's response[2] to the NRC.

4. This equipment is subject to BECo's ongoing surveillance program.

References

1. Evaluation of Possible Detrimental Effects of the Drywell Event on Physical Science Electrical Penetrations, 600 V Power, Wyle Report No. 17536-6, dated December 1981.

2. Boston Edison Company, Response to the NRC Safety Evaluation Report on the Environmental Qualification of Safety-Related Electrical Equipment, Docket No. 50-293, dated January 1982.

III-2.2 Non-Safety-Related Electrical Equipment Essential to Plant Operation

Non-safety-related electrical equipment located in the drywell which is considered to be essential to plant operation is identified in Table III-1. The evaluation of possible detrimental effects of the event addressed each item of equipment on this list. The results of this evaluation are presented in individual damage assessment reports on file at Boston Edison. The information presented in this section is an item-by-item description of the results of these evaluations and presents specific actions, where required, which either have been performed or which are planned for completion prior to continued operation of PNPS-1. The format of the presentation is similar to that employed for safety-related electrical equipment, with the exception that qualification deficiencies are not addressed per se. The presentation for each item consists of:

a. Identification of the Equipment

b. Summary of Equipment Evaluation

c. Resolution of Deficiencies

d. Justification for Continued Operation

e. References

TABLE III-1.

LISTING OF NON-SAFETY ELECTRICAL EQUIPMENT ESSENTIAL TO PLANT OPERATION

No.	Equipment Tag No.	Equipment Description System	
1	TE 5050-xxx	Omega Thermocouples	Drywell Cooling
2	TE 5050-A thru P	PYCO RTD's 22-3027-2	Drywell Atmosphere
3	TE 263-66 A1,A2	PYCO Thermocouples (magnetic head)	Reactor Vessel Temp.
	TE 263-66 B1,B2		Reactor Vessel Temp.
	TE 263-67 A1,A2		Reactor Vessel Temp.
4	TE 263-66-xx	PYCO Thermocouples (welded)	Reactor Vessel Temp.
	TE 263-67-xx		Reactor Vessel Temp.
	TE 263-69-xx		Reactor Vessel Temp.
5	TE 6271-A,B through TE 6276-A,B	Leeds & Northrup Thermocouples (welded)	Safety Valve 203-4A,4B,4C & 4D

1) DRYWELL UNIT COOLER DISCHARGE TEMPERATURE
(OMEGA MODEL BT-000-T-2¾-48-1 THERMOCOUPLES)

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
TE-5050-xxx	New* 1981 Outage	40 ft	N/A	$176^oF(80^oC)$ (Stepping switch)

* PDCR 81-57

Summary of Equipment Evaluation[1]

New temperature elements, designed as TE 5050-xxx, are being installed during the present outage to monitor the performance of the drywell cooling system and to display the temperature on an indicating panel located outside the drywell.

Thermocouples are installed on representative air and water inlet and outlet and water outlet of Air Coolers and on the air outlet from the motors of the reactor vessel recirculation pumps (P201A and B).

These instruments are being added to collect process temperatures of the drywell cooling system during the forthcoming operating cycle. These additional data would supplement data from existing temperature elements for the evaluation of the performance of the cooling system which is being repaired, cleaned and rebalanced during the present station outage.

Resolution of Deficiencies

This equipment has no known deficiencies for the application in the drywell.

Justification for Continued Operation

Use of this equipment for continued plant operation is justified on the basis that this equipment is new and the conditions under which it must function are well within the equipment design limits.

2) DRYWELL ATMOSPHERE TEMPERATURE (PYCO TEMPERATURE ELEMENTS (RTD), MODEL NO. 22-3027-3)

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
TE-5050 A-P	New* 1981 outage	Various	244°F	1100°F

* PDCR 81-41

Summary of Equipment Evaluation[1]

New temperature elements have been installed during the present outage for all TE 5050-A through P designated tag numbers which measure drywell atmosphere temperature. The Pyco Model 22-3027-3 is an air temperature RTD in a perforated tube with only two non-metallics, GE RTV 116 and fiberglass. For the range of temperatures to which this equipment will be exposed, both of these materials are insensitive to aging degradation. RTV 116, for example, has an expected life of 925 years at an ambient temperature of 250°F, based on 50% loss of initial elongation.

Resoluton of Deficiencies

This equipment has no known deficiencies for the appplication of the drywell.

Justification for Continued Operation

Use of this equipment for continued plant operation is justified on the basis that this equipment is new and the conditions under which it must function are well within the equipment design limits.

Reference

1. Evaluation of Possible Detrimental effects of the Drywell Event on Pyco Temperature Elements (RTD), Model No. 22-3027-3, Wyle Report No. 17536-24, dated December 1981.

3) REACTOR VESSEL SURFACE TEMPERATURES (PYCO TEMPERATURE ELEMENTS, MODEL NO. 13-2012-T-1200)

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
TE 263-66A1,A2	Assumed 1971	38-92 ft	250°F	400°F
TE 263-66B1,B2				
TE 263-67A1,A2				

Summary of Equipment Evaluation[1]

3.0

These PYCO temperature elements are magnetic thermocouple type temperature detectors which consist of a spring-loaded thermocouple mounted in the center of a ring-type magnet which firmly holds the measuring junction against the reactor pressure vessel. The vessel has a design pressure and temperature of 1250 psig at 575°F). This indicates that the maximum temperature exposure of the magnet/thermocouple junction was 575°F. The critical temperature for this junction is 700°C, above which the magnet begins to lose its holding power. It is concluded, then, that this portion of the unit suffered no detrimental effects due to the drywell high temperature of 250°F (121°C).

Nonmetallic components for the temperature elements exposed to the drywell atmosphere consist of fiberglass lead wire insulation with tinned copper overbraid and a glass-filled phenolic electrical connector. The temperature ratings for these materials are 900° (482°C) and 400°F (205°C), respectively. The 400°F (205°C) temperature limit established for this equipment is based on the phenolic plug. It is assumed that the plug, which is on the end of the leads opposite the magnet, was exposed only to drywell ambient temperature of 250°F (121°C) and not to the 575°F (302°C) RVP maximum temperature. Therefore, its temperature rating was not exceeded. This glass-filled phenolic plug has an expected life of 14.1 years at 250°F (121°C) and of 4432 years at 160°F (71°C). Based on the Pilgrim DHT profile, the plant life equivalency is 615 years at 160°F (71°C). When subtracted from the 4432 years expected life, an equivalent remaining life of 3817 years results. It is concluded that, at the drywell temperature of 160°F (71°C), the glass-filled phenolic connector plug is insensitive to thermal aging effects. At 190°F this equipment has an expected remaining life of 417 years.

Because the thermocouple readout instrumentation is located outside of the drywell, an aging analysis was not performed on its nonmetallic components.

Resolution of Deficiencies

The evaluation indicated that the drywell event had no unique detrimental effects on the serviceability of this equipment.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1. The equipment was evaluated for possible detrimental effects of the event with the result that:

   a) Equipment design limits were not exceeded

   b) Materials design limits were not exceeded

   c) The equipment contains no materials known to be age-sensitive

   Having met the above acceptance criteria, it is concluded that the drywell event had no detrimental effects on the serviceability of this equipment.

Reference

1. Evaluation of Possible Detrimental Effects of the Drywell Event on Pyco Inc. Part No. 13-2012 Magnetic Thermocouples, Wyle Report No. 17536-22, dated December 198[illegible].

4) REACTOR VESSEL SURFACE TEMPERATURES (PYCO THERMOCOUPLES)

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
TE 263-66-xx*	Assumed 1971	Various	250°F	Unknown
TE 263-67-xx	Assumed 1971	Various	250°F	Unknown
TE 263-69-xx	Assumed 1971	Various	250°F	Unknown

* xx denotes an alphanumeric code of some 29 thermocouples at various locations on the reactor vessel, exclusive of magnetic head thermocouples which are addressed separately.

Summary of Equipment Evaluation[1]

These thermocouples are in contact with various reactor vessel structures and sense temperatures in excess of the drywell atmosphere. Consequently, these thermoucples are not considered to have experienced any unique detrimental effects due to the drywell event.

Resolution of Deficiencies

The evaluation indicated that the drywell event had no unique detrimental effects on the serviceability of this equipment.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis that this equipment is designed to function in an environment which is considerably more severe than that associated with the drywell event.

5) SAFETY VALVE RV 203-4A,B,C,D TEMPERATURE
(LEEDS & NORTHROP THERMOCOUPLES, MODEL NO. 8734-T-1-4-7/8)

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
TE 6271-A,B through	Assumed 1971	47 ft	240°F	Unknown
TE 6276-A,B	Assumed 1971	47 ft	240°F	Unknown

Summary of Equipment Evaluation[1]

These thermocouples are welded into recessions that protrude into the discharge piping of the safety valves and sense the temperatures of the piping during normal operation. Consequently, these thermocouples are normally exposed to temperatures in excess of the drywell atmosphere. It is concluded that the drywell event had no unique detrimental effects on the serviceability of this equipment.

Resolution of Deficiencies

The evaluation indicated that the drywell event had no unique detrimental effects on the serviceability of this equipment.

Justification for Continued Operation

Continued operation of this equipment justified on the basis that this equipment is designed to function in an environment which is considerably more severe than that associated with the drywell event.

III-2.1 Mechanical Equipment

Mechanical equipment located in the drywell associated with both safety-related systems as well as non-safety systems considered essential for normal plant operation were evaluated for possible detrimental effects of the drywell event. This evaluation placed emphasis on inspections and tests as a means of determining physical damage; although materials degradation analyses were performed where possible. Mechanical equipment considered in this evaluation are identified in Table III-2. The information presented in this section is an item-by-item description of the results of this evaluation and presents specific actions, where required, which either have been performed or which are planned for completion prior to continued operation of PNPS-1. The format of the presentation is similar to that employed for electrical equipment. The presentation for each item consists of:

a) Identification of the Equipment

b) Summary of Equipment Evaluation

c) Resolution of Deficiencies

d) Justification for Continued Operation

e) References

TABLE III-2

LISTING OF MECHANICAL EQUIPMENT EVALUATED

No.	Description Equipment
1.	Bergen Patterson Snubbers
2.	ITT Air Operators
3.	Hydroline Air Operators

1) BERGEN PATTERSON SNUBBERS

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
SS-2-20-1	1979	42 ft	$240^{o}F$	Unknown
SS-2-20-20	1979	16 ft	$137^{o}F$	Unknown
SS-2-20-21	1979	19 ft	$152^{o}F$	Unknown
SS-2-20-25	1979	16 ft	$137^{o}F$	Unknown
SS-2-30-10	1977	11 ft	$131^{o}F$	Unknown
SS-2-50-26	1977	16 ft	$137^{o}F$	Unknown
SS-6-10-1	1977	42 ft	$240^{o}F$	Unknown
SS-6-10-4	1977	42 ft	$240^{o}F$	Unknown
SS-6-10-7	1977	41 ft	$167^{o}F$	Unknown
SS-6-10-8	1980	44 ft	$240^{o}F$	Unknown
SS-10-20-2	1977	52 ft	$240^{o}F$	Unknown
SS-10-20-3	1977	52 ft	$240^{o}F$	Unknown
SS-10-20-8	1979	24 ft	$152^{o}F$	Unknown
SS-10-30-1	1977	52 ft	$240^{o}F$	Unknown
SS-23-10-1	1980	42 ft	$240^{o}F$	Unknown
SS-23-10-2	1977	42 ft	$240^{o}F$	Unknwon
SS-10-3-10	1977	90 ft	$250^{o}F$	Unknown

Summary of Equipment Evaluation

1. All snubbers were visually inspected for indication of leakage, physical distortion, discoloration, or obvious physical damage. None of these indications were observed.

2. Sixteen snubbers distributed between the +11 and +52 ft levels were tested in the snubber test rig. All snubbers passed all tests including bleed rate, lockup, leakage, and piston movement. Testing is in progress on snubber SS 10-3-10. In the event it fails, sufficient snubbers between +52 and +90 feet will be tested to ensure the integrity of remaining snubbers.

3. Seals were verified to be comprised of ethylene propylene rubber (EPR) and the hydraulic fluid verified to be SF 1154.

Resolution of Deficiencies

No deficiencies were noted.

Justification for Continued Operation

Continued use of this equipment is justified based upon a representative sample of the snubber population successfully passing all inspections without discrepencies. In the event snubber SS 10-3-10 fails, it will be rebuilt/replaced and sufficient snubbers between the +52 and +90 elevations will be tested (and rebuilt, as necessary) to ensure the integrity of the remaining snubbers.

2) ITT HAMMEL-DAHL AIR OPERATORS

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
AO-220-44,46,47,51	(Assumed 1972)	74 ft	242°F	180°F*
AO-220-52	(Assumed 1972)	90 ft	250°F	180°F

* The pressure regulator/filter supplied with the air operator is rated at 150°F by the manufacturer.

Summary of Equipment Evaluation[1]

The manufacturer's rated temperature has been exceeded on all listed air operators listed above. All of these air operators have age sensitive parts; Buna-N O-rings and Buna-N/Nylon diaphragms. The expected life of Buna-N at 150°F for 20% loss of initial elongation is 39 days. For these operators, an inspection/maintenance program is required to physically evaluate possible detrimental effects of the drywell event. This program should include, as a minimum, the following actions:

1. Inspection and replacement, as needed, of non-metallic materials in the remaining air operators and regulators.

Resolution of Deficiencies

Air operators will be disassembled, inspected and rebuilt starting with the operator at the highest elevation (A0220-52) and proceeding through operators at lower elevations until air operators are found that have no evidence of non metallic materials degradation. Internals will be replaced as required. Air regulators on all listed air operators will be replaced.

Justification for Continued Operation

Following completion of the above inspection/replacement, continued operation of this equipment is justified based upon the fact that all degraded components of operators subjected to elevated temperatures will have been replaced.

References

1. Evaluation of Possible Detrimental Effects of the Drywell Event on ITT Hammel-Dahl Air Operator, Wyle Report No. 17536-27, dated December, 1981.

3) HYDROLINE AIR OPERATORS

Identification of Equipment

EQUIPMENT TAG NO.	INSTALLATION DATE	ELEVATION	MAXIMUM TEMPERATURE	MANUFACTURER'S RATING
AO-2031A,B,C,D,	(Assumed 1972)*	27 ft	152°F	200°F**

*Refurbished 1977 outage
**Based on polyurethane seal temperature rating.

Summary of Equipment Evaluation[1]

The manufacturer's rating has not been exceeded on Air Operators AO-203-1A,B,C, and D. Additionally, these Air Operators were completely refurbished during the 1977 outage and all nonmetallics were replaced. Analysis indicates that the viton, teflon, and polyurathane seals and o-rings are insensitive to aging at 152°F.

Contact with the manufactueres, as referenced in the Wyle report, indicates an operational check should be conducted to verify operational integrity.

Resolution of Deficiencies

The evaluation indicated that the drywell event had no detrimental effect on the operational integrity of this equipment.

Justification for Continued Operation

Continued operation of this equipment is justified on the basis of the following arguments:

1) The equipment was evaluated for possible detrimental effects of the drywell event with the result that:

   a) Equipment design limits were not exceeded.

   b) Materials design limits were not exceeded.

   c) Material degradation was within acceptable limits such that the remaining expected life is considerably greater than the remaining plant desigh life.

   Having met the above acceptanct criteria, it is concluded that the drywell event had no detrimental effects on the operational integrity of this equipment.

2) This equipment will be subjected to operation checkout in accordance with BECo's startup management program.

References

1) Evaluation of Possible Detrimental Effects of the Drywell Event on Hydroline Air Operators, Wyle Report No. 17536-17, January, 1982

III-3 DRYWELL EQUIPMENT ACTION ITEMS

Table III-3 presents, in tabular form, a summary of the safety evaluation for each item of equipment in the drywell considered to be 1) safety-related electrical equipment, 2) non-safety electrical equipment essential to plant operation, and 3) mechanical equipment. This data summary addresses equipment maintenance and replacement actions which were implemented to resolve any possible detrimental effects arising from the event.

TABLE III-3
SUMMARY OF EQUIPMENT ACTION ITEMS

A1	ASCO Sol. Valve	None
A2	AVCO Sol. Valve	None
A3	NAMCO Limit Sw.	Install gasket & seal replacement kit per mfg. 18-month maintenance procedure
A4	Target Rock Sol Valve	Rework per mfg's 6-year maintenance instructions
A5	Limitorque MO	Maintenance/replacement (See detailed write-up)
A6	Acoustic Monitors	None
A7	Junction Boxes	Inspection/replacement of enclosure seals as req'd.
A8	Terminations	Inspection to determine proper install. clearance & need for Raychem insul. splice.
A9	Kerite Cable	Periodic testing per Kerite recommendation
A10	Okonite Cables	Replace above 41 ft. None below 41 ft.
A11	GE SIS Wire	None
A12	Raychem Splices	None
A13	GE Penetrations	Pressure Test
A14	Phys. Science Penet.	Pressure Test

B1	Omega Thermocouples	New - None required
B2	Pyco RTDs	New - None required
B3	Pyco Mag. Hd. Thermo.	None
B4	Pyco Thermo (welded)	None
B5	Leeds & Northrop Thermocouples	None
C1	Snubbers	Maint./inspection per write-up
C2	Air Operators (Hammel-DAHL)	Inspection/maintenance, replacement as required
C3	Air Operators (Hydroline)	None

A - Safety-related electrical equipment
B - Non-safety electrical equipment
C - Mechanical equipment