

BALTIMORE GAS AND ELECTRIC COMPANY

GAS AND ELECTRIC BUILDING
BALTIMORE, MARYLAND 21203

November 20, 1979

ARTHUR E. LUNDAVALL, JR.
VICE PRESIDENT
SUPPLY

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attn: Mr. Darrel G. Eisenhut, Acting Director
Division of Operating Reactors

Subject: Calvert Cliffs Nuclear Power Plant
Units Nos. 1 & 2, Dockets Nos. 50-317 & 50-318
Follow-up Actions Resulting from TMI-2 Incident
(Lessons Learned Short Term)

Gentlemen:

This letter provides additional description and explanation as required by H. R. Denton's letter of 10/30/79. Enclosure 1 is a revised version of our previous submittal in our letter of 10/19/79. Revisions are the result of continued design, more detailed scoping and scheduling, and review of the "clarifications" in your 10/30/79 letter.

We are continuing to make every effort to implement all requirements as they apply to our plant, within your time limits. However, several of the items required by 1/80 cannot be completed by that time. These items are:

NUREG	ITEM	EARLIEST INSTALLATION		OUTAGE REQ'D
		START	COMPL	
2.1.3a	Valve Pos'n Indic.	5/01/80	5/26/80	2 wks.
2.1.3b	Subcooled Monitor	5/12/80	6/09/80	3 wks.
2.1.4	CIS Reset	5/01/80	5/26/80	4 wks.
2.1.7a	Aux. Feed Auto Start	12/26/79	1/14/80	None

We have developed detailed schedules to track and expedite this implementation, and we are pursuing every avenue available for improvement of the schedules. In all of these cases, the equipment specification/purchase/delivery sequence is on critical path. For each of these requirements, existing measures for accomplishing these functions have been reviewed, and procedures have been improved where necessary or interim methods devised to minimize any perceived risk during the time required to complete the permanent installation; detailed discussions of these interim measures are included in Enclosure 1 for each item.

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For the 1/81 items, we have indicated in the enclosure that we intend to be complete by the required date. We are approaching the expediting of these jobs from every conceivable angle and will do all that is possible to meet this date. It must be realized, however, that this commitment is based on optimistic schedules, which in turn are based on preliminary and sketchy information. We, therefore, must qualify our commitment with these uncertainties, and wish all involved to be aware of this situation.

Normal refueling outages in 1980 would be in spring or fall for Unit 1 and in fall for Unit 2. We now are looking toward a late spring/early summer outage for Unit 2, the purpose of which will be to install the remaining Lessons Learned 1/80 items listed above. If our schedules are maintained, this would point to an outage of three to four weeks for Unit 2, beginning 5/12/80, though the exact shutdown date and duration will depend on the jobs as they develop. This would be followed by a longer outage for Unit 1, which we propose to begin about 9/1/80, to install the Lessons Learned items and to refuel. An additional outage will then be required for Unit 2 for refueling and installation of Lessons Learned 1/81 requirements.

We have arrived at this plan after careful consideration of all known factors. Our conclusion is based on a continuing and successful all-out effort to install required items in the shortest possible time, and also takes consideration of the following general points:

- (1) There is overwhelming justification for not shutting down both units at the same time; grid stability and reliability would be adversely affected; manpower available for installation is limited and can not be used as effectively or as efficiently, increasing total unit-days outage time, and increasing total exposure; and the economic cost to our customers would be substantial (see 2d below).
- (2) Minimizing the number of outages (by, for example, combining refueling and Lessons Learned installations in the same outage) is justified by the following:
 - (a) Minimization of plant transients, which are inherently less safe than steady state conditions.
 - (b) Limiting thermal cycling of the plant and components.
 - (c) Conservation of resources, specifically oil, which is our replacement fuel when a nuclear unit is not available. Current allocations are inadequate even with Calvert Cliffs available, exemplified by the fact that BG&E used its November allocation by the 19th of the month.
 - (d) Extremely high cost to our customers. The additional four-week outage now planned for May will cost customers in excess of \$11,000,000.

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- (3) An extended shutdown during the period 7/15 through 9/1 should be avoided if possible because of tremendous heat in containment, which results in increased health risk to workers, and reduced reliability and productivity.
- (4) Interim measures being implemented to accomplish the functions of the pending Lessons Learned requirements have been carefully reviewed, and provide confidence that safety is not in any way being sacrificed or assigned secondary importance by the delay in permanent implementation.

We are making a sincere and diligent effort to comply with your requirements, while at the same time considering and weighing all ramifications. We would be happy to meet with you to discuss any concerns or questions you may have.

Very truly yours,



Enclosures

cc: J. A. Biddison, Esquire
G. F. Trowbridge, Esquire
Mr. E. L. Conner, Jr.

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Enclosure 1
Revised Response
to
NUREG 0578

2.1.1 Emergency Power Supply Requirements

Pressurizer proportional heaters are currently supplied from an emergency power supply (diesel generators). Two banks of pressurizer back-up heaters will be fed from the 480 V engineered safety features system load centers: One 300 kW bank from bus 11B and the second from bus 14B. This will provide the redundant capability as described in FSAR section 8.3.3. The load center breakers supplying these banks as well as the proportional heaters are tripped on an undervoltage signal after a loss of offsite power. These will be closed manually when required after diesel generator load sequencing is complete. Time required to close these breakers from the switchgear rooms is consistent with requirements for initiation and maintenance of natural circulation.

One bank of back up heaters along with the proportional heaters provide 450 kW capacity, which preliminary analysis indicates is sufficient to establish and maintain natural circulation. Installation of the changes and implementation of procedures should be complete by 1/1/80.

The motive components of the power operated relief valves (PORV's) and the PORV block valves are supplied from safety related 480 V motor control centers which have a diesel backup. The control components of the PORV's and PORV block valves are supplied from either the same safety related 480 V motor control centers or from safety related 125 V DC battery buses.

Two of the pressurizer level instruments for each unit are powered from the vital dc buses and the third is powered from offsite AC power with diesel backup.

We will review the motive and control power interfaces with the emergency buses to determine if these interfaces satisfy safety-grade requirements. This review and steps necessary to upgrade components will be completed by 1/1/80.

2.1.2 Relief and Safety Valve Testing

A program for testing power operated relief valves (PORVs) and safety valves (SVs) used for primary system pressure control under design bases operating conditions is being developed by the CE Owners Group. This program includes definition of test conditions and qualification requirements for all specified valves in operating reactors designed by Combustion Engineering. This program will be included in the generic efforts being undertaken by the industry (through the Electric Power Research Institute, EPRI and the Nuclear Safety Analysis Center (NSAC). It is planned to have a description of the program and a projected schedule available for discussions with the NRC staff to establish generic resolutions no later than January 1, 1980. Calvert Cliffs will comply with the schedule for completion of the test program which is agreed to during these discussions.

2.1.3a Direct Valve Position Indication

The direct position indication of the two PORV's and the SV's per unit will be accomplished with an acoustic monitoring system. The system will be safety grade without redundant sensors and will be alarmed with indication in the main control room. By 1/1/80, the equipment for both units will have been ordered, and the final engineering will be completed for one unit and 90% complete for the other unit. Equipment delivery is expected in April.

Presently there are various indicators to help determine the status of the valves, such as power indication to the PORV solenoids, and quench tank pressure, temperature, and level indication, all of which is located on the front of the same control panel in the control room. In addition, there are thermocouples located in the two discharge lines which indicate in the control room. These indications provide adequate means of determining valve position during the period of time required to receive equipment and complete installation. Plant procedures previously in effect to provide guidance to operators in use of these indications to determine valve status were reviewed in detail in light of the TMI incident, and were upgraded to include any necessary improvements. Operators have been re-trained in the use of these procedures, and because of this training and the TMI experience are acutely aware of all existing methods of determining valve position.

2.1.3b Instrumentation for Detection of Inadequate Core Cooling

Procedures and Additional Instrumentation

To the extent possible within the framework of existing analyses, procedures have been upgraded in response to IE Bulletin 79-06B to aid operators in detection of inadequate core cooling and to assure appropriate actions are taken. Additional procedures to be used by an operator to recognize inadequate core cooling are being developed based on analyses and guidelines required by Item 2.1.9, Transient & Accident Analysis, Analysis of Inadequate Core Cooling.

If the analyses or the guidelines indicate the need for the design of new instrumentation, the conceptual design of such instrumentation will be available by 1/1/80 for discussions with the NRC staff to establish generic resolutions and appropriate schedules.

Subcooled Margin Monitor

A conceptual design for the subcooled margin monitor will be completed by January 1, 1980. The monitors themselves will have been ordered, and purchasing processes for additional equipment will be well underway. The limited range on existing process temperature signals, e.g., hot leg temperature, and heavy signal loop loading are problems currently under engineering review. Resolution of these range problems will have a major impact on final design. It should be noted that the proposed revision of Reg Guide 1.97 and NUREG 0578 are in conflict as regards the subcooled margin monitor.

The approaches under review are listed below in order of ascending order of implementation difficulty:

1. Modification of existing signal loops by replacing the existing narrow range transmitters with wide range transmitters, and picking off a narrow range signal for the existing devices, including the RPS TM/LP trip calculator.
2. Use of in-core thermocouple signals.
3. Replacing existing temperature elements with dual elements.
4. Installation of new elements and wells in the primary system piping.

The major problems associated with each approach are listed below:

1. Ability to maintain necessary accuracy on narrow range signal is in question and is being investigated.
2. In-core thermocouples are not safety-related and upgrading may be very difficult.
3. Inability to obtain dual element sensors for the existing wells with response time and accuracy of at least one element adequate for the narrow range signal loop which feeds the RPS TM/LP trip calculator.
4. Need to install new wells in primary system piping

We are pursuing all approaches until one emerges as the most favorable.

As an interim measure, we have begun development of a program addition to the plant computers which will provide comparable information until the permanent installation can be completed. This program will utilize the incore thermocouple inputs to the plant computer. The calculated thermal margin to saturation will be available for direct observation or assignment to a trend recorder. We intend to have this program operable by 1/1/80.

2.1.4 Diverse Containment Isolation

The containment isolation system design for Calvert Cliffs is being modified so that there will be diversity in the parameters sensed for initiation of containment isolation. This change will be completed by 1/1/80 for both units.

A preliminary study of all fluid lines penetrating the containment for the purpose of reconsidering the definition of essential and non-essential systems has been completed. A generic review of all systems penetrating the containment on all operating plants with a CE designed NSSS is being

conducted by the CE Owners Group. This review will produce generic criteria for the definition of essential systems, identification of all such systems, and specification of the bases for selection of each system. Criteria are also being developed for selective unisolation of non-essential systems which may be beneficial. This review should be completed by January 1, 1980.

The preliminary study used, as its basis for selecting essential systems, the criteria that essential fluid systems will be those that are actively required during the early stages of an accident to control and mitigate the consequences of an accident such that exposure to off-site individuals is not in excess of the limits specified in 10 CFR, Part 20.

The essential fluid systems into containment are:

1. High and low pressure safety injection;
2. Containment spray water;
3. Service water to containment structure cooling units;
4. Auxiliary Feedwater (after approximately 15 minutes only); and,
5. Reactor Coolant System Charging line.

The essential fluid systems out of containment are:

1. Containment structure sump recirculation to safety injection pumps;
2. Service water outlet;
3. Main steam lines; and
4. Containment pressure instrument lines.

These essential fluid systems are not automatically isolated by containment isolation which is initiated by the safety injection actuation system (CIS/SIAS), nor by the containment isolation actuation signal (CIS).

The remaining fluid lines penetrating the containment, except the three described below, are considered non-essential. These penetrations are normally locked closed, close on CIS/SIAS, close on SGIS, or close on containment high radiation signal (CRS), or close on loss of piping integrity via check valves. FSAR Table 5-2 describes all of the fluid system containment penetrations and their valve arrangements.

Three containment penetrations are not classified as either essential or non-essential, but are rather classified as being highly desirable during certain plant evolutions. As such, these penetrations are only isolated on high containment pressure (CIS), because this parameter is the only one indicative of the need to isolate these lines. Though

safe shutdown and cooldown and accident response could be accomplished without these fluid systems, their availability adds a measure of reliability and flexibility (and by extension safety) to accident response capability.

These penetrations are:

1. Instrument air supply into containment;
2. Cooling water to the reactor coolant pumps (RCP's); and
3. Cooling water from the RCP's.

The first two of these contain fluid flowing into containment with check valves in the lines; fluid in the last of the three flows into a closed piping system designed for this service. This latter system, cooling water out of containment, is neither part of the reactor coolant pressure boundary nor is it connected directly to the containment atmosphere.

This isolation scheme for CIS is consistent with SRP 6.2.4, section II.5.

To effect a design change to the control schemes for the containment automatic isolation valves, so that manual resetting of the isolation signal will not result in the automatic reopening of the containment isolation valves, requires additional relaying in the control schemes. For the reasons described below, this additional relaying will not substantially improve the safe operations of the plant nor reduce the risk of radiation exposures to off-site personnel. The additional relaying is viewed as actually decreasing the reliability of the containment isolation system by adding additional, unnecessary hardware that has the potential for malfunctioning and negating control room operator control of the isolation valves and/or disrupting normal operation and creating unnecessary challenges to the safety systems.

The reasons for not implementing this reset modification are described in the following paragraphs. It is earnestly requested that you consider these features as sufficient justification for not insisting that we modify our existing systems, and so advise us.

The Calvert Cliffs containment isolation system (part of ESFAS) is not automatically reset when the initiating parameters return within bounds; the system does not allow blocking nor overriding the isolation signals, neither manually nor via any process control signal. Procedures following any actuation which initiates isolation of containment require placing individual hand controls for all isolation valves in their actuated (i.e., safe) position. These procedures contain detailed checklists which list each valve and its required position. When initiating parameters return within bounds, several deliberate operator actions are required before the valves can open: The operator must first manually reset the individual trips, then individually return each isolation valve handswitch to the non-accident position. These features make it unnecessary to modify the automatic isolation system.

In spite of our concerns described herein, we are proceeding with the necessary tasks to facilitate this change. Delivery of safety-related handswitches and control relays is not expected before April 1980.

2.1.5 Post-Accident Hydrogen Control

The Calvert Cliffs post-accident hydrogen control system is described in FSAR Section 6.8. This system currently meets short and long term requirements of the NUREG.

2.1.6a Integrity of Systems Containing Radioactivity

Prior to 1/1/80, integrated leak tests of systems outside containment which will contain highly radioactive materials following an accident will be completed. Sources of leakage will be identified and leakrate reduced to as-low-as-practical levels. Remaining leakages will be measured and the leakrates reported in a summary description of the best program to be submitted by 1/1/80.

Systems to be tested are shutdown cooling, high pressure safety injection, low pressure safety injection, containment spray, containment sump recirculation, containment atmosphere sampling and reactor coolant sampling. The chemical volume control and waste gas systems will not be tested as they will not be used to process highly radioactive fluids following an accident. As an alternative to using these systems, provisions either already exist or will be provided to facilitate:

1. degas of the reactor coolant within the containment,
2. routing reactor coolant pump seal bleedoff flows to the reactor coolant drain tank,
3. routing reactor coolant sample line purge flow back to the containment, and
4. Venting the quench tank directly to the containment.

Since the chemical and volume control system will not contain highly radioactive material following an accident, the release path identified for the North Anna Unit 1 incident described in your letter of October 17, 1979, is not applicable to Calvert Cliffs.

2.1.6b Plant Shielding Review

A design review of plant shielding of spaces for post-accident operations is being conducted. The first phase of this review will be complete by 1/1/80, and will:

1. identify lines which may contain radioactive material in greater concentrations than our current design;
2. identify areas that must be accessed for post-accident operations;
3. provide a general scope of areas that will require additional shielding or other design changes, based on preliminary dose estimates using (1) and (2);

4. provide the basis for detailed design.

The study will be extended to consider effects of an increased radiation field on safety equipment in these areas. By 1/1/81, shielding and/or design changes will be provided where needed.

2.1.7a Automatic Initiation of Auxiliary Feed

An engineering package for automatic start of the auxiliary feed pumps on loss of main feed was developed. The features of the package were described in our letter of November 9, 1979, along with an expression of our concerns relative to the proposed modification. Current design and operating procedures call for manual initiation and control of auxiliary feed on loss of main feed. Following any reactor trip, auxiliary feed comes under the direct control of a dedicated operator. The changes described in our November 9, letter will be completed by mid-January 1980, depending on NRC review.

Investigation is continuing on alternative design modifications of the auxiliary feed system, including automatic start and control of the total system. This requirement will be included with the overall system review and modifications resulting from the Bulletins and Orders group.

2.1.7b Auxiliary Feed Flow Indication

Upgrading of auxiliary feedwater flow indication on Unit 2 is in progress. Currently, auxiliary feed flow indication is in service on Unit 1 and it will be upgraded to safety grade. This upgrading is expected to be complete by January 1, 1980.

2.1.8a Improved Post-Accident Sampling Capability

A preliminary design and operational review of both the reactor coolant and containment atmosphere sampling facilities will be done by January, 1980, to determine our capability to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18 3/4 Rems to the whole body or extremities respectively. Results of the preliminary review and an outline of resulting procedures and any plant modifications will be available by January, 1980. Plant changes currently under review include control changes to containment isolation valves in the sampling, systems, and piping/shielding changes to the systems.

A preliminary design and operational review of the radiological spectrum analysis facilities will be performed to determine our capability to promptly quantify (less than 2 hours) certain radioisotopes that are indicators of the degree of core damage. The review will consider radiation effects as described in the NUREG. Results of the preliminary review and an outline of resulting procedures and any plant modifications will be available by January, 1980.

Our current procedures regarding the analysis of boron and chloride will be reviewed to determine if changes should be made to allow analysis assuming a highly radioactive initial sample while maintaining exposures

within the limits described above. The procedural review will encompass the feasibility of completing the analysis for boron within an hour and the chloride within a shift. Procedure changes will be completed by 1/1/80.

We plan to complete plant modifications resulting from any of the above by 1/1/81.

2.1.8b Increased Range of Radiation Monitors

Wide range noble gas monitors with associated indicators and recorders will be installed at the appropriate release points by January 1, 1981. These monitors will be powered from the vital instrument buses and will be seismically qualified. Our present monitors presently measure up to 1 Ci/cc with a release rate of 45 Ci/sec, which is greater than those at TMI. Our schedule calls for the proposed system will be sent to the NRC for review in April.

A review of our main vent instrumentation measurement capability for noble gases, radioiodines and particulates will be performed as well as our procedures for sampling and analysis of grab samples. Procedures will be developed by January, 1980, to estimate release of noble gases, radioiodines and particulates in the event the main vent instrumentation goes off scale.

Redundant high range radiation monitors for the containments with associated indicators and recorders will be installed by January 1, 1981. They will be seismically qualified and powered from a vital instrument bus. Our schedule calls for the proposed system will be sent to the NRC for review in February.

2.1.8c Improved In-Plant Iodine Instrumentation

Current iodine measurements are performed using gamma energy spectrum analysis. Our current method of analyzing in-plant samples uses a computer-based multichannel analyzer. This method is more than adequate to monitor the radioiodine concentrations which will be collected on charcoal cartridges. In addition, the laboratory contains adequate devices to reduce noble gas interferences. This appears to meet all requirements of the NUREG.

Portable single channel analyzers (SCA's) would not seem to offer any improvement over our existing system as intermediate monitoring devices. The SCA is useful only in low radiation areas. Although designed to discriminate out energies other than the energy of interest, other energies do influence the background counting level. Use of SCA's could result in a false indication of activity levels and a delay in analytical time. In most cases the sample would be in a region where the radiation fields would interfere with the counting of the charcoal cartridge, resulting in either an inadequate minimum detectable activity level or a false indication of radioiodine concentrations.

Though we see negligible benefit in their use, we are initiating purchase of a number of SCA's to satisfy NRC concerns. These will be available on site by about April 1980.

2.1.9 Transient and Accident Analysis

The response to Transient and Accident Analysis requirements is being developed by the CE Owners Group in conjunction with generic resolution meetings with the NRC Bulletins and Orders Task Force. These responses will be submitted on the schedule agreed to by that Task Force and the CE Owners Group and will be referenced for specific application to Calvert Cliffs.

Additional Equipment:

Containment Pressure Indication

Continuous indication in the control room of containment pressure in the range of (-) 5 to 150 psig will be installed by January 1, 1981. The transmitters will meet the requirements of Regulatory Guide 1.97 Revision 1. The new revision of Regulatory Guide 1.97 will be reviewed for appropriateness when received. Presently there is the capacity to measure from (-) 4 to 60 psig with indication in the main control room.

Containment Water Level Indication

By January 1, 1981 a level indication of the entire normal sump range will be installed in the two units and displayed in the control room. The instruments will meet the requirements of Regulatory Guide 1.89. A wide range level transmitter will be installed in both containments with indication in the control room by January 1, 1981. The transmitters will meet the requirements of Regulatory Guide 1.97, revision 1. The new revision of Regulatory Guide 1.97 will be reviewed for appropriateness when received. Presently two level transmitters per unit exist to measure the entire emergency sump level (into which the normal sump overflows) with indication and alarm in the control room. These transmitters are qualified for a LOCA environment.

Containment Hydrogen Indication

There are redundant hydrogen recombiners inside each containment. Each one is designed to maintain the hydrogen concentration at 3 volume percent. The flow rate needed to maintain the 3 volume percent is one half the rated flow rate of each recombiner. The recombiners are designed to operate under LOCA conditions, are seismically qualified, are physically located on opposite sides of the containment, are operated from the control room, and are powered by separate, redundant, emergency sources. Calculations for the design of these recombiners were based on the conservative assumptions of Safety Guide 7. Since recombiners are designed to limit hydrogen content to 3 volume percent, hydrogen analyzer has a range of 0-4%. Any reading over 2% will alarm in the control room. The projected radiation level for the room in which the hydrogen analyzer control panel is located is acceptable for short-term occupancy after a LOCI. However, remote indication in the main control room will be installed by 1/1/81. Because of this system, a 0-10% H₂ concentration range for an analyzer is not needed.

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RCS Venting

Preliminary engineering and procurement of major pieces of equipment have begun. The design concept, but not the final design drawings, will be complete and provided to NRC for review by 1/1/80. Installation of equipment in both units will be complete by 1/1/81.

2.2.1a Shift Supervisor Responsibilities

The requirements of the NUREG position will be instituted by January 1, 1980.

2.2.1b Shift Technical Advisor

The present plans to satisfy the NUREG position regarding the Shift Technical Advisor (STA) are based on discussions with the NRC staff held on this subject at the topical meeting in Bethesda, MD on October 12, 1979. Even though our plans do not meet the literal requirements of the NUREG position, we believe we will meet the intent of the NUREG as expressed at the October 12 meeting. Specifically, we intend to separately institute measures to satisfy the two major goals of the NUREG discussion regarding the duties and functions of the STA, as set forth below.

Operating Experience Assessment

Before January 1, 1980, we intend to establish a standing committee to regularly review the operating experience at the Calvert Cliffs plant and at other plants of like design. This committee will be primarily composed of staff engineers located on-site and will be augmented as necessary by engineers from our Engineering Department (located in Baltimore). The purpose of the reviews conducted by the committee will be to provide an independent monitoring of the safety of plant operations and to provide, if necessary, a counter-balanced perspective to the needs of the commercial aspects of plant operation. Additionally, by the review of the operating experiences of other plants of like design, valuable input may be derived which could improve the safety of the Calvert Cliffs plant.

The Chairman of this committee will be located onsite and will make reports directly to the Offsite Safety Review Committee. The onsite committee will not be staffed by members of the plant operations unit, although such persons may be called upon by the chairman to provide input as necessary to the committee's deliberations.

In order to assure that the facets of the review process which are pertinent to operator and technician training are properly factored into the training program, the plant training coordinator will be a permanent member of the committee.

Presently, many of the elements of such a review process are carried out by diverse organizations both onsite and offsite. This being true, the actual time spent presently in conducting such reviews is not formally documented. Additionally, since the formation of this committee will intensify these reviews, future demands on the personnel involved is

not presently clear. Should the workload of performing these reviews become overly burdensome to present staff personnel, the staff complement will be increased to provide an adequate number of engineers to properly conduct the committee reviews.

Accident Assessment and Response

In order to provide the dedicated accident assessment and response function set forth by NUREG 0578, we plan to utilize the present complement of Senior Operator License (SOL) persons assigned to the shift operating organization. In doing so, we believe it is of paramount importance to assure that at all times a trained, qualified individual who is current in the operational status of the units is immediately available to the Control Room to make such assessments. This individual can have no "hands-on" responsibilities to perform during the response phase of an accident condition. In order to assure that the individual performing this function remains detached from "hands-on" duties, we plan to institute the following measures by January 1, 1980.

1. Each unit which is in an operational mode above Cold Shutdown will always have two Operator License (OL) persons assigned to the unit. Since the units share a combined Control Room, during normal operation this will provide four O.L. persons in the Control Room. Administrative procedures will establish the manning level such that only one of the four can be absent at any one time and then only for brief periods (approximately ten minutes). Since all of our Control Room Operators are licensed on both units, this will insure that each operating unit will always have two full-time, dedicated "hands-on" operators in the event of an accident situation. A majority of the time, the "extra" operator from the non-affected unit will also be available to assist in "hands-on" functions for the affected unit.
2. Since our Technical Specifications require the continual presence of two SOL operators onsite, the implementation of OL manning level described above will assure that one of the SOL operators will always be available to perform the desired accident assessment function. We plan to assign this responsibility to the Senior Control Room Operator (SCRO), thus allowing the Shift Supervisor to perform the "Command and Control" functions associated with an accident response. The specific responsibility of the SCRO in performing accident assessment functions will be included in a management procedure and emphasized in the training of these personnel.

As discussed at the October 12 topical meeting, an accelerated training course will be provided for SOL personnel to enhance their ability to perform the accident assessment functions. The training will include such general subjects as reactor physics, reactor thermodynamics, fluid mechanics, heat transfer and reactor control theory. Specific training in the response of the plant to accident and transient conditions will also be included in the course of instruction.

The plant staff is presently working with several training consultants and the NSSS Supplier to formulate the training program which will be required to upgrade the existing Senior Operator License holders to perform the "accident assessment" function. Preliminary estimates are that each group of attendees would be in training for approximately 20-25 weeks. Our present goal is to divide the available personnel into three groups; two groups will remain on-shift while one group is in a training status.

Additionally, it is our goal to commence the training as early in 1980 as practicable. Within the framework of this schedule, the first two groups will complete the training prior to the end of 1980 and will be available for assignment to the shift organization, thus allowing the shifts to be continuously manned after 1/1/81 by personnel who have undergone the training. As described above, the 1/1/80 requirement to have STA on duty is already being met.

2.2.1c Shift and Relief Turnover Procedures

The requirements set forth by the NUREG position will be instituted by January 1, 1980. In our initial review of the concepts set forth therein, we feel that most of the elements of this position are already in place, even though they are not in a formalized checklist. In some cases, to provide such a checklist would be a duplication of already existing mechanisms; for example, the logging of plant parameters and limits on the checklist would be a duplication of our existing Control Room Log. In cases where such duplication would exist, the checklist may be used as a mechanism of documentation, rather than instituting a duplicative procedure.

2.2.2a Control Room Access

The requirements of the NUREG position will be instituted by January 1, 1980.

2.2.2b Technical Support Center

To implement this position, we will establish by 1/1/80 an onsite Technical Support Center (TSC) in the 55' elevation of the Log and Test Instrument Room, shown on FSAR figures 1-8 and 1-12. This room is approximately 23 feet by 40 feet and is included in the ventilation system provided for the Control Room/Cable Spreading Room (described in FSAR Section 9.8.2.3). Additionally, this room is within the Seismic I boundary of the Auxiliary Building. As can be seen from the FSAR figures, the room can be entered directly from the Turbine Building, and also is connected to the Control Room via a single door on the 45' elevation.

The present communications capabilities from this room includes three conventional telephones and a plant page system. Upgrading of the entire plant telephone system is presently in progress, and expansion of the phone system for the TSC will be included in this upgrading. The final plans for this upgrading are expected to be completed by the end of this year.

A plant "Record Retention and Retrieval Center" is presently located in the northwest corner of the Control Room. This center provides controlled copies of all the documents listed in the NUREG position. Since this center is presently located in the rear of the Unit 1 control panels, access intruding into the front-panel area of the control panels. Consideration is being given to relocating the center to the Log and Test Instrument Room.

Design and procurement is underway to locate a portion of the "Startup/Physics Test Panel" in the shop. The extension of this panel to the shop will provide the capability to readout (with temporary recorders) the following parameters:

1. Steam Generator Pressure
2. Steam Generator Level
3. Feed Water Flow
4. Steam Flow
5. Steam Header Pressure
6. Turbine First Stage Pressure
7. Hot Leg RCS Temperature
8. Cold Leg RCS Temperature
9. Pressurizer Pressure
10. Pressurizer Level
11. RCS Flow
12. Steam Generator "Delta" Pressure
13. Feed Water Pressure
14. Power Ratio
15. Turbine Bypass Valve Position
16. Atmospheric Dump Valve Position
17. Narrow Range Nuclear Instruments

This installation should be complete by 2/15/80.

Plans and procedures for support staffing of the TSC are being developed in conjunction with the revision to the Site Emergency Plan, the first draft of which is scheduled to be done in January, 1980.

Longer range plans will be outlined by 1/1/80, and implemented to meet all requirements by 1/1/81.

2.2.2c Operational Support Center

The requirements of the NUREG position will be instituted by January 1, 1980.

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