PROPOSAL

ANTICIPATED TRANSIENTS WITHOUT SCRAM-RECIRCULATION PUMP TRIP

(ATWS-RPT)

PILGRIM

416-4208-HK1

March 1979

GENERAL ELECTRIC COMPANY

NUCLEAR ENERGY PROJECTS DIVISION

BWR SERVICES DEPARTMENT

F/1

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1. Background

The NRC has issued the requirement for installation of an Anticipated Transients Without Scram-Recirculation Pump Trip (ATWS-RPT) modification for all operating plants within two years. Further, they have indicated licensing acceptance of the General Electric Company "Monticello ATWS-RPT" design for all operating plants. Finally, they have requested the utility's implementation schedule by April 9, 1979.

The General Electric Company proposes to furnish the plant improvement program described in Section 4.5 of Topical Report NEDO-25016, "Evaluation of Anticipated Transients Without Scram for the Monticello Nuclear Generating Plant", to meet this requirement.

2. Design Objectives

The intent of the ATWS-RPT design is to meet the following objectives:

1. Shut down all recirculation pump motor generator sets with redundant logic from the following inputs:

   ~~a) Manual Initiation~~ Rev 1 MHD

   b) Reactor Vessel High Pressure

   c) Reactor Low Water Level

2. The system is to be diverse from the Reactor Protection System (RPS).

3. The system is to be testable in service.

4. The system is to be designed so that as much as possible no single component failure can prevent the tripping of both recirculation pumps.

5. The hardware should be high quality and environmentally qualified.

6. The system's performance characteristics are as follows:

   a) Logic delay for trip, including dynamic response of the sensors, logic, action of the breakers and collapse of the generator field — $\leq$ 0.53 seconds

   b) Low level delay timer — (to be confirmed by plant unique analysis)

3. Justification

Economic justification for the utility to accept the proposal results from successful realization of the objectives discussed in Section 2. Further, the ATWS-RPT system improvement package is a standard, generic design, with appropriate licensing report.

4. Hardware Description

1. Transmitters

The reactor pressure and reactor water level will be sensed by analog transmitters. The pressure and level transmitters used are designed and manufactured to General Electric specifications. Calibration and shutoff valves, 3-way balancing manifolds and fittings are provided for field assembly. See Figures (1) and (2) for typical sketches of a Pressure Transmitter and Differential Pressure Transmitter Valve subassembly.

2. Trip Units

The trip unit used to generate the trip signal is designed and manufactured to General Electric specifications. Features of this system are:

1. Trip units are functionally tested or calibrated in place by means of a portable readout assembly.
2. Trip unit calibration current is a controlled ramp of 1 ma/sec.
3. The output of the trip unit is a voltage to a Class 1E relay.
4. The master trip unit has a display meter which monitors the transmitter current for gross failures as well as failures in the current to voltage and filtering sections of the master trip unit.
5. Trip unit "out-of-card file annunciation" is provided.
6. The trip unit provides long-term trip point stability.
7. The calibrator is capable of introducing a step current for transient testing of the logic.
8. The readout assembly is a portable unit which may be used with all calibration units. This provides for portability of the secondary standard from card file to card file to measure the value of the trip point in various cabinets.
9. Card extenders, a bench test unit, and calibration units are supplied to facilitate maintenance as required.

10. The trip relay is a general purpose relay with four form C contacts which is rated for the ambient environment at its mounting location.

3. Cabinets

General Electric designed cabinets which house the trip unit card file, DC power supplies, and trip relays are illustrated in Figures 3 and 4. The large cabinet (Figure 3) has space to house up to three trip unit card files thereby permitting use of the cabinet for future installation of the Analog Trip System or other plant improvements. The small cabinet (Figure 4) has space for only one card file. Each card file has space to accommodate twelve trip units. Additional card files can be factory supplied for large cabinets per the price tabulation in the quotation letter, or added later by field installation.

The 25 volt DC power supply used is a ferro resonant type which is recognized for its high reliability. Two complete power supplies are mounted in each trip unit cabinet and connected in parallel. With this arrangement a single power supply can fail without affecting the trip function. The power supplies in the large cabinet have sufficient capacity to supply power for the maximum number of trip units (36) that may eventually be housed in a large cabinet. The power supplies for the small cabinet have sufficient power to supply the 12 trip unit capacity of that cabinet.

The cabinets may be located in the control room, auxiliary equipment room (location of existing logic cabinets), or in the vicinity of the local instrument rack in the reactor building. The cabinets must be located in an area where maximum ambient temperature will be between 40-145°F. The electrical components within the cabinet are qualified to operate up to 150°F at 99% relative humidity. The cabinets are seismically qualified per IEEE-344, 1975, to the Safe Shutdown Earthquake (SSE) acceleration response spectra shown on Figure 5. The large cabinets are floor mounted; the small cabinets are wall mounted.

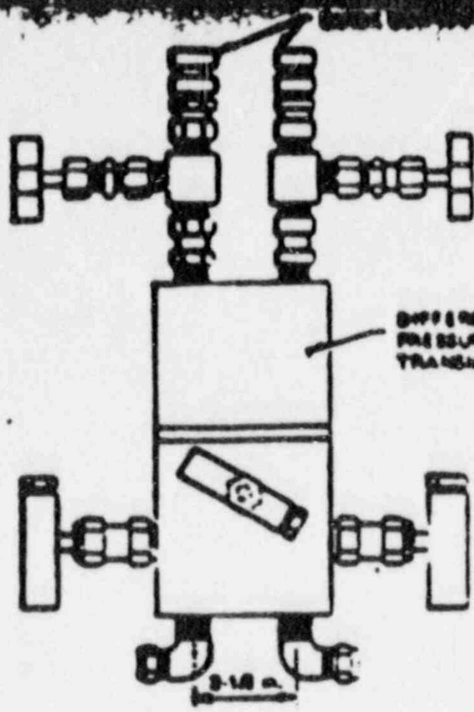


Figure 1 Differential Pressure Transmitter

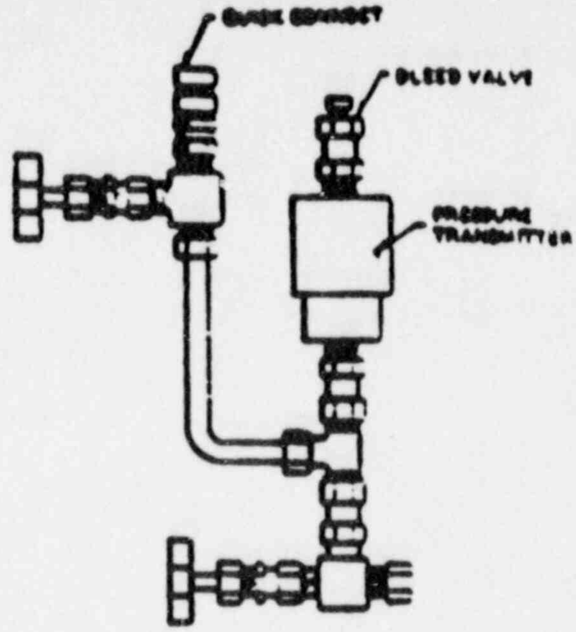


Figure 2 Pressure Transmitter

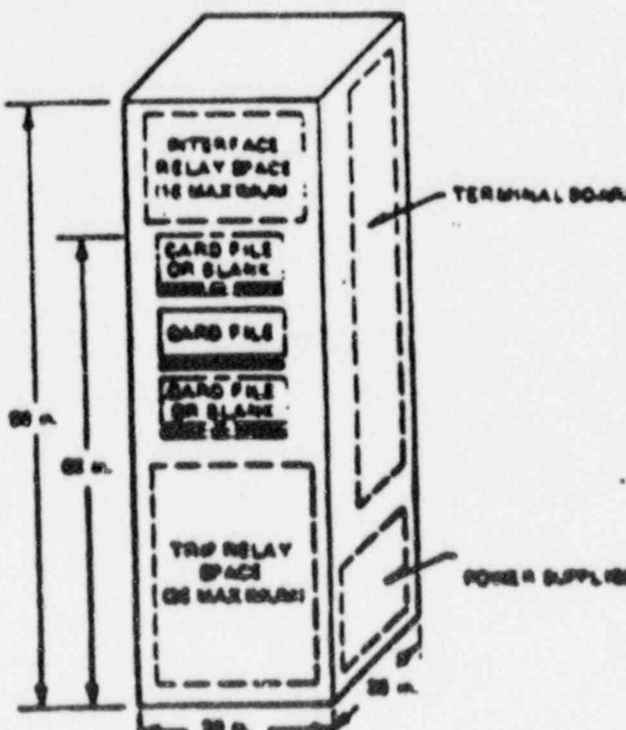


Figure 3 Large Trip Unit Cabinet

PRELIMINARY

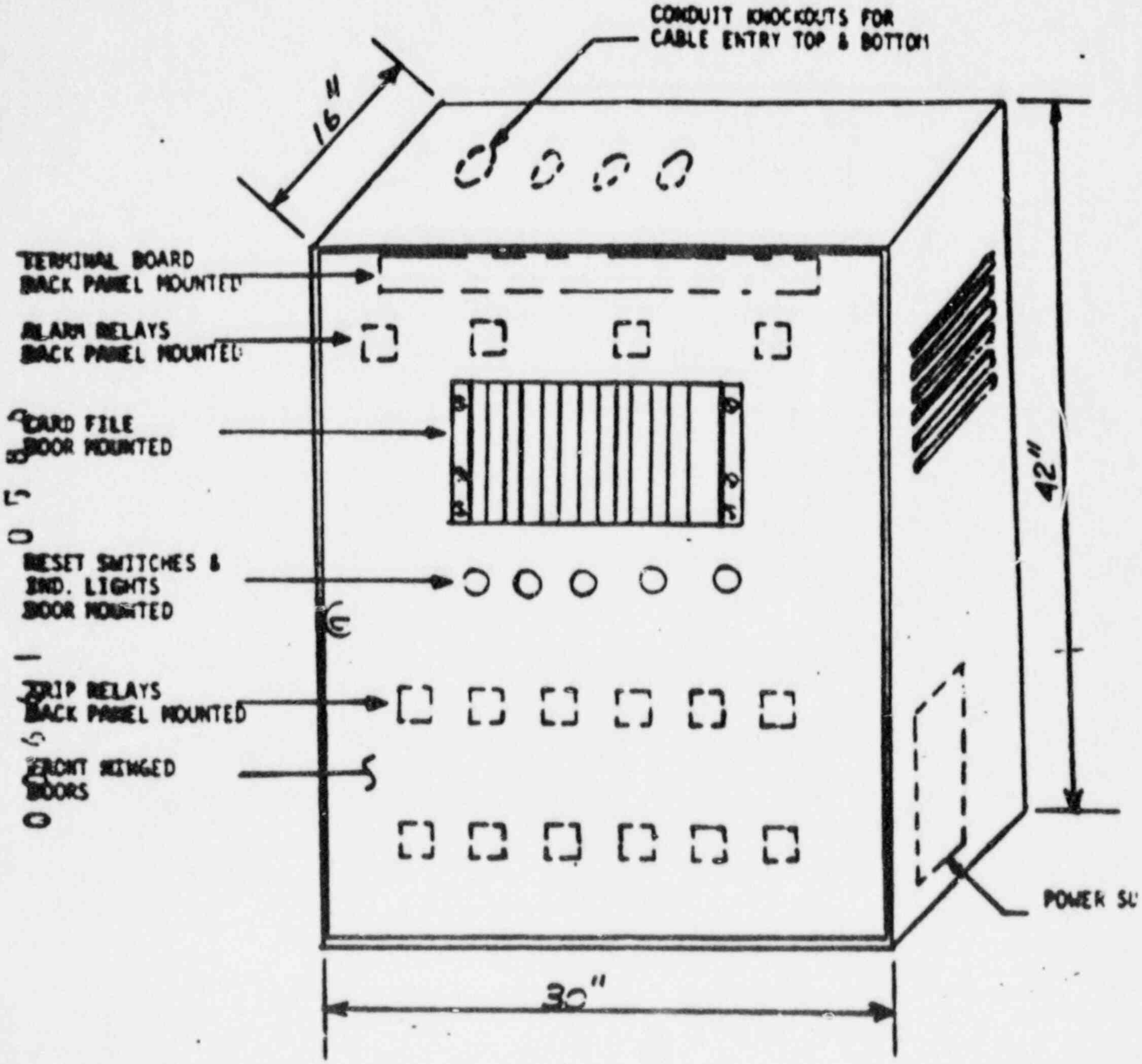


Figure 4
Small Trip Unit Cabinet (Wall Mounted)

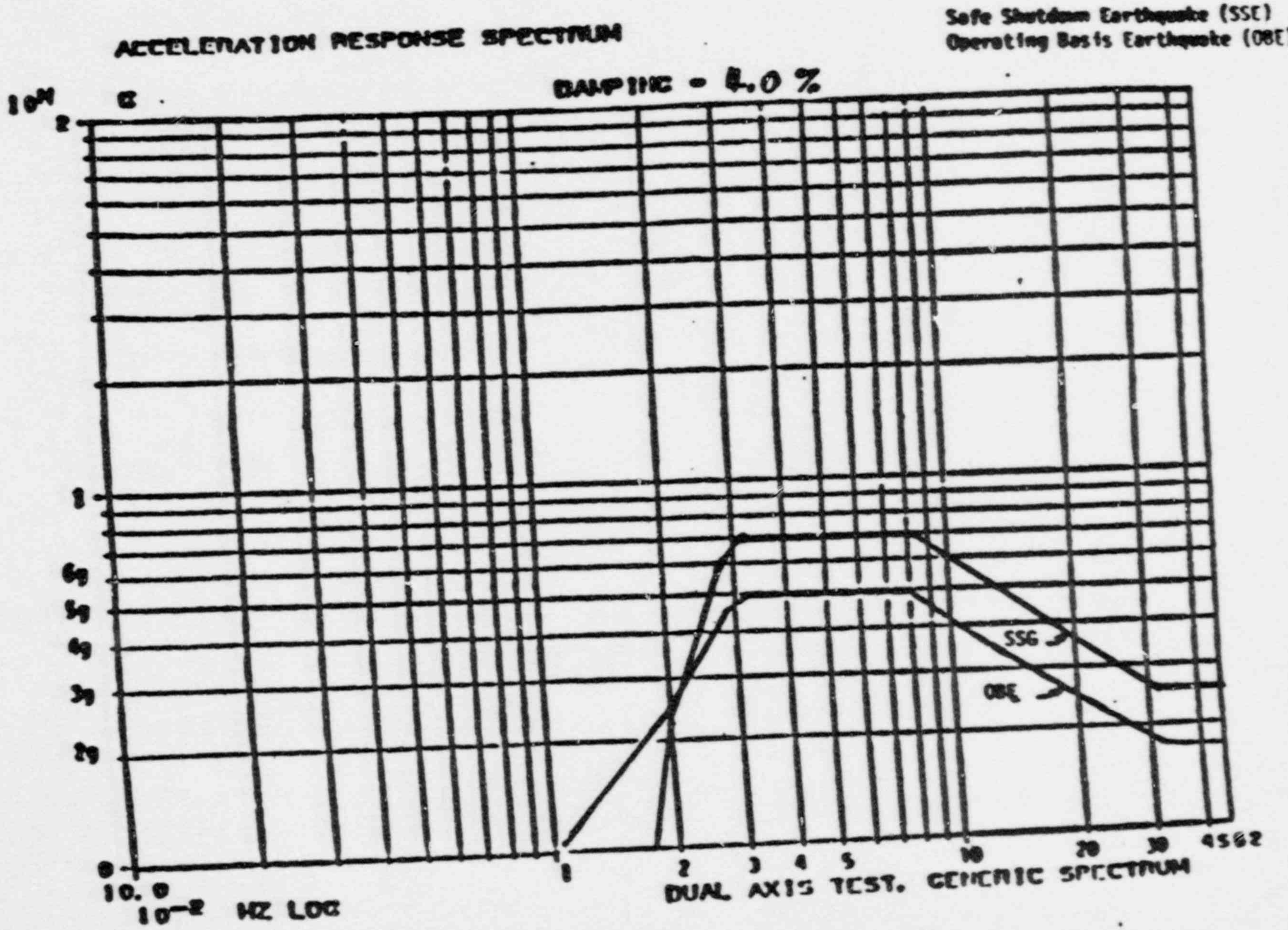


FIGURE 5 TEST RESPONSE SPECTRUM FOR CONTROL & INSTRUMENTATION EQUIPMENT

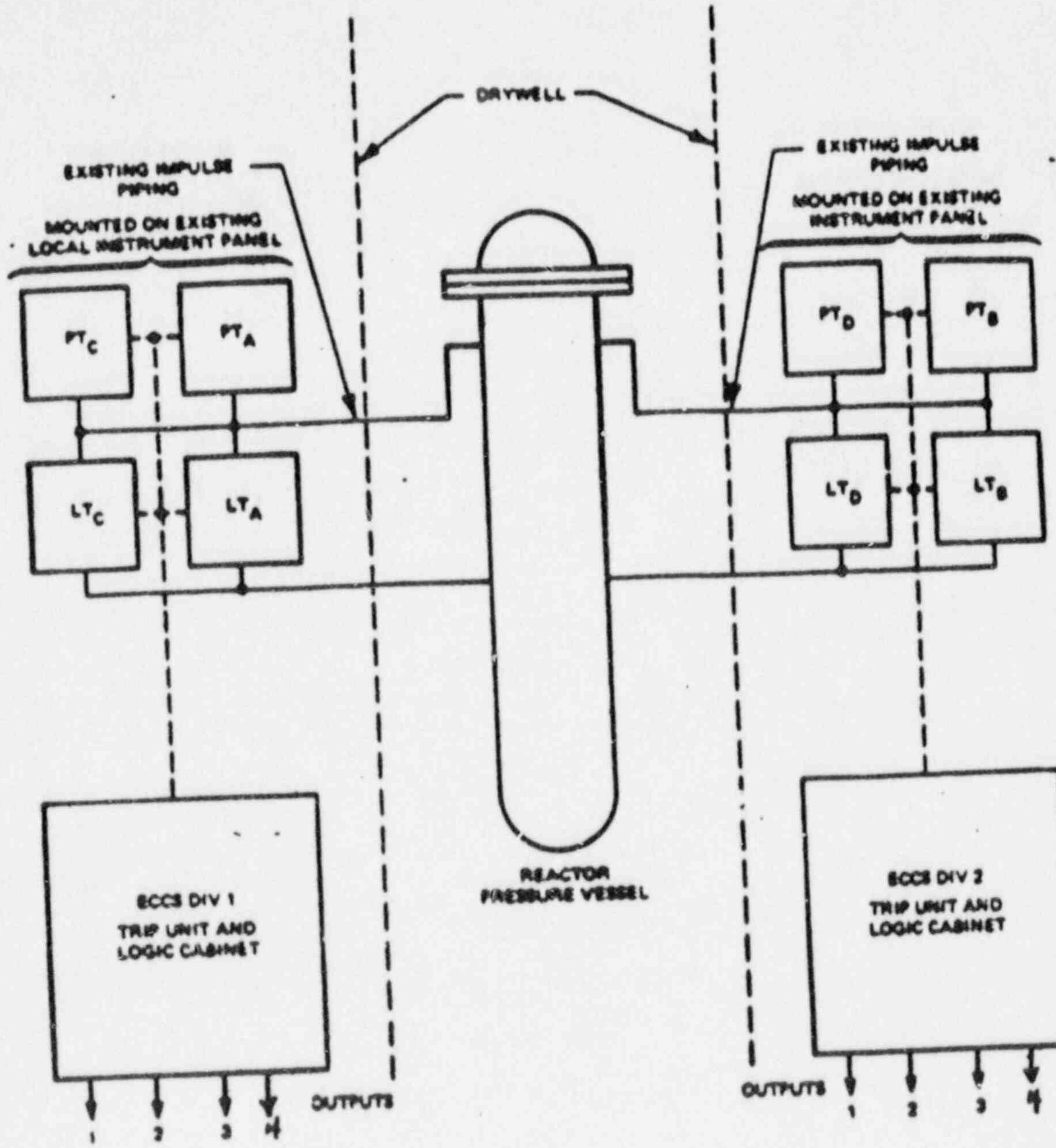
4. <u>Inverter</u>

An additional DC to AC inverter is required in each ECCS division to enable the ATWS-RPT trips to have their backup power source from the station batteries. The inverter is to be located and installed by the purchaser external to the trip cabinets.

5. <u>System Description and Application</u>

Since normal scram is assumed to be unavailable for reducing the reactor power, and since the transient event is one in which power reduction is necessary, the ATWS-RPT system provides another method of reducing power for the first 15 seconds of the event. The trip of both recirculation pumps causes a quick reduction in core flow thus reducing the power. This quick power reduction brings the reactor pressure, neutron flux and fuel surface heat flux down in time to acceptably limit the peak pressure, clad oxidation and peak fuel enthalpy.

The initiating variables, reactor pressure and reactor water level, are sensed by analog transmitters. The transmitters are mounted on reactor pressure instrumentation racks and connected through calibration and isolation manifolds to existing instrument sensor piping to the reactor pressure vessel. Manifold and calibration valves are supplied to facilitate installation of the transmitter to existing process piping. Figure 6 is a schematic diagram showing the arrangement of the transmitters and the divisional separation. The trip units are mounted in cabinets which conform to ECCS separation criteria. The trip units provide the inputs to the logic which trips both Recirculation Pump Motor-Generator Sets' field breakers, thus shutting down the recirculation pumps.



DESTINATION OF OUTPUTS

1. CONTROL ROOM MANUAL INITIATION [illegible]
2. DIVISIONAL ECCS LOGIC CABINET (OUTPUT FROM INVERTER)
3. RECIRC PUMP BREAKER TRIP COIL
4. ARI Scram Pilot Air Header Values

Figure 6

6. <u>General Electric Responsibilities</u>

1. Prepare cabinet outline drawings
*2. Design, build and wire check the trip unit cabinets.
3. Provide the hardware listed in Table 1.
4. Prepare Elementary Diagrams showing devices added with location, termination numbers and interconnection per the options in the quotation letter.
5. Provide instruction manuals.
6. Provide an ATWS-RPT system specification.
7. Provide transmitter and related valve assembly drawings.
8. Provide an installation specification with installation, calibration, and startup instructions.
9. Provide an ATWS-RPT system description suitable for licensing submittal.

* Cabinet design will vary with the cabinet option purchased.

7. Purchaser Responsibilities

1. Physically locate and fasten the cabinets securely to the floor or wall, as required (if cabinets are supplied).
2. Run raceways from the trip units to the transmitter racks, as required.
3. Mount, assemble and pipe the transmitter/valve subassembly to the process piping. Leak test the new transmitter installation.
4. Make electrical connections between the transmitter to the trip unit and the trip unit relay to trip coils and vent solenoids.
5. Run power cables from the existing logic cabinets to each trip unit cabinet. Existing spare cables that have been made available from other plant modifications may be used.
6. Run annunciator loop cables from the control room to each trip unit cabinet.
7. Revise the plant unique P&ID's, FCD's, Elementaries, ID's, etc.
8. Provide labor for installation.
9. Provide all licensing for the application of the hardware into the power plant.
10. Checkout and placing in service.
11. Verify the seismic spectrum the cabinets were tested to (Figure 5) is greater than the floor/wall response for the mounted location of the cabinets.
12. Perform seismic analysis as may be required to verify the seismic conditions at the locations where the transmitters will be installed do not exceed the allowable equipment design limits.
13. Verify the environmental conditions at the locations where the equipment will be installed do not exceed the allowable equipment design limits.
14. Incorporate new surveillance testing requirements for the analog trip units and transmitters into the technical specifications.
15. Provide detail test and calibration procedures based upon GE supplied instructions.
16. Locate and install the DC to AC inverters external to the trip cabinets.

8. Major Components Included in this Proposal

Table 1 is a list of major hardware items which will be supplied by the General Electric Company.

TABLE 1

Major Hardware Items Supplied

Bill Power has

Item	Quantity	Description
1	~~4~~ 5	Level Transmitter, Manifold Valve & Fittings
2	4	Pressure Transmitter, Valve and Fittings
3	8	Trip Unit With Analog Output
4	~~44~~ 42*	Trip and Alarm Relays
~~5~~	~~2~~	~~Reset Pushbutton~~
6	4	Manual Initiation Pushbutton
7	As Required	Light & Holder
8	As Required	Fuse
9	5	Trip Coil for MG Set Field Breakers
10*	2	AC/DC Power Supply
11	3	Readout Assembly
12	3	Card Extender
13	1	Bench Test Unit
14	2	Card File
15	2	Calibrator
16*	2	Cabinet
17	2	DC/AC Inverter
18	4	Isolation valve
* 19	2	Electrical Solenoid operated air valves

*Capacity will depend on cabinet option purchased.

A Correction due to discussions between GE/BG&E 7/3/79 and Rev 1 to Proposal 416-4208-HK1 dated Mar. 1979

[signature]

9. Quality Assurance

1. Equipment and Services shall be provided in accordance with the General Electric BWR quality assurance program as described in Topical Report NEDO-11209-04A.
2. The provisions of 10 CFR Part 21 apply.
3. A Product Quality Certification (PQC) shall be provided by General Electric as the primary quality assurance record for Equipment classified as "important to safety".
4. BWR owner access requirements for audits and/or witness of inspection points for supplied Equipment and Services shall be arranged upon request at mutually agreeable terms.
5. All General Electric Nuclear Energy Divisions' work at the BWR owner's plant site shall be under the cognizance of the BWR owner quality assurance program.

10. Licensing

A description of the ATWS-RPT plant improvement retrofit package suitable for licensing submittal will be supplied. Additional detailed assistance can be provided on a consulting basis at our commercial rates.

June 25, 1985

Docket No. 50-293

DISTRIBUTION
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MEMORANDUM FOR: Richard W. Starostecki, Director
Division of Reactor Projects
Region I

FROM: Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

SUBJECT: LICENSING ACTION REVIEW

Re: Pilgrim Nuclear Power Station

Your assistance is requested in conducting a review of the responses by the Boston Edison Company (BECo) to Generic Letter 83-28, Items 3.1.1 and 3.1.2, 3.2.1 and 3.2.2, and 4.5.1. This work should be conducted under TAC numbers 52948, 53785 and 54095, respectively.

Enclosed are copies of BECo letters dated November 7, 1983 and June 28, 1984, which address the above items. If further information is needed from the licensee, it should be obtained via Project Manager Paul Leech (FTS) 492-4952.

In accordance with NRR Office Letter No. 44, each safety evaluation performed by a technical division shall have a separate SALP input provided. For the purposes of these reviews, the Regional personnel are considered part of the technical divisions. Therefore, we are requesting that your forwarding memorandum contain a SALP input for the evaluation performed.

As discussed with Don Haverkamp of your staff, the requested target date for completion of your review is March 31, 1986. The period from now until that date was established in anticipation of the need for additional information from the licensee. Please confirm this target date or provide a revised date as soon as possible.

Original Signed by FJMiraglia for/
Hugh L. Thompson, Jr.

Hugh L. Thompson, Jr., Director
Division of Licensing
Office of Nuclear Reactor Regulation

Enclosures:
As stated

DL:ORB#2	DL:ORB#2	DL:AD-OR	DL:D
PLeech:ajs	DVassallo	GLainas	HThompson
06/24/85	06/24/85	06/29/85	06/25/85

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Generic Letter 83-28

1.1.1 Restart of Pilgrim Station is acceptable only if the following criteria are satisfied:

1) The cause of the unscheduled reactor shutdown has been determined and appropriately corrected or the Operations Review Committee (ORC) has determined that the safety of equipment, site personnel, and the public are not threatened by a restart, based on an independent review (see response to 1.1.6).

2) Components within systems designed for automatic response to abnormal parameters did indeed respond properly to the appropriate initiating signals, or exceptions evaluated and approved by the appropriate administrative controls (ORC, Nuclear Engineering Department, Safety Evaluation).

3) The station manager, or his designated alternate, has given approval to commence restart.

1.1.2 Post-trip review activities are conducted by the on-duty Nuclear Operating Supervisor and Shift Technical Advisor under the direction of the Nuclear Watch Engineer. If the cause of the trip is not readily apparent, or cannot be determined beyond a reasonable doubt, the Chief Operating Engineer or Day Watch Engineer will take charge of the investigation until the cause of the trip has been determined. The individual in charge of the investigation will take responsibility for making appropriate recommendations to the station manager, or his designated alternate, based on the criteria of 1.1.1 above.

1.1.3 The qualifications of the facility staff are addressed in Section 6.3 of Pilgrim's Technical Specifications. This requires that the requirements of ANSI N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants" be met.

The Shift Technical Advisor is, as a minimum, qualified to the requirements described in NUREG-0737.

1.1.4 Output of the alarm typer on the plant process computer provides the primary information source for chronological reconstruction of the sequence of events surrounding a trip occurrence. Recorder strip charts are utilized for evaluation of long-term trends which may not be indicated by the alarm typer output. The balance of essential information is provided by plant personnel, whose combined knowledge is relied upon for reconstruction of human activities prior to and during the trip event (maintenance activities, operator actions, etc.).

1.1.5 Technical Specifications identify the trip settings for Reactor Protection, Containment Isolation, and Emergency Core Cooling Systems actuation. The information sources identified in 1.1.4 above indicate the presence of any signals exceeding those trip settings. Expected plant behavior is based on system design responses described in the FSAR and on the experience and training of operators and supervisors.

1.1.6 If the post-trip task force identified in 1.1.2 above is unable to establish the cause of the trip, the Operations Review Committee is convened to provide independent assessment of the event. The ORC utilizes the same information as the task force. Both groups may call on the technical and engineering expertise of other personnel in the Nuclear Organization or other appropriate group, internal or external.

1.1.7 We are in the process of reviewing a draft INPO procedure concerning post-trip reviews for possible incorporation into a Nuclear Operations Procedure (NOP). Currently, Section 6 of NOP8301, "Conduct of Operations," deals with this issue (Attachment A).

1.2.1.1 The GE-PAC 4020 Process Computer System provides on-line monitoring of several hundred input points (digital, analog, and pulse) representing significant plant process variables. The system scans digital and analog inputs at specified intervals and issues appropriate alarm indications and messages if monitored analog values exceed predefined limits or if digital trip signals occur. It performs calculations with selected input data to provide the operator with essential core performance information through a variety of logs, trends, summaries, and other typewriter data arrays. The Sequence of Events printer responds to digital signals, the Data Recall Log is analog.

1.2.1.2 The monitored parameters are listed in Attachment B.

1.2.1.3 The log gives a time field for events in hours, minutes, seconds and then to the nearest 1/60 (.0166) second.

1.2.1.4 The format for displaying data and information is as follows:

Sequence of Events:

Time	Cycle	Point ID	Name	Status
XXXXXX	XX	XXXX		XXXX

Date Recall Log

Time	Point ID 1 .... Point ID 19
	(Value)
XXXXXX	XXXXXX XXXXXX

The Data Recall Log prints values preceding the event in black (2.5 minutes before event), values following the event are printed in red (2.5 minutes after the event).

1.2.1.5 The computer has core storage to record the status of the first 80 events in sequence for the NSS log and the first 20 events in sequence for the BOP log. The data is typed on the alarm typer. During this interval of time, (approx. 5 min.) all change of contact status is ignored. At the termination of the typer routine, the program rescans all digital points again and outputs any change of status on the typer, and reinitializes the program again.

The hard copy of the printout is controlled by the Document Control Group and is retained in the records vault at Pilgrim.

1.2.1.6 The power source for the process computer is non-interruptible and is non-Class IE.

1.2.2.1 As described above, the GE-PAC 4020 Process Computer System provides the major information source for assessing the time history of analog variables. Additionally, several variables (see 1.3 response) are recorded on strip-charts for trend evaluation.

1.2.2.2 The Data Recall Log monitors up to 38 preselected analog points, which are scanned continuously at 5-second intervals.

The Data Recall Log currently supplies information on the following:

APRM Channel "A"
APRM Channel "C"
Reactor Pressure
Core Plate P
Reactor Core Flow
Control Rod Drive Flow
Reactor Feedwater Flow "A" Loop
Reactor Feedwater Flow "B" Loop
Reactor Water Level (inches)
Outlet Steam Flow
Feedwater Temperature "A" Loop
Feedwater Temperature "B" Loop
Recirculation Flow "A" Loop
Recirculation Flow "B" Loop
Reactor Saturation Temperature
Calculated Seawater Flow
Hotwell Outlet Temperature
Drywell Temperature (64' elevation)
Suppression Chamber Level
Stator Cooler Header Inlet (°C)
Stator Cooler Header Outlet (°C)
Alternator Air to Cooler (°C)
Alternator Air from Cooler (°C)
Condensate Demineralizer Differential Pressure
Reactor Feedpump Suction Pressure
Condensate Pump Discharge Header
West Condenser Pressure (inches Hg)
East Condenser Pressure (inches Hg)

Reactor Building Closed Cooling Water System (RBCCW) "A" Loop Flow
RBCCW "B" Loop Flow
RBCCW Residual Heat Removal (RHR) Heat Exchanger Loop "A" Flow
RBCCS RHR Heat Exchanger Loop "B" Flow
RBCCW "A" Outlet Temperature
RBCCW "B" Outlet Temperature
Torus Pressure
Drywell Pressure
Service Water Loop "A" Flow
Service Water Loop "B" Flow

1.2.2.3 The above parameters are stored in a special Scan Table section of computer memory for 2.5 minutes. Upon occurrence of a designated plant trip event, the data currently in the Scan Table is transferred to a special Output Table and frozen, while the program continues to collect and save data at the same 5-second scan rate for the next 2.5 minutes. This five minutes of data is then displayed on the alarm typer. Strip chart information is continuously displayed, so that time history is dependent only on the requirements of the evaluation team (see 1.3 response).

1.2.2.4 The format for displaying data and information is standard for the GEPAC 4020 (see Response to 1.2.1.4).

1.2.2.5 Retention and retrievability is provided by the Document Control Group. The hard copy of the printout is controlled by this group and retained in the records vault at Pilgrim.

1.2.2.6 The power supply to the process computer is non-interruptible and non-Class IE.

1.3 The following is a list of instrumentation available in the main Control Room which may be used as needed for the assessment of unscheduled shutdowns:

1) PR-3392 Condenser Vacuum Strip Chart (reads in inches of mercury).

2) PR-3050 Turbine Main Steam Pressure (850-1050 PSIG).

3) VR-3000 Turbine Vibration Trip T/G at 12 mils.

4) 640-26 Two pen recorder: black pen records vessel level 0"-60"; the red pen records feedwater flow $0\text{-}10 \times 10^6$ lbs/hour.

5) 640-27 Two pen recorder: black pen records wide range vessel pressure 0-1500 PSIG; red pen records reactor steam flow $0\text{-}10 \times 10^6$ lbs/hour.

6) 640-28 Two pen recorder: black pen records turbine steam flow $0\text{-}10 \times 10^6$ lbs/hour; red pen records narrow range pressure 950-1050 PSIG.

7) 750-10 A, B, C, D APRM Neutron Flux 0-125%, Two pen recorder for 6 channels of APRM.

1.4 General Electric has been contracted to replace the existing plant process computer, with expected completion by the end of 1986.

2.1.1 Safety-related systems, structures, and components (SS&C) are identified in the Pilgrim Nuclear Power Station (PNPS) Q-List which is described in the item 2.2.1.2 response. Documents (Purchase Orders, Maintenance Requests) used to control activities associated with the Q-Listed equipment are identified as "Q" and subject to the requirements of 10CFR50, Appendix B and the Boston Edison Quality Assurance Manual (BEQAM) (see response to 2.2.1.4). Components which are required to function for a reactor trip are identified in the Q-List and are, therefore, controlled at a quality level consistent with their safety-related functions.

2.1.2 Records documenting the original qualification and testing of existing safety-related equipment are retained as quality assurance records and controlled in accordance with the 10CFR50, Appendix B, Criterion XVII requirements described in Boston Edison's Quality Assurance Manual (BEQAM), Volume II. This encompasses documentation for equipment which serves a reactor trip function.

Nuclear Operations Procedure (NOP83A1) defines Pilgrim's Technical Group as responsible, when requested, for station evaluations of, among other externally generated information, Bulletins, Circulars, Service Information Letters, and Technical Information Letters.

We realize that there is an effort by GE (through the BWROG) and a NUTAC (see 2.1.3) on vendor interface which may require changes to our existing systems. We shall inform the NRC of plans for such changes, if necessary, after we have assessed what GE and the NUTAC have provided to us.

2.1.3 We are participating in the Nuclear Utility Task Action Committee (NUTAC) on vendor interface, which is expected to provide results and recommendations in February, 1984. We wish to review this material, assess what impact it has on current BECo programs and procedures, and develop appropriate commitments and schedules. Based on the NUTAC's February date, we shall provide our commitments and date of completion in April, 1984.

2.2.1.1 Components within systems classified as safety-related are themselves considered safety-related if they function in some capacity to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposure of 10CFR Part 100. This will henceforth be referred to as a safety-related function.

This criteria is applied as follows:

1. Civil Structures which are required to maintain their integrity to assure performance of a safety-related function are considered safety-related. This includes all elements of the structure which are essential to maintenance of its structural integrity. A safety-related structure may provide its assurance of safety either (a) directly (Reactor Building perimeter provides Secondary Containment), (b) indirectly through support of safety-related equipment

(specific block walls), or (c) indirectly through housing of safety-related equipment, such that failure of the structure could threaten the performance of a safety-related function (Main Control Room perimeter).

2. Mechanical piping and components which are considered safety-related have been and are designed and installed in accordance with Seismic Class I design requirements. This includes those systems or portions of systems which either directly serve a safety-related function or are in such close proximity to safety-related equipment that failure of the pressure boundary could potentially affect a safety-related function. For those Class I portions of the former type, the Functional Class I Breaks are clearly identified on Piping and Instrument Diagrams (P&ID's) and/or piping isometrics, whereas those Class I portions of the latter type are identified in piping isometrics (with notes on P&ID's to indicate that some portions are Class I). All mechanical piping and components determined to be Seismic Class I are designated as safety-related either passively (pressure boundary only) or actively. Also considered safety-related are the supports and hangers which provide the Seismic Class I protection.

3. For each mechanical component above that serves its safety-related function by actively responding to some electrical stimulus, the electrical assemblies critical to the performance of that safety-related function are considered safety-related. This includes such items as cables, penetrations, junction boxes, conduits, cable trays, panels (and associated internals), supports, and power supplies.

4. Power supplies to safety-related devices are traced back to the originating emergency supply (Battery, Diesel Generator) through applicable switchboards, transformers, switching and breaking devices (including controllers), Motor Control Centers, Distribution Panels, Special Local Control Panels, and all of the associated cables, junction boxes, etc. which are required to transmit power between these stations.

2.2.1.2 Operable safety-related SS&C's are identified in the PNPS Q-List. This list was originally developed by Bechtel from a criteria similar to that described in 2.2.1.1. In February, 1983, BECo completed an effort to verify the contents of the Bechtel-generated list against the latest approved engineering drawings and against the criteria of 2.2.1.1. This assured accuracy of the Q-List for completed plant design changes, as reflected in those engineering drawings. Plant design changes which have been implemented, but are not yet reflected on engineering drawings, and hence are not yet incorporated into the PNPS Q-List, are described in PDC packages which identify the work controls applicable to the associated SS&C's. These are dispositioned on a case-by-case basis. A system is now in effect to identify safety-related SS&C's on a Bill of Materials in the design phase of plant changes.

Validation of the results of the February, 1983 effort was made through independent review by representatives from each engineering discipline, Operations, Maintenance, and Quality Assurance prior to its release. Subsequent revisions are validated by independent and well-documented engineering reviews of requests for Q-List revision.

2.2.1.3 BEQAM, Volume II is applied to all activities affecting safety-related SS&C's. Activities falling within the scope of the QA Program categorically include: designing, purchasing, fabricating, handling, shipping, storing, cleaning, erecting, installing, inspecting, testing, operating, maintaining, repairing, refueling, training, and modifying. Any procedure, maintenance request, work order, purchase order, design change, or other document used to control one of the above activities is required by procedure to indicate either "Q" or "Non-Q" control of the applicable activity. Since safety-related SS&C's are identified in the PNPS Q-List, any activity associated with a Q-Listed item is designated as a "Q" activity and controlled appropriately in accordance with the QA Program work controls.

2.2.1.4 The Boston Edison Quality Assurance Program is defined in the Boston Edison QA Manual (BEQAM), Volume II, and applies to quality-related and quality assurance activities.

The BEQAM requires that structures, systems, and components designated as safety-related, and other items for which the Vice Presidents agree to use the QA Program management controls, be identified on the Q-List. The Q-List is the "information handling system" referred to in NUREG-1000. The BEQAM requires that the Q-List be established and maintained by the Nuclear Engineering Manager.

The Nuclear Engineering Manager implements this responsibility through NED Procedure 6.07, "Maintaining the Q-List." The Q-List is controlled, and the latest revision is distributed to the locations of use.

Three in-process checks are done by the Quality Assurance Department to ensure proper routine use of the Q-List.

1. Plant Design Change Review

   The QAD reviews and approves all proposed plant modifications according to QAD Procedure 3.02, "Review of Plant Design Changes and Major Field Revision Notices." All changes are designated safety-related (Q), or non-safety-related (non-Q) according to the Q-List Classification of the system or component being modified.

   The validity of the Q or non-Q designation is checked by comparing each system or component to be modified with the Q-List. Associated drawings, safety evaluations, and available procurement documents are also checked for consistent Q or non-Q designations.

   The QAD is required to signify approval by signing the Plant Design Change or Major Field Revision Notice.

2. Procurement Document Review

The QAD reviews and approves all preliminary procurement documents according to QAD Procedure 4.01, "Review of Preliminary Procurement Documents Prepared by BECo." All procurement documents are designated Q or non-Q according to the Q-List classification of the item or end use of the service being purchased.

The validity of the Q or non-Q designation is checked by comparing each item or end use service application to be purchased with the Q-List.

The QAD is required to signify approval by signing the preliminary procurement document.

3. Maintenance Request Review

Currently, the Operations Quality Control (OQC) Group reviews all Maintenance Requests for work at Pilgrim Station under the BECo QA Program using QC instruction 5.01, Revision 1, "Quality Control Review of PNPS Maintenance Requests." A checklist is used to ensure proper review and classification of the work based on Station procedures and the PNPS Q-List.

The OQC Group does not review Maintenance Requests for work performed by contractors under their own BECo-approved QA programs.

In addition to the in-process checks, the QAD performs random surveillance inspections and periodic scheduled audits of all QA Program related activities. The preparation, validation, and routine use of the Q-List is within the scope of these inspections and audits. Details of these functions are as follows:

1. Periodic Audits

Planned periodic audits are performed to verify that procedures for preparation, validation, and routine use of the Q-List have been followed and are effective. These audits are performed by qualified personnel not having responsibilities in the areas being audited and using written checklists according to QAD Procedure 18.01. Audit results are documented and reviewed by management, and followup action on deficient areas is taken.

QA audits evaluate the entire Q-List update process to assure that:

- Required changes are forwarded to the Systems and Safety Analysis (S&SA) Group (via a DRN, Revision to "Q" Request (RQR), Plant Design Change Bill of Materials, etc.), which is responsible for maintaining the Q-List.
- An index is maintained by the S&SA Group of requested changes received.

- o Requested changes are reviewed and approved by appropriate personnel for inclusion in a Q-List revision.
- o The Q-List is updated, as required, to reflect approved Q-List changes.
- o The Q-List is controlled and distributed to required personnel.

2. <u>Surveillance Inspections</u>

Currently, surveillance inspections of various plant activities are performed by the OQC Group on a random, unscheduled basis in accordance with QAD Procedure 10.03 and QC Instructions 7.02 and 10.04. Checklists are generally not used. Selection criteria is not formalized expect as delineated in QAD Procedure 10.03. Surveillance inspection reports are issued to document these surveillances.

Based on recommendations from the NRC and INPO, a draft change to QAD Procedure 10.03 has been prepared to redefine the scope, purpose, and implementation of the surveillance (monitoring) function. The QAD Procedure will require that surveillances be scheduled and unscheduled, random and selective, and with sufficient detail to effectively monitor and report the conditions at PNPS. The surveillances are performed in support of, and as supplements to, audits and inspections to provide quality assurance coverage of station in-process activities. The scope of monitoring includes verification that procedures for preparation, validation, and routine use of the Q-List have been followed.

This expanded surveillance would be both planned (on a monthly basis), and unplanned (e.g., response to INPO SOER's, NRC I&E Information Notices and Circulars, and other relevant nuclear industry reports and information).

2.2.1.5 Attachment C provides a sample Production Order for the purchase of safety-related equipment.

2.2.1.6 The Boston Edison Nuclear Organization recognizes three levels of major classification, "Q", "non-Q" and 1/Q (See Attachment A, NOP8301, Section 5 for further definition). The "Q" designation applies to all safety-related equipment and activities. The PNPS Q-List identifies safety-related SS&C's at a level which does not recognize the classification of piece-parts within listed assemblies. These are generally dealt with on a case-by-case basis, however, the Organization recognizes a level within "Q" of those piece-parts which are not engineered for specific nuclear application and require no vendor-certified qualifications testing. These are designated as Commerical Quality Control Items and are specifically identified in a section of the Q-List. Quality Controls are being established in Specifications for these items.

2.2.2 Addressed above in our response to 2.1.3.

3.1.1 Plant maintenance at Pilgrim is done in accordance with the requirements of Procedure 1.5.3, "Maintenance Requests" and is tracked by the Maintenance Request (MR) form which reflects 1.5.3. This procedure is in accordance with ANSI 18.7 (1976).

The MR process incorporates a series of steps and check-offs prior to beginning maintenance. These steps are to ensure that the necessary disciplines (Operations, Maintenance, Quality Control, and Fire Protection, as necessary), may review the request and designate any steps or procedures necessary to satisfy existing requirements.

The specific issue of post-maintenance testing is determined by both the Maintenance Staff Engineer (MSE) and the Operating Supervisor (OS). The OS determines what tests are required prior to beginning work, for example, testing a redundant system prior to removing its duplicate for maintenance. The parameters for acceptance are contained in Technical Specifications or surveillance procedures.

The OS also determines what tests must be performed before the system can be returned to service. In some cases, where QC has indicated necessary during the MR review, QC must be notified prior to the performance of the test.

After the OS has made his determination, the Watch Engineer reviews the MR and, should the Watch Engineer disagree, the MR is returned to the OS for resolution prior to the start of work.

Post-maintenance testing other than for surveillance is determined by the MSE. This testing is to demonstrate that the maintained item performs in accordance with procedures or vendor information.

We believe this process allows appropriate determinations to be made by those most familiar with plant conditions at the time work will take place. We also believe this process adequately ensures appropriate post-maintenance testing; therefore we plan no further action at this time.

3.1.2 As part of our Performance Improvement Plan, we instituted a Procedure Update Program (PUP) for operations and maintenance procedures. The PUP is a one time effort. After completion of PUP, future revisions to procedures and Vendor Manuals are to be handled by existing organization procedures in an ongoing, timely manner. At this time, the PUP is ongoing, and is expected to be completed by October 31, 1984.

The PUP has been implemented using a systems approach, with work assigned and scheduled by the PNPS Operations Department Management. System procedure update priorities are determined by cognizant operations personnel based on their experience and knowledge of plant systems.

The inputs to this program are listed below:

- o Plant Design Change Information
- o 1.C.6 Independent Valve Verification Requirements

- o Operator Experience Feedback
- o Training Department Feedback
- o INPO Recommendations
- o Modification Management Group Feedback
- o Procedure Classification Changes (Safety-related or Non-safety related determination)
- o Vendor Manual Information

The vendor manual validation process is being incorporated into a Nuclear Organization Procedure. This NOP is in its review cycle and is expected to be emplaced by January 1, 1984.

We believe the PUP, while not specifically initiated in response to Generic Letter 83-28, satisfactorily addresses its concerns.

3.1.3 Surveillance frequencies contained in Pilgrim's Technical Specifications for both the Reactor Protection System and other systems were initially formulated using vendor information and established probability techniques.

As operating experience and new information has developed, we have amended the Technical Specifications, after careful review and with NRC concurrence, to emplace changes which would enhance safety.

We are considering an evaluation of relevant nuclear industry and Pilgrim failure rate data to assess appropriate actions concerning Technical Specification post-maintenance testing requirements.

3.2.3 Addressed above in 3.1.3.

3.2.1 Addressed above in our response to 3.1.1.

3.2.2 Addressed above in our response to 3.1.2.

3.2.3 Addressed above in our response to 3.1.3.

4.5 PNPS performs on-line functional testing of the reactor protection system, including independent testing of the diverse trip features. Initiating circuitry is tested in accordance with appropriate Technical Specification requirements. For this testing, the logic is checked from process parameter input through to the actuating device.

General Electric, through the BWROG, is reviewing the adequacy of existing surveillance and the periodic testing of Backup Scram Valves.

The results of this effort is expected in March, 1984. After reviewing GE's recommendations and results, we shall submit any appropriate actions and completion dates. Based on March, 1984 as our receipt from GE of their findings, we will submit our results in June, 1984.

The results of the BWROG may also indicate a need to change Technical Specifications, and we will assess such recommendations at that time. However, we wish to reinforce our response in 3.1.3 of this letter that our Technical Specifications is a "living" document which has been and continues to be refined by operating experience.

ATTACHMENT A