Wisconsin Electric POWER COMPANY
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November 27, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Dear Mr. Denton:

DOCKETS 50-266 AND 50-301
IMPLEMENTATION OF NUREG-0578
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On November 5, 1979, we received your letter dated October 30, 1979, which discusses implementation schedules and provides clarifications of the NUREG-0578 Short Term Lessons Learned requirements.

Attached herewith is a detailed summary of our commitments and schedules for implementation of these short-term requirements. We note that certain of the material provided in Enclosure 1 to your October 30 letter are modifications of the initial proposals transmitted with your letter of September 13. Also, we understand your Staff did not have time to complete its review of our October 20 response thereto. This reply incorporates the clarifications and other matters requested in your October 30 letter. Based on telephone conversations with your Staff on November 19, and our telegram of November 20, we have also included supplemental information to address and clarify specific items raised by your Staff.

We call your attention to two items where we are not in complete agreement with the Staff position. These are items 2.2.1.b, Shift Technical Advisor, where we propose a duty and call system in which the Technical Advisor would be available in thirty minutes instead of ten minutes; and in the completion of the permanent technical support center where state of Wisconsin certification, building permit and physical construction requirements may make completion by January 1, 1981 uncertain. For further details in respect to these and the other items, we refer you to our October 20 response and to the attached summary.

We trust that the Staff will complete its review of our October 20 initial response regarding implementation of the Lessons Learned items from NUREG-0578. We have made a specific and detailed analysis of the Three Mile Island accident and its relation to our Point Beach facility. Our commitments and schedules for implementation have been carefully and deliberately specified. They represent our best realistic assessment of work required for accomplishment.

In some areas, we have offered comments as to the purpose and need for the Staff recommended action. We believe these comments are pertinent and of sufficient substance to have them considered by the Staff in its continuing review and analysis of lessons learned from Three Mile Island. These additional comments do not indicate a lack of commitment to comply with the actions required or the implementation schedule, but are intended to provide constructive input to this ongoing analysis.

We have also called attention to the major design differences between Point Beach and Three Mile Island as identified in our Task Force Report attached to our October 20, 1979 letter. These differences significantly reduce the possibility of and consequence of a Three Mile Island-type accident at Point Beach and, of course, affect the nature of implementing the lesson learned requirements.

We are confident that the measures and schedules for compliance with NUREG-0578 requirements are satisfactory and realistic. We believe that continued operation of Point Beach without these changes does not present any undue hazard to the health and safety of the public or to the plant employes charged with its safe operation and maintenance.

We would again offer to meet with you and members of your Staff to review any of these items further.

Very truly yours,

Sol Burstein

Executive Vice President

Sol Burstein

Enclosures

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2.1.1 Emergency Power Supply Requirements for the Pressurizer Heaters, Power-Operated Relief Valves and Block Valves, and Pressurizer Level Indicators in PWRs

Pressurizer Heater Power Supply

A. Position

The current Point Beach Nuclear Plant (PBNP) pressurizer heater power supply, as described in detail in the October 20, 1979 submittal meets all of the criteria stated in the Staff position and clarification. Original plant design includes appropriate equipment for this purpose.

B. Discussion

Procedures to specify the proper timing and loading of the pressurizer heaters will be completed in conjunction with the emergency procedure effort underway within the Westinghouse Owners' Group. In establishing these procedures, careful consideration will be given as to which ESF loads may be appropriately shed for a given situation such that the operator has clear criteria to prevent overloading a diesel generator.

C. Schedule

The procedure effort will be completed by January 1, 1980.

Power Supply for Pressurizer Relief and Block Valves and Pressurizer Level Indicators

A. Position

The current PBNP pressurizer relief and block valves and pressurizer level indicators power supplies, as described in the October 20, 1979 submittal and detailed below, meet all of the criteria stated in the Staff position and clarification.

B. Discussion

The attached Figure 2.1.1-1 and Tables 2.1.1-1 through -3 provide the pressurizer relief, solenoid and block valve configuration and power supply distribution for the pressurizer relief and block valves, air compressors, and pressurizer level instrumentation channels, respectively. Both the red and blue instrument buses are vital instrument grade buses. The yellow and white instrument buses are non-vital. The white bus is supplied from the respective unit B02 AC/AC motor-generator set which serves to eliminate short term transients (power spikes) on the white bus.

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B. Discussion (Continued)

Nos. 1 and 2 batteries are shared between units; the inverters are unique to each unit. Also attached is Figure 2.1.1-2 (previously supplied to the NRC with other submittals) which is a simplified electrical distribution diagram for the PBNP and combines Figures 8.2-1 and 8.2-2 of the Final Facility Description and Safety Analysis Report (FFDSAR). Prefix numbers in this figure and the tables refer to unit designation (e.g., 1(2)-PCV-431C indicates the same configuration applies to the valve for both Units 1 and 2).

The power operated relief valves, 1(2)PCV-430 and 1(2)PCV-431C, are air operated valves which close on loss of air pressure to the valve actuator diaphragm. Instrument air system pressure is applied to the power operated relief valve actuator through solenoid valves 1(2)SOV-430 and 1(2)SOV-431C, respectively. The solenoid valve is actuated by two pressurizer pressure bistable/relay combinations. Both bistables must actuate to operate the solenoid valve allowing the relief valve to open. If a pressurizer pressure channel fails low (Channel 429, 430, 431, or 449), the power operated relief valve associated with that channel cannot open.

Instrument air is supplied through containment penetrations from either of two instrument air compressors, K2A and K2B. This is backed up by the service air system which is supplied by either of two service air compressors (K3A and K3B). Instrument air pressure at 90 psig is normally supplied by the instrument air compressors. If the instrument air system falls to 80 psig, service air is then supplied to the instrument air system. This is sufficient pressure to operate air operated valves supplied by the instrument air system. The instrument air compressors are stripped on a safety injection coincident with a loss of AC. They must be manually restarted.

The instrument air headers inside containment are isolated when a containment isolation signal is initiated for each unit. In the case of automatic safety injection, this isolation remains until safety injection is reset and the instrument air system is unisolated. The instrument air system may be unisolated before safety injection is reset as long as the control switch for the instrument air containment isolation valve is manually held in the "open" position. A backup pressurized nitrogen gas bottle is provided to supply nitrogen gas to operate the power operated relief valves in the event of instrument air system pressure loss when low temperature-overpressure protection is required during plant shutdown.

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C. Schedule

No additional action is required.

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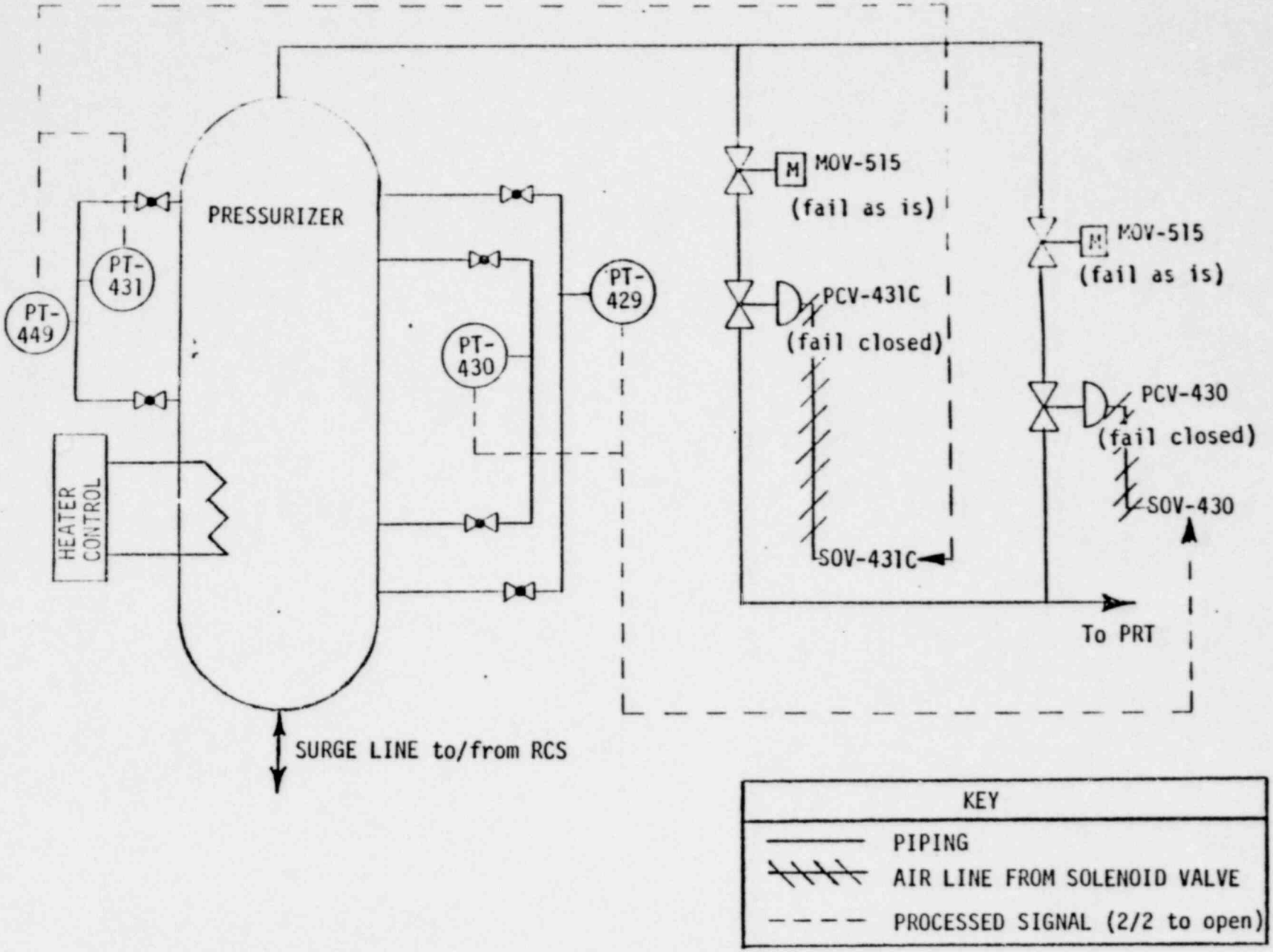

FIGURE 2.1.1-1 PBNP PRESSURIZER POR AND BLOCK VALVE SCHEMATIC

Figure 2.1.1-2

ELECTRICAL SYSTEM

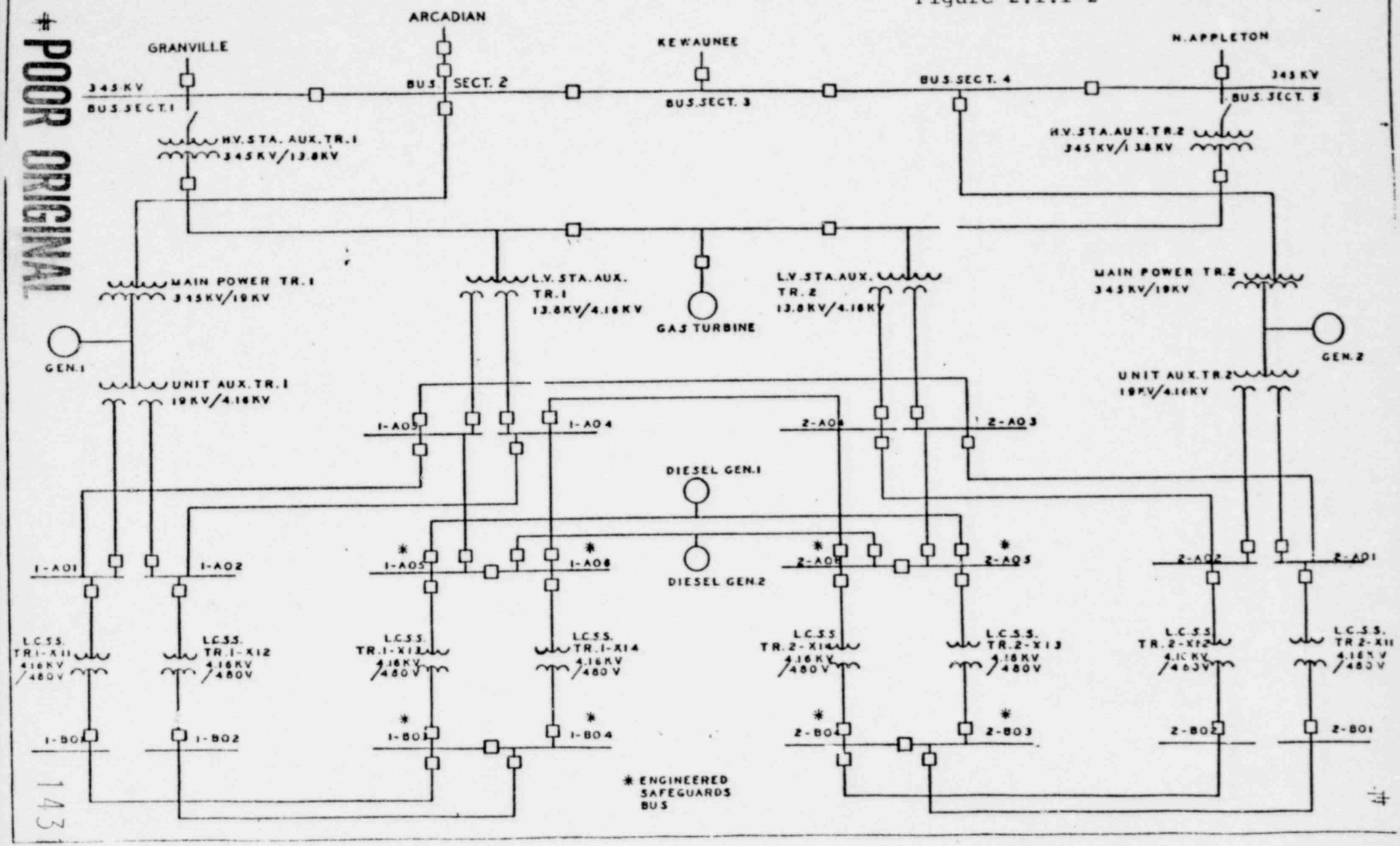

TABLE 2.1.1-1

PBNP PRESSURIZER POWER OPERATED RELIEF AND BLOCK VALVE POWER SUPPLIES

ITEM	POWER SUPPLY	VALVE APPLICABILITY					
		1(2)PCV		1(2)MOV		1(2)SO3	
Bistable/Relay Combinations		430	431C	515	516	430	431C
Channel 429	No. 1 battery/inverter for the unit, red instrument bus AC	X	Backup to Ch.431				
Channel 430	No. 1 battery/inverter from Unit 2, red instrument bus AC	Unit 1					
	No. 2 battery/inverter from Unit 1, blue instrument bus AC	Unit 2					
Channel 431	No. 2 battery/inverter for the unit, blue instrument bus AC		X				
Channel 449	1(2) BO4 ESF bus, yellow instrument bus AC	Backup to unit's Ch. 430	X				
Motive Power							
	1(2)BO3 ESF bus			X			
	1(2)BO4 ESF bus				X		
Control Power							
	1(2)BO3 ESF bus			X			
	1(2)BO4 ESF bus				X		
125V DC							
	No. 1 battery					X	
	No. 2 battery						X

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TABLE 2.1.1-2

PBNP AIR COMPRESSOR POWER SUPPLIES

COMPRESSOR	AIR SUPPLY SYSTEM	ENGINEERED SAFEGUARDS BUS			
		1B03	2B03	1B04	2B04
K2A	Instrument	X			
K2B	Instrument		X		
K3A	Service			X	
K3B	Service				X

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TABLE 2.1.1-3

PBNP PRESSURIZER LEVEL INSTRUMENTATION
POWER SUPPLIES

PRESSURE LEVEL CHANNEL	POWER SUPPLY
1(2)L426	No. 1 battery/inverter for the unit, red instrument bus
1(2)L427	1(2)B02 AC/AC motor-generator set nonsafeguards bus, white instrument bus
1(2)L428	No. 2 battery/inverter for the unit, blue instrument bus
1(2)L433 (Cold Calibration)	No. 1 battery/inverter for the unit, red instrument bus

1431 330

2.1.2 Performance Testing for BWR and PWR Relief and Safety Valves

A. Position

As stated in the October 20, 1979 submittal, the Licensee is a member of the Westinghouse Owners' Group which is working in conjunction with the other PWR owners and EPRI to develop an appropriate industry testing program and schedule to demonstrate valve operability under various flow conditions. We will continue our participation in this program and will provide technical and financial support as required.

B. Discussion

We believe that our previous comments on the need for such testing are pertinent and of sufficient substance to merit NRC Staff consideration in its ongoing review of lessons learned from Three Mile Island.

The current design basis and Code specifications in this area require redundant Code safety valves. nonisolatable, on any pressurized system as a means of overpressure protection. In addition, the discharge from the Code safety valves must be unrestricted in its capability to ultimately relieve to the atmosphere.

To reduce the number of challenges to the Code safety valves because they may not reclose and cannot be isolated, the design of Point Beach and other PWR primary coolant systems also includes one or more power operated relief valves (PORV). Automatic control systems are provided to operate the PORVs and provide relief of RCS pressure at a setpoint below that of the Code safety valves. Manual actuation is also provided. It is our understanding that, during power operation, this design has eliminated all challenges to the Code safety valves and that, in fact, no Code safety valve has yet been required to open on a Westinghouse PWR plant. Additionally, the basic design of the inverted U-tube steam generators with a large secondary side water inventory (thirty minutes or greater to dryout without AFW) and safety grade auxiliary feedwater system provide substantial post-accident heat removal capability. This serves to limit the primary side heatup and pressure transient and severely restricts even the challenges to the PORVs.

Since the PORVs are expected to open, the design also provides block valves in series to insure isolation of the RCS should the PORVs fail open. The operation of the PORV and block valve is ensured by separation of power supplies and actuation logic as described in

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B. Discussion (Continued)

item 2.1.1. This design is a proven, accepted practice in pressurized systems, both nuclear and non-nuclear, and meets all Code and NRC licensing requirements.

The transient and accident analyses specifically exclude both the protection (relief) and control (relief-isolation) capabilities of the PORV and block valves. Instead, the overpressure conditions analyzed must use the Code safety valve setpoints as the upper bounds of the calculations. Conversely, the LOCA analyses must consider a range of breaks which cover an assumed PORV or safety valve opening. Thus, the function of reclosure is strictly an operational benefit without unanalyzed safety concerns. It should be noted that while the pressurizer relief tank (PRT) provides for the suppression of PORV or safety valve releases and eventual unrestricted release to containment via the tank rupture disk, containment pressure calculations must assume direct release to the containment for maximizing pressure. Piping system integrity downstream of the safety valves is, thus, not necessary.

In light of these Code requirements, licensing and design bases, and analyses, it is unclear as to the objectives of the committed testing program required by the NRC. We know of no plan for a redetermination as to whether the Code safety valves are to provide control in addition to overpressure protection. Studies of PORV-block valve configurations, piping, and supports are not safety concerns and should rather be part of long term operational studies. We would be pleased to participate in any review of this matter.

C. Schedule

The Westinghouse PWR Owners' Group or industry (EPRI-NSAC) program description will be submitted by January 1, 1980, with a testing program to follow.

1431 332

2.1.3.a Direct Indication of Power-Operated Relief Valves and Safety Valve Position for PWRs and BWRs

A. Position

The power operated relief valve indication, as described in the October 20, 1979 submittal, meets the NRC requirements for direct indication. The stem-mounted limit switches from which the indication is derived are safety-grade, seismically qualified, and environmentally qualifed as discussed below. Direct indication of position for the Code safety valves will be provided.

B. Discussion

The basic operational and safety requirement is that the operator be able to determine the condition of the plant and take appropriate action in response to an accident condition. The Licensee continues to be committed to providing sufficient and necessary instrumentation in the control room such that the operator has this accident assessment capability.

The present RCS instrumentation provides the operator with redundant, safety grade instrumentation which permits the determination of the existence of a LOCA regardless of whether the LOCA is due to an open Code safety or relief valve or valves. Revised emergency procedures have been developed and the operations of the safety systems are based on plant parameters which are independent from any valve position indication. Instrumentation is provided, as detailed in the October 20, 1979 submittal, so that the position of individual power operated relief valves and Code safety valves can be determined. Even if new redundant, seismically and environmentally qualified (safety grade) valve position indications were to be provided, the availability of this information to the operator would not change any action(s) to be taken during the course of an accident.

The current PORVs have direct, positive indication in the control room from seismically qualified, stem-mounted limit switches for both units. Replacement of these switches will upgrade them so that they are also environmentally qualified. This upgrading was completed during the recent November 1979 refueling outage for Unit 1. It is expected to be completed during the Spring 1980 refueling outage for Unit 2 scheduled to start in March. There is also an alarm on high pressure at the setpoint at which the valves will open.

1431 533

C. Schedule

The replacement of limit switches on the Unit 2 power operated relief valves will be completed in April of 1980. Implementation of direct position indication for Code safety valves will be completed by January 1, 1980, depending upon equipment availability. To accomplish this schedule for Code safety valves, acoustic devices to indicate valve operation or leakage are being procured. It is possible that unit shutdowns will be required to permit installation.

1431 334

2.1.3.b Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs

I. Procedures and Description of Existing Instrumentation

Analyses and procedure guidelines will be submitted to the Bulletins and Orders Task Force through the Westinghouse Owners' Group as detailed in the October 20, 1979 submittal.

II. Subcooling Meter

A. Position

The installation of RCS saturation condition monitoring and alarm will be provided by the Licensee as stated in the October 20, 1979 submittal.

B. Discussion

1. A two step approach was previously described utilizing the plant process computer initially and a separate system later. The initial process computer system implementation has been completed and uses multiple incore thermocouples for temperature input and can use either the redundant safety grade pressurizer pressure signals or the wide range pressure signal as input for the pressure. This system will be upgraded and include extended temperature range for the incore thermocouples. Display of the subcooling meter information will be provided on operator demand and an alarm will sound should less than 50°F subcooling occur. The current system is described in the attached tabulation.

2. Implementation of the separate system for monitoring subcooling as an integral part of a larger diagnostic system is currently under review, with the aid of a consultant.

C. Schedule

1. The interim computerized system described above has already been installed.

2. Implementation of a separate system will proceed according to the schedule presented in our letter of October 20, 1979.

1431 335

III. Design and Installation of New Instrumentation

A. Position

1. The Westinghouse Owners' Group program for analyses and procedures to address inadequate core cooling will provide the basis for any new instrumentation.

2. Although the Owners' Group analysis has not yet been completed, we will provide reactor vessel level instrumentation as requested by NRC.

B. Discussion

The request for vessel level indication to cover the full range from normal water level to the bottom of the core will be included. The additional features requested by the NRC are being included in the evaluation. Means of reactor vessel level indication are being considered together with reactor vessel head vent arrangements.

C. Schedule

1. The schedule for the analyses and procedures by the Owners' Group is given in item 2.1.9 of the October 20, 1979 submittal and concludes by the end of the first quarter of 1980, as agreed with the Bulletins and Orders Task Force.

2. We will provide the conceptual design of reactor vessel level instrumentation by January 1, 1980. Detailed design will be provided as soon as possible thereafter.

3. A description of any other new instrumentation, functional design requirements and schedule for installation will be provided after completion of the Owners' Group analysis. NRC proposal review is required before implementation per the NRC clarification schedule of October 30, 1979.

4. Instrumentation requiring modification to the RCS and additional wiring to be pulled into containment would be implemented at the earliest possible scheduled refueling outage for the unit. Present schedule for Unit 2 is Spring 1980 and for Unit 1 is Fall 1980. Implementation of vessel level indication will be completed by January 1, 1981, assuming valve and instrument delivery schedules.

1431 336

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	T, Tsat, Press as selected
Display Type (Analog, Digital, ERT)	Digital
Continuous or on Demand	On Demand
Single or Redundant Display	Single
Location of Display	Control Room Reactor Operators Console
Alarms (include setpoints)	Alarm typewriter, 50°F setpoint
Overall uncertainty (°F, PSI)	(±5°F, ±15 psig)
Range of Display	0 to 9999.9°F
Qualifications (seismic, environmental, IEEE323)	Original Plant Design

Calculator

Type (process computer, dedicated digital, or analog calculator)	PRODAC 250 Computer
If process computer is used specify availability (% of time)	100% (Monitoring Computer Only)
Single or redundant calculators	Single
Selection Logic (highest T., lowest press)	Single Pressure Channel - Wide Range Average Incore Thermocouple
Qualifications (seismic, environmental, IEEE323)	Original Plant Design
Calculational Technique (Steam Tables, Functional Fit, Ranges)	Steam Tables

Input

Temperature (RTDs or T/Cs)	Incore Thermocouples
Temperature (number of sensors and locations)	All Available Incore Thermocouples - Max. 39
Range of Temperature Sensors	0 - 2000°F

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Uncertainty* of Temperature Sensors (°F at 1)	±5°F @ 620°F (175% x temp)
Qualifications (Seismic, Environmental, IEEE323)	Original Plant Design
Pressure (specify instrument used)	Foxboro Model 611 Pressure Emitter
Pressure (number of sensors and locations)	1 Sensor Reactor Coolant Hot Leg
Range of Pressure Sensors	0 - 3000 psig
Uncertainty* of pressure sensors (PSI at 1)	0.5% of full scale (±15 psig)
Qualifications (Seismic, Environmental, IEEE323)	Original Plant Design

Backup Capability

Availability of Temp and Press	Located in Control Room - Wide Range Pressure Incore Thermocouple Readout
Availability of Steam Tables, etc.	Located in Control Room
Training of Operators	Yes
Procedures	EOP 1, 2 and 3 revisions include subcooling implications

*Uncertainties must address conditions of forced flow and natural circulation

1431 338

2.1.4 Containment Isolation Provisions for PWRs and BWRs

A. Position

All of the NRC position and clarification items are satisfied by the current Point Beach Nuclear Plant containment isolation system actuation and reset features.

B. Discussion

As stated in the previous submittal on October 20, 1979, the required information pertaining to the containment isolation system utilized at the Point Beach Nuclear Plant has been described in detail to the NRC Staff in Wisconsin Electric Power Company's response to Item 9 of Bulletin 79-06A dated April 27, 1979. That submittal demonstrates that all of the NRC position and clarification items are satisfied by the current Point Beach Nuclear Plant containment isolation system actuation and reset features. Specifically, resetting of the containment isolation signal will not automatically result in the reopening of any containment isolation valves. Deliberate operator action is required to reopen any isolated valves.

C. Schedule

The evaluations committed to be provided are currently in progress and will be completed by January 1, 1980. These include the identification of essential and nonessential systems and system isolation status (normally isolated during operation, isolated on containment isolation, and not isolated).

1431 339

2.1.5.a) Post-Accident Hydrogen Control Systems for PWR and
2.1.5.c) BWR Containments

A. Position

Items 2.1.5.a and 2.1.5.c do not apply to Point Beach Nuclear Plant as related to hydrogen recombiners since recombiners are not a part of the design or licensing basis for the plant.

C. Schedule

The purge system design review calculation committed to in item 2.1.5.a in the October 20, 1979 submittal has been made and verifies the original design with respect to the flow capacity of the purge system. This completes the required action for these items.

1431 340

2.1.6.a Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems) for PWRs and BWRs

A. Position

We will implement an immediate and continuing leak identification and reduction program and provide a summary description consistent with the NRC clarification.

B. Discussion

All identified systems will be included in this program. The methods, criteria, and provisions for testing of gaseous piping systems are being evaluated. A consultant has been retained to perform this task consistent with the committed schedule.

C. Schedule

The identification of systems and summary description will be provided by January 1, 1980. Initial testing of liquid and gaseous systems will be performed by January 1, 1980. The complet~~ion~~ program will be implemented by January 1, 1981.

1431 341

2.1.6.b Plant Shielding Review

A. Position

We will perform a shielding analysis for Point Beach Nuclear Plant and provide shielding or procedural modification where necessary.

B. Discussion

1. The postulated source term is extremely conservative and appears to us to be unrealistic. We will, however, perform the analyses assuming this non-mechanistic source term.

2. The "$\leq$15 mR/hr" criterion, we believe, is not intended for cases where continuous occupancy does not involve the same person for 12 hrs/day for 30 days. However, the analysis will identify those cases, if any, where credit is taken for occupancy by different persons for areas otherwise assumed to require continuous occupancy. Although not apparent from the statement of the criterion, we are making appropriate allowance for radioactive decay, e.g., an average dose rate of 15 mRem/hr when averaged over an assumed period of 30 days following an accident.

C. Schedule.

1. Initial analysis will be complete by January 1, 1980, but detailed engineering design for required modifications may not be complete before July 1, 1980.

2. System identification has already begun. A consultant has been retained, and calculational work is in progress. Steps of the analysis are described in our submittal of October 20, 1979.

3. If shielding modifications are to be made, every effort will be made to complete such modifications by January 1, 1981. The nature and extent of any modification is, of course, presently unknown. We will keep the NRC Staff informed of progress in this area.

2.1.7.a Automatic Initiation of the Auxiliary Feedwater System for PWRs

A. Position

A separate response prepared in reply to the September 21, 1979 letter from Mr. D. Eisenhut to Mr. Sol Burstein entitled, "NRC Requirements for Auxiliary Feedwater Systems at Point Beach" satisfies this item.

C. Schedule

The submittal, dated October 29, 1979, from Mr. Sol Burstein to Mr. D. Eisenhut addressed all of the requirements not covered in the October 20, 1979 submittal and completes this item.

1431 343

2.1.7.b Auxiliary Feedwater Flow Indication to Steam Generators for PWRs

A. Position

The flow indication instrumentation and schedule remain as stated in the October 20, 1979 submittal.

C. Schedule

Control grade indication has been provided. It is anticipated that based on availability of safety grade equipment, this item will be completed by January 1, 1981.

1431 344

2.1.8.a Post-Accident Sampling

A. Position

We agree to perform a sampling analysis and to make plant modifications where required.

B. Discussion

1. The postulated source term, e.g., 100% of noble gases in liquids and 100% of noble gases in gases, is extremely conservative and, we believe, is unrealistic.

2. Chloride analysis capability is neither appropriate nor necessary for determination of plant status or for plant-follow during post-accident conditions for Point Beach Nuclear Plant, which is located on a body of freshwater. However, we will provide the ability to determine chloride as NRC requires.

3. We note that pH and hydrogen determinations are new additions in the NRC letter of October 30, 1979. We will incorporate these into our analysis.

4. The specific one-hour requirement for both sampling and analytical determinations may be difficult to meet. We are making every effort to meet these criteria; we will advise the NRC if any difficulties arise in this area.

5. We believe that nearby offsite analysis (<5 miles) should be entirely satisfactory in lieu of "onsite analysis" in the event normal plant laboratory facilities become uninhabitable.

6. A containment under negative pressure has no potential for leaking and, therefore, has no impact on public health and safety. Nonetheless, we will provide the capability to sample the containment under both positive and negative pressure conditions.

7. We have not been able to verify that a "factor of 2 error" can be met in a one hour analysis for a high level sample. We will provide state of the art uncertainty if the factor of 2 proves unobtainable due to dilution errors, dead time, gain shifts, or other technical constraints.

1431 545

C. Schedule

1. This analysis will be conducted along with that for item 2.1.6.b. As described in the response to that item, initial analysis will be completed by January 1, 1980.

2. A consultant has been retained for this work, and calculational work is in progress.

3. If sampling modifications are required at the plant, work will be scheduled to be completed by January 1, 1981. We will keep the NRC Staff informed of progress in this area.

1431 346

2.1.8.b High Range Radiation Monitors

A. Position

We will provide increased range radiation monitoring instrumentation to the extent of commercial availability.

B. Discussion

1. As described in our letter of October 20, 1979, and as further discussed in the Wisconsin Electric Power Company Three Mile Island Accident Review Task Force Report transmitted therewith, we believe in-containment monitors are not necessary, since there is no purpose for such information from the viewpoint of either public health and safety or operation and recovery. In-containment monitors are of questionable reliability and accuracy. Containment sampling is believed to be far more practical and reliable for these purposes. However, we will provide these devices as directed by NRC.

2. We believe that nearby offsite analysis (<5 miles) should be entirely satisfactory in lieu of "onsite analysis" in the event normal plant laboratory facilities become uninhabitable.

3. We note that readouts of the high range effluent monitors are required in both the technical support center and the emergency operations center. The NRC has only described the concept of an offsite emergency operations center and that it be provided with appropriate communications and facilities for assisting agencies. We are unaware of further requirements, except for the effluent monitor readout. There may be other readouts appropriate to the technical support center or the offsite emergency operations center. We will seek clarification of these requirements. However, we will meet the requirements prevailing at the time of the development or construction of the permanent offsite emergency operations center.

C. Schedule

1. The analysis for interim procedures regarding high range effluent determination will be performed along with that for 2.1.6.b and 2.1.8.a. This initial analysis is to be completed by January 1, 1980. A consultant has been retained and analytical work is in progress.

C. Schedule (Continued)

2. We expect to install permanent effluent instrumentation by January 1, 1981, subject to equipment deliveries.

3. The required in-containment monitors will be ordered as soon as specification and procurement activity can be completed, with intent of meeting the January 1, 1981 installation date.

4. Because of the limited number of suppliers and the relatively large number of nuclear plants, equipment deliveries may be extended which could affect installation dates. We will proceed on an expeditious schedule and will keep NRC informed of our progress.

1431 548

2.1.8.c Improved In-Plant Iodine Instrumentation under Accident Conditions

A. Position

We will reexamine iodine sampling procedures at Point Beach Nuclear Plant as described in our letter of October 20, 1979.

C. Schedule

1. As described in our letter of October 20, 1979, an agreement between the Kewaunee Nuclear Plant and Point Beach Nuclear Plant is expected to be implemented by January 1, 1980, to provide backup analytical laboratory services for both facilities.

2. Review of existing plant methods and procedures is underway and will be completed by January 1, 1980.

1431 349

2.1.9 Transient and Accident Analysis

A. Position

Our position, as detailed in the October 20, 1979 submittal, meets all of the requirements as discussed in NUREG-0578. The Westinghouse Owners' Group and the Bulletins and Orders Task Force are establishing the scope and schedule for additional analyses. We support these efforts as members of the Owners' Group.

C. Schedule

We will meet the early 1980 completion commitment through the Owners' Group sponsored analyses.

1431 550

2.1.9 Containment Pressure, Water Level and Hydrogen

A. Position

The commitment to provide the required instrumentation, as stated in the October 20, 1979 submittal, remains unchanged.

B. Discussion

Relative to the implementation of containment water level, the equivalent capacity of the wide range level instrument for Point Beach Nuclear Plant will be in excess of 350,000 gallons. The sources are identified as follows:

Source	Maximum Volume (Gallons)
Refueling Water Storage Tank	285,000
Boric Acid Storage	5,000
Accumulators (2)	17,000
Reactor Coolant System (above the nozzles on the vessel)	37,000
Containment Spray Additive Tank	3,000
TOTAL	347,000 gallons

Since the sump is the lower portion of the containment building proper, only a wide range instrument is presently contemplated. The current level indication is discreet at 3', 5', 7', and 9' which exceeds the range of interest (7 feet ≈350,000 gallons). By limiting the initial range setting to 7' for this instrument and allowing upward recalibration following an accident, maximum accuracy will be maintained over the range of most interest.

C. Schedule

A consultant has been retained to assist in meeting the scheduled date of January 1, 1981, for completion.

1431 351

2.1.9 Reactor Coolant System Venting

A. Position

Reactor coolant system venting capability beyond that available with the current penetrations to the Point Beach Nuclear Plant primary system will be provided. Implementation of reactor vessel venting modifications to the plant will occur after the required proposal review by the NRC.

B. Discussion

The Wisconsin Electric Three Mile Island Accident Review Task Force discusses Reactor Coolant System Venting at Section 5.2.1. The reactor vessel venting arrangement is being developed along with vessel level indication provisions and will consider venting to the RCS vent tank or to containment or both.

C. Schedule

The conceptual design for a RCS venting system will be submitted to the NRC by January 1, 1980.

January 1, 1981 will be the target date for completion of required plant modifications consistent with procurement of equipment. Plant shutdowns will, of course, be required for any vessel head penetration or vessel instrument modifications.

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2.2.1.a Shift Supervisor's Responsibilities

A. Position

As stated in the October 20, 1979 submittal, the required review of the responsibilities and authority of the Shift Supervisor will be conducted.

B. Discussion

The review, while emphasizing the aspects of Item 2 of the NRC position, has not done this to the exclusion of other items of the position. This is clearly stated in the submittal for Items 3 (training programs) and 4 (administrative duties). Relative to Item 1, the highest level of corporate management at Wisconsin Electric Power Company which is responsible for defining and assigning the Shift Supervisor duties is the Manager, Nuclear Operations, Point Beach Nuclear Plant. We believe that this level of management has both the required direct knowledge and credibility to adequately inform the Shift Supervisors of their responsibilities and duties and the capability to monitor the resulting performance of these personnel. The safe operation of the plant under all conditions has long been the Wisconsin Electric Power Company's primary policy as established by management and carried out by all operating personnel. This is evidenced by the plant's record and the high standards of performance which have been maintained during its entire operating history. The existing pre-TMI plant policies and procedures have resulted in periodic reviews of the authority and responsibilities of key plant personnel, including Shift Supervisors, and will continue to do so. The NRC is appraised of these changes as previously stated.

We believe that these policies, procedures, and performance warrant consideration by the NRC as meeting the intent and implementation directives of Item 1 and its clarification. We propose to continue Point Beach Nuclear Plant operations with this policy. We will, in addition, issue a special directive, signed by the Executive Vice President responsible, which clearly reaffirms the policy, recognizes specific review responsibilities, and reiterates the safety objective.

C. Schedule

The review, required revisions, and safety directive will be completed by January 1, 1980.

2.2.1.b Shift Technical Advisor

A. Position

The implementation of this item both in intent and schedule remain unchanged from the October 20, 1979 submittal.

B. Discussion

Justification was provided with that submittal for the stated position. The increased commitments for additional analyses, procedures, training, and plant instrumentation will further enhance the adequacy of the Duty and Call Technical Advisor concept. In fact, the increase in control room personnel capabilities, instrumentation, and control functions provides the technical capabilities in operational personnel and significantly reduces the need for such a position. We request the NRC review our Duty and Call Technical Advisor proposed alternative to a shift position.

C. Schedule

Implementation of the Duty and Call Technical Advisor position remains as stated in the October 20, 1979 submittal. Personnel will be assigned by January 1, 1980 and meet the short-term qualifications with full qualification to be achieved by January 1, 1981.

2.2.1.c Shift and Relief Turnover Procedures

A. Position

The shift and relief turnover procedure revisions remain as stated in our submittal of October 20, 1979.

C. Schedule

This item will be completed by January 1, 1980.

2.2.2.a Control Room Access

A. Position

We currently limit and will continue to limit control room access at Point Beach Nuclear Plant. The recognition by NRC of the desirability of such limitation is commendable.

B. Discussion

Current policy, as described in our letter of October 20, 1979, limits NRC access to the control room to one individual. It is our understanding that the NRC Staff does not agree with this access control but has not specified the number of NRC personnel for which access is required. As soon as NRC advises us of the number of NRC personnel for whom NRC anticipates access, we will revise our policy accordingly.

C. Schedule

1. Policy restricting access to the control room has been formalized by special memorandum of October 8, 1979, as described in our letter of October 20, 1979.

2. Revisions to this policy will be issued immediately upon receipt of NRC instructions regarding number of NRC personnel to be granted access.

2.2.2.b On-Site Technical Support Center

A. Position

We will establish an interim and a permanent Technical Support Center.

B. Discussion

Designation of the Point Beach control room as a backup to the interim Technical Support Center may not be appropriate, particularly in view of its size and configuration. We are planning to designate our reactor engineering office and computer room above the control room as the interim alternate.

C. Schedule

1. Establishment of a permanent Technical Support Center will require construction. Such construction may require prior Public Service Commission of Wisconsin authorization and certain building and other permits. We are proceeding as rapidly as possible on this item but we believe that a completion date of January 1, 1981, may be unrealistic.

2. Also, we are consolidating a number of separate efforts to achieve a coordinated and well balanced enhancement of our facilities. These efforts include the provision of the Technical Support Center, the offsite Emergency Operations Center, additional space for our training program, and permanent office space for resident NRC inspectors. At the same time, we are attempting to provide a coordinated approach to all the "short-term lessons learned" instrumentation implementation. These plans are in the conceptual stage at present.

3. Conceptual planning for a permanent Technical Support Center has already begun. It is anticipated that detailed engineering will begin by January 1, 1980, and regulatory and construction requirement will help define completion schedules shortly thereafter.

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2.2.2.c Operational Support Center

A. Position

An interim and a permanent Operational Support Center will be designated.

B. Discussion

Refer to October 20, 1979 letter for further details.

C. Schedule

Permanent designation of an Operational Support Center is related to other emergency centers. See response to 2.2.2.b.