VIRGINIA ELECTRIC AND POWER COMPANY

RICHMOND, VIRGINIA 23261

November 26, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 906
PO/DLB:smv
Docket Nos. 50-280
50-281
50-338
50-339

Dear Mr. Denton:

LESSONS LEARNED SHORT TERM REQUIREMENTS
SURRY POWER STATION UNITS 1 AND 2
NORTH ANNA POWER STATION UNITS 1 AND 2

We have reviewed your letter of October 30, 1979 regarding implementation of the short term requirements resulting from the staff investigation of the TMI incident. A conference call was held on November 16, 1979 with members of your staff to discuss our response to your letter of September 13, 1979 regarding the short term requirements. The attachments provide the information requested in your October 30, 1979 letter and also provide clarification of several items as requested by your staff during the conference call. Attachment A addresses Surry Units 1 and 2 and Attachment B addresses North Anna Units 1 and 2.

Our initial responses to the short term requirements were forwarded in our letter of October 24, 1979 for Surry Units 1 and 2 and North Anna Unit 1 and our letter of October 25, 1979 for North Anna Unit 2. At that time, due to uncertainties in design and procurement, we were unable to provide firm estimates of the completion dates for several items. Accordingly, for those items we were unable to provide assurance of our ability to comply with the completion dates specified in NUREG 0578. Since that time we have made substantial progress in implementing the short term requirements and in refining our completion schedule. The attachment includes revised schedules for all short term requirements for which we were originally unable to commit to the specified completion date. As explained in the attachment, in almost every case we have been able to expedite implementation to meet the required completion date.

There remain only a very few items which, due to procurement delays, may not be installed by the specified date of January 1, 1980. These NUREG 0578 items are as follows: 2.1.1.1 (Surry 1), 2.1.3.a (Surry 1), 2.1.3.b (North Anna 1 & 2 and Surry 1) and 2.1.7.b (Surry 1).

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J. BEARD	1	1
D. VERRELLI	1	1

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We will continue to inform you of our progress in implementing the short term requirements and will be glad to answer any questions you may have.

Very truly yours,

C. M. Stallings

C. M. Stallings
Vice President - Power Supply
And Production Operations

DLB/smv:E2

cc: James P. O'Reilly

ATTACHMENT A

ADDITIONAL INFORMATION ON THE IMPLEMENTATION OF
NRC SHORT TERM REQUIREMENTS RESULTING FROM
THE REVIEW OF THE THREE MILE ISLAND INCIDENT
SURRY POWER STATION UNITS 1 AND 2

Following is information regarding the status of implementation of the short term lessons learned requirements as presented in NUREG 0578 and in H. R. Denton's letter of September 13, 1979. This information is submitted in response to Mr. Denton's letter of October 30, 1979 and in response to questions discussed with the NRC staff in a conference call on November 16, 1979. The information presented falls into the following three categories:

1. Updated scheduling information for all items for which we were originally unable to commit to the NUREG 0578 completion date.

2. Clarifications requested in or required by Mr. Denton's letter of October 30, 1979.

3. Clarifications or additional information requested by the NRC staff in a conference call on November 16, 1978.

Items are numbered as in NUREG 0578. Only those NUREG sections where specific updating or new information is required are listed.

2.1.1.3.1 PRESSURIZER HEATER POWER SUPPLY

Based on the clarification provided in your letter of October 30, 1979, the qualification review of pressurizer motive and control power interfaces with the emergency buses will be completed by the required date of January 1, 1980 rather than our previous commitment date of March 31, 1980.

Response to Clarifications of October 30, 1979 Letter

1. Each redundant group of backup pressurizer heaters has access to only one Class 1E 480V bus.

2. Each redundant group of heaters is supplied from emergency power sources and has the capability to provide the heat required to maintain natural circulation.

3. Existing diesel generator loading indicates that with the pressurizer heaters energized at any time following a LOCA with a loss of offsite power each diesel generator will be within its continuous kw rating.

4. Change over of the heaters from the normal offsite power to emergency power does not require manual action. Power will be automatically transferred at the 4kv bus voltage level via the existing diesel generator start and load scheme.

5. Based on the existing scheme manual reloading of the pressurizer heaters is not required. (See discussion of clarification 4.)

6. The Class 1E interfaces for motive and control power are protected by safety grade circuit breakers.

7. The pressurizer heaters are not automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal. Per the discussion of clarification 3 above, this will not result in an overloading of the diesel generators.

2.1.1.3.2 POWER SUPPLY FOR PRESSURIZER RELIEF AND BLOCK VALVES AND PRESSURIZER LEVEL INDICATORS

The qualification review of the motive and control power connections to the emergency buses for the PORVs and block valves will be completed by the required date of January 1, 1980, rather than our previous commitment date of March 31, 1980.

Our previous response to position 1 should be revised to state that the motive power to the PORVs will be upgraded by modification of the high pressure air supply system. Additional review has determined that a minor modification to the high pressure air supply system is required to ensure redundant motive power to the PORVs. This will be completed on Unit 2 prior to startup from the current outage for steam generator repairs, and on Unit 1 during the outage described in our response to item 2.1.3.a.

Response to Clarifications of October 30, 1979 Letter

1. During reactor power operation, each PORV is opened by energizing two solenoid valves which admit air to the PORV actuator to open the valve. De-energizing or loss of power to the solenoid valves causes the PORV to close. The respective PORV solenoid valves are powered from separate 125V DC emergency buses.

   Each block valve is motor operated and requires power to open and close. The block valves are powered from separate 480V AC emergency buses. During normal shutdown water solid operation, each PORV is opened by energizing a different third solenoid valve which admits air to the PORV actuator. The PORV closes on de-energizing the third solenoid valve. The PORV third solenoid valve is powered from separate 125V DC emergency buses.

2. The motive and control power for the block valve is supplied by a 480V AC emergency power bus that is not associated with the 125V DC emergency bus which supplies the PORV solenoid valves.

3. PORV solenoid valve control power is from 125V DC emergency battery buses and thus will not shift on loss of normal offsite power. Block valves are connected to the emergency buses which are normally fed from the offsite power supply. Failure of the offsite power supply will cause the motive and control power for the block valves to shift automatically to their respective emergency onsite power supplies.

4. To ensure motive power to the PORVs during the loss of normal offsite power, high pressure seismic instrument air will be provided to supplement the normal air supply system to the PORVs. This will be accomplished by relocating check valves in the crossconnects to the recently added solenoid valve for the dedicated water solid operation control.

2.1.3.a. DIRECT INDICATION OF VALVE POSITION

In our initial response of October 24, 1979, we committed to install acoustic monitors on the reactor coolant system safety valves. We will be unable to have this system installed by January 1, 1980. The equipment has been ordered and is scheduled for delivery in January, 1980. The system will be installed on Unit 2 prior to startup from the present outage for steam generator repair. The system will be installed on Unit 1 during an outage to include modifications and installations required by NUREG 0578 items 2.1.1, 2.1.3.a., 2.1.3.b. and 2.1.7.b. The outage will be the first outage of sufficient duration following receipt of all materials and design information required for these four items. The installation of these items will be no later than the refueling outage scheduled for June 1980.

In the interim, the following indications of safety valve positions are available:

1. Temperature indication is provided on the dicharge piping of each safety valve.

2. Both temperature and level in the pressurizer relief tank are monitored.

3. The PORVs are set to open at a lower pressure than the safety valves and the PORVs have direct indication.

Should the safety valves actuate after the PORVs have actuated, and if blowdown, as evidenced by items 1 and 2, continues after the indication of PORV closure, a malfunctioning safety valve would be recognized.

The alarms on the limit switches of the PORVs required by the clarification 2 on page 7 of your October 30, 1979 letter will be installed on Surry Unit 2 prior to startup from the present outage for steam generator repair. The alarms will be installed on Unit No. 1 during the above mentioned outage for other NUREG 0578 items.

Response to Clarifications

The direct position indication of each of the two Power-Operated Relief Valves (PORV) is presently derived from limit switches. The existing limit switches will be replaced by environmentally-qualified limit switches and a common control room alarm for each unit will be added to alarm when the PORVs or the pressurizer safety valves are not closed. The environmentally-qualified limit switches will be installed during the first scheduled outage of sufficient duration after material delivery.

The valve acoustic monitoring system will provide two channels for each pressurizer safety valve, one for indication, the other for backup. One channel is actively connected to the monitoring equipment while the second channel is terminated in a plug passively

mounted in the valve monitor cabinet. The active channel may be unplugged and replaced with the passive channel at the valve monitor cabinet mounted in the control room should the need arise. Additionally, backup indication of valve position is provided by temperature detectors in the piping downstream of the safety valves.

Each Pressurizer Safety Valve will have a valve position alarm indicator mounted on the main control board. In addition, a common control room alarm for each unit will sound when any Pressurizer Safety Valve or PORV is not in the fully closed position.

The power supply to the valve monitoring system will be derived from vital buses in Units 1 & 2 with an automatic transfer on the loss of one vital bus.

The valve monitoring cabinet will be seismically supported in the control room. The valve acoustical monitoring system will be seismically and environmentally qualified by the vendor (Babcock and Wilcox Co.). The Vendor is establishing a test program to bound the worst case environmental and seismic conditions of its application. The purchasing utilities will submit the worst case conditions early in December and a test program will be established early in 1980. The completion date for the qualification will be provided when the extent of the test program has been established.

2.1.3.b INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

Subcooling Meter

The subcooling meter may not be installed by January 1, 1980. We have committed to a fully qualified system that meets the requirements of NUREG 0578 and are making a concentrated effort to have it installed and operational by January 1, 1980.

In an effort to ensure that the system fully meets the NUREG 0578 requirements, and to ensure we obtain the best system currently available, a purchase order was not placed until after a thorough review of vendor proposals and the clarifications in your letter of October 30, 1979. An order was placed with Westinghouse Electric Corporation for their subcooling meter on November 16, 1979. Their estimated equipment delivery would support installation by January 1, 1980. However, Westinghouse will not be able to finalize their interface requirements until the first week of December. The engineering and design required to ensure that the installation of the system does not adversely impact the reactor protection or engineered safety system was initiated in October but will extend beyond January 1, 1980. In addition, until particular interface requirements are identified by Westinghouse, we are unable to purchase some long lead time items such as isolation devices.

Installation of the subcooling meter will be completed on Unit 1 not later than the outage discussed in our response to item 2.1.3.a. Installation on Unit 2 will be completed prior to startup from the current outage for steam generator repairs.

In an effort to install the subcooling meter as soon as possible, the system may not be fully qualified to Class 1E application upon installation. However, Westinghouse has assured us that based on previous testing, the equipment can be qualified to these standards. In the short term the hot leg and cold leg temperatures, twenty existing incore thermocouples and the Reactor Coolant System pressure and pressurizer pressure inputs will be provided by paralleling existing computer input terminations for the control loops and wiring them to each subcooling meter electronics.

It is the opinion of Westinghouse that the system as permanently installed will meet all the requirements of NUREG 0578, item 2.1.3.b.1. The detailed information requested in your October 30, 1979 letter will be presented as it is available. A brief technical description of the system is presented below.

> The core subcooling monitor utilizes inputs from the existing hot leg RTDs, reactor coolant system pressures, and 20 selected incore thermocouples. A microprocessor is employed to calculate saturation temperature for the existing reactor coolant system pressure and determine the margin to saturation based on the various temperature inputs. The information display consists of a main control board indication of the margin to saturation and expanded information at the electronics drawer which will be located with the existing plant protection and control equipment.

The system is divided into two redundant channels, each accepting inputs from two hot leg RTDs and 10 thermocouples. The redundant analog meters mounted on the main control board have nonlinear scales, expanded near saturation, with the range extended to the limits of the incore thermocouples. In addition to the numerical indication, the meter faces are color coded to increase visual acuity for the operators.

The auctioneered low reactor coolant system pressure is utilized by the microprocessor to calculate the saturation temperature for the existing system pressure. By subtracting the auctioneered high incore thermocouple signal from the calculated saturation temperature, the current margin to saturation is calculated. This information is then displayed on the main control board meters.

The margin to saturation from the auctioneered high hot leg RTD is also available on the main control board meters by means of a two position switch to be located on the main board.

Two levels of alarm are provided; the first to indicate the development of off-normal conditions, the second the approach to loss-of-core subcooling. The actual set point for either alarm is controlled by the microprocessor and is easily modified by keyboard entries at the main processing unit.

Expanded information is displayed on the front of the electronics drawer to enable the operator to determine core cooling conditions. Each of the redundant microprocessors will utilize the computed saturation temperature for comparison with the following plant parameters: individual loop wide range T_{hot}, auctioneered high range T_{hot}, average incore thermocouple, auctioneered high incore thermocouple. In addition, the following information will be available for display: the temperature difference between the average of the incore thermocouples and individual loop T_{hot} and time to a loss of subcooling based on the rate of change of reactor coolant pressure and temperature.

In addition to the above digital displays, a graphic, color, 15 lamp display of the incore thermocouples is provided on the electronics drawer. The core is divided into five regions, four peripheral and one central, with inputs from two thermocouples used to generate the display for each region. Each region has three lamps, green, yellow and red. Two sets of three lamps are also provided for the hot leg RTDs input into each drawer. If all temperatures are sufficiently below saturation, the lamps will be green. If any individual temperature reaches an off-normal condition, the lamp display changes to yellow and alarm contacts for the

plant annunciator are closed. Should conditions continue to degrade to the preset margin to saturation, the lamp display changes to red and an additional set of alarm contacts are closed for the plant annunciator.

Until this system is operational other mechanisms have been provided to the operators to enable them to immediately assess the primary coolant margin to saturation conditions.

Changes to emergency procedures have been made to emphasize the need to insure adequate coolant flow and to insure that the reactor coolant temperature and pressure is maintained or immediately restored to achieve an appropriate margin to saturation. All licensed reactor operators and current trainees have received special instruction in the TMI type incident with particular emphasis on the use of existing instrumentation to determine core conditions.

The plant computer will be used to monitor plant saturation conditions. The operator can initiate a trend printout of the system saturation temperature and one or more of the critical reactor coolant system temperatures.

A saturation curve has been posted on the control board and provided in current procedures. This curve, combined with the nearby indications of reactor coolant system temperatures and pressures, enables the operator to quickly determine the systems margin to saturation.

As explained in our earlier response of October 24, 1979, Westinghouse has developed a listing of all instruments required for an operator to diagnose the type of plant event, take the necessary manual actions, and to monitor critical plant conditions. Surry currently has all of the recommended instrumentation.

We believe that the above listed instrumentation, procedures and operator training will provide assurance that the operator can adequately assess plant saturation conditions until installation of the subcooling meter is completed.

Additional Instrumentation

Regarding instrumentation to monitor reactor vessel water level, we have not at this time identified a reactor vessel level measurement technology which is unambiguous under all conditions. We are continuing our evaluation of potential reactor vessel level measurement techniques. As soon as a suitable technology is identified, we will initiate design and procurement activities. We will make a concentrated effort to have the selection and installation of the level instruments completed by January 1, 1981.

2.1.4 DIVERSE CONTAINMENT ISOLATION

Our initial response of October 24, 1979 provided a brief description of the Surry containment isolation scheme and a listing of the "essential" and "non-essential" containment penetrations. In a conference call on November 16, 1979, members of the NRC staff requested a brief summary of the bases for designation of a system as "essential" or "non-essential". These bases are as follows:

Essential systems are divided into two categories: 1) systems required to mitigate the consequences of various types of accidents, and 2) systems which are required to maintain the operation of critical systems and functions. Included in category 1 are the engineered safeguards features including the containment and recirculation spray systems, the safety injection systems, and the service water system which provides cooling to be the recirculation spray heat exchangers. Category 2 includes auxiliary feedwater, and component cooling required for reactor coolant pump operation, containment air cooling, and residual heat removal.

Non-essential systems include all other systems not required for the above functions.

2.1.5.c <u>CAPABILITY TO INSTALL HYDROGEN RECOMBINER AT EACH LIGHT WATER NUCLEAR POWER PLANT</u>

The following items of clarification are provided as requested by the staff in the November 16, 1979 conference call.

As stated in our initial response, Surry Units 1 and 2 each have two hydrogen recombiners located inside the containment. They are remotely controlled from the Control Room Annex. Due to the location of the recombiners and the remote operation feature, complications involving personnel exposure during system lineup do not apply to Surry.

The operational bases and procedures for hydrogen recombiner use were reviewed in response to IE Bulletin 79-06A. Due to the location and remote operation of the Surry recombiners an additional review of recombiner operation to address shielding concerns is not considered necessary.

In our October 24, 1979 response we stated that provisions exist for installation of external recombiners. The installed in-containment recombiners provide the requisite hydrogen removal capability as defined in the FSAR. Since the existing in-containment recombiners meet all requirements as defined in the FSAR, no specific provisions or procedures exist or are necessary for use of the external recombiners.

2.1.6.a INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIALS

In our response of October 24, 1979, we stated that a review was in progress to identify those systems outside containment which, in addition to mitigation systems, are likely to contain highly radioactive fluids during or following an accident. This review is now complete and forms the basis for our immediate and continuing leak reduction programs. By January 1, 1980 on Unit 1 and prior to startup on Unit 2, we will complete the following:

1. All outside containment systems which are likely to contain highly radioactive fluids during or following an accident will be inspected and tested to identify any leaks. All practical leak reduction measures will be applied to reduce leakage from these systems.

2. Leakage rates from all such systems will be measured or estimated during operation or during a simulation of operating conditions. A summary of the findings will be forwarded.

3. A long term leakage measurement and preventive maintenance program will be developed. The leak test interval will be no greater than a refueling cycle. Details of the program including a listing of the systems covered and systems not covered will be forwarded.

2.1.6.b PLANT SHIELDING REVIEW

Shielding reviews for Surry Units 1 and 2 and North Anna Units 1 and 2 are in progress. Both reviews are being performed by the same review team. Due to similarities in design between the two stations, a large portion of the review work has been applicable to both stations. As a result, we have been able to improve on our original schedule. The radiation and shielding design review for Surry Units 1 and 2 will be completed by January 1, 1980.

2.1.7.b AUXILIARY FEEDWATER FLOW INDICATION TO STEAM GENERATORS

Response to Clarification of October 30, 1979 Letter

The auxiliary feedwater flow indication for each steam generator is powered from a semi-vital bus. Each existing flow loop will be changed from the semi-vital bus to a vital bus which meets the diversity requirements. This modification will be completed on Unit 2 prior to startup from the present outage for steam generator repair. This modification will be completed on Unit 1 during the outage identified in our response to item 2.1.3.a.

The auxiliary feedwater flow indication is testable from the transmitter back to the indicator.

At present, the Auxiliary Feedwater Flow indication is not single failure proof. However, safety grade steam generator level on a separate vital power supply is provided as a backup indication for each loop.

The Auxiliary Feedwater Flow indication satisfies the requirement of ±10 percent accuracy.

2.1.8.a IMPROVED POST-ACCIDENT SAMPLING CAPABILITY

The clarifications presented in your letter of October 30, 1979 resolved many of our questions concerning specific sampling requirements. Therefore, a review of our sampling capability as well as a conceptual design of an improved sampling capability will be provided for Surry Units 1 and 2 by January 1, 1980.

2.1.8.b INCREASED RANGE OF RADIATION MONITORS

Response to Clarifications

An evaluation of interim methods of estimating high level noble gas releases which meets the requirements of table 2.1.8.b is in progress for monitored effluent locations and other potential release points. The interim methods will employ the use of dedicated portable and/or rigidly mounted detectors at potential release points which will be sufficiently shielded from background and will be collimated, if required, to obtain readings within the range required. Conversion of the measured dose rate into a concentration will be performed using standard volume source calculations and predetermined conversion factors. The capability will exist to obtain a radiation reading every 15 minutes, either by remote read-out or by phone communication with an operator. The power source will be arranged such that alternate AC power will be available, if required, or a DC source will be available for seven consecutive days. Procedures will be developed for the removal and analysis of the radioiodine particulate media. Procedures also will be developed for dissemination of information, calibration of equipment and minimizing occupational exposures.

Since these short term requirements were not identified until your October 30, 1979 letter, a firm schedule for implementation is not yet available. However, the majority of the methods will probably use currently available equipment and any delay in installation due to equipment delivery will be a minimum. Procedures will be developed, approved and implemented as soon as the methods are finalized. Therefore, some of the procedures may not be in effect on January 1, 1980. It is expected that all procedures will be in effect by February 1, 1980.

In our initial response to the long term requirements, we identified our concerns that commercially available monitors may not be available for the extended ranges required. We still believe this to be the case but will make every effort to meet the requirements for inplant monitor capabilities by January 1, 1981.

2.1.8.c IMPROVED IN-PLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

The adequacy of the silver zeolite sampling cartridges under accident conditions will be documented by January 1, 1980. If necessary provisions to flush the cartridge with clean gas will be provided by January 1, 1981. Provisions for removal of the cartridge to a low background, low contamination area for analysis will be provided by January 1, 1981.

2.1.9 TRANSIENT AND ACCIDENT ANALYSIS

Containment Pressure Indication
Response to Clarification of October 30, 1979 Letter

The present design for containment pressure indication has four channels of protection grade transmitters with continuous indication in the control room. Their power supply is from Vital Buses I, II, III and IV. The lower range requirement (minus 6 psig) is met. The upper range of indication will be increased to approximately 180 psia. The containment pressure indications will be upgraded to meet the intent of the design and qualification of Regulatory Guide 1.97 including qualifications, redundancy and testability. These modifications will be accomplished by the required date of January 1, 1981.

Our method of meeting the intent of Regulatory Guide (RG) 1.97 for this and subsequent items utilizes plant licensing criteria as follows:

A. RG 1.97 position C4 imposes RG 1.80 Qualification Requirements. The qualification of existing instrumentation is in compliance with existing plant licensing/FSAR commitments. Qualification requirements of new equipment will address the requirements of RG 1.89. As a minimum, the new equipment will meet IEEE 323-1971 and IEEE 344-1976.

B. RG 1.97 position C7 cites RG 1.75 criteria. Our redundant monitoring channels will be fed from Class 1E power and separated in accordance with our licensed plant separation criteria.

Containment Hydrogen Indication
Response to Clarification of October 30, 1979 Letter

A continuous indication of hydrogen concentration in the containment atmosphere will be provided in the control room. Modifications are required to provide remote isolation valves and control room readout. Measurement capability is presently provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

The existing hydrogen analyzers are being evaluated to determine if they need to be modified to meet the intent of Regulatory Guide 1.97. One method being evaluated is to remove the electronics from the hydrogen analyzer package and mount it in an environmentally less severe location (away from the sample line and sensor). The sensor could be upgraded to qualify it to radiation requirements, if required. If the evaluation concludes that the existing hydrogen analyzers could not meet the intent of Regulatory Guide 1.97, new hydrogen analyzers will be purchased which will meet the requirements of Regulatory Guide 1.97. These modifications will be accomplished by January 1, 1981.

Containment Water Level Indication
Response to Clarification of October 30, 1979 Letter

The wide range indication will be provided by installing redundant level transmitters located in the containment. These transmitters

will have a range above the containment floor which is equivalent to about 600,000 gallons and which encompasses the maximum calculated water level during an accident. Indication from both transmitters will be continuously displayed in the control room. This will be completed by January 1, 1981. The transmitters will be environmentally qualified to Surry Units 1 and 2 LOCA conditions, as well as seismically qualified. The indication channels will meet the intent of the design and qualification provisions of Regulatory Guide 1.97 for wide range.

The narrow range indication channel will be upgraded by January 1, 1981, as required, to meet the intent of the design and qualification provisions of Regulatory Guide 1.97.

Reactor Coolant System Vent

Design of reactor coolant system and reactor vessel high point vents is underway. However, there are many questions concerning proper design and operation that remain unresolved. By Januay 1, 1980, we will present a conceptual design of the RCS Vent System. We will make a concentrated effort to install the system by January 1, 1981. However, the system cannot be installed until a proper and thorough design review has been performed and adequate procedures are developed for its operation.

2.2.1.b SHIFT TECHNICAL ADVISOR

In our initial response of October 24, 1979 we indicated that the operating experience assessment function would be performed by the Operating Supervisor. In discussions with the staff on November 16, 1979, we agreed to designate an individual other than the Operating Supervisor, whose primary responsibility would be the operating experience assessment function. An individual with the appropriate operating experience and technical knowledge will be assigned to this position by January 1, 1980.

2.2.2.b ONSITE TECHNICAL SUPPORT CENTER

Items 1.A through 1.G of your October 30, 1979 clarification will be completed by January 1, 1980. Our schedule for upgrading the Technical Support Center, as required by items 2-10 of your clarification, will be provided in our response to item 1.G.

ADDITIONAL INFORMATION ON THE IMPLEMENTATION OF
NRC SHORT TERM REQUIREMENTS RESULTING FROM
THE REVIEW OF THE THREE MILE ISLAND INCIDENT
NORTH ANNA POWER STATION UNITS 1 AND 2

Following is information regarding the status of implementation of the short term lessons learned requirements as presented in NUREG 0578 and in H. R. Denton's letter of September 13, 1979. This information is submitted in response to Mr. Denton's letter of October 30, 1979 and in response to questions discussed with the NRC staff in a conference call on November 16, 1979. The information presented falls into the following three categories:

1. Updated scheduling information for all items for which we were originally unable to commit to the NUREG 0578 completion date.

2. Clarifications requested in or required by Mr. Denton's letter of October 30, 1979.

3. Clarifications or additional information requested by the NRC staff in a conference call on November 16, 1978.

Items are numbered as in NUREG 0578. Only those NUREG sections where specific updating or new information is required are listed.

2.1.1.3.1 PRESSURIZER HEATER POWER SUPPLY

Based on the clarification provided in your letter of October 30, 1979, the qualification review of pressurizer motive and control power interfaces with the emergency buses will be completed by the required date of January 1, 1980 rather than our previous commitment date of March 31, 1980.

At North Anna both backup heater groups are rated at 270 KW rather than 270 and 215 KW as stated in our previous response.

Response to Clarifications of October 30, 1979 Letter

1. Each of the two backup pressurizer heater groups are connected to a separate redundant safety related 480 volt load center in order not to compromise independence between the Class 1E power supplies.

2. Each redundant group of backup pressurizer heaters is supplied from emergency power sources and has the capability to provide the heat required to maintain natural circulation.

3. The emergency power system, with no loss of offsite power, has the capacity to feed the backup heater groups and LOCA required loads. During a loss of power (L.O.P.) which requires the buses to be powered from the Emergency Diesel Generators, the backup heater groups will be automatically tripped. Under the L.O.P. conditions, the backup heater groups, if required, can be manually and selectively connected by an operator who is cognizant of diesel load conditions and plant conditions.

4. The emergency backup heater groups are always powered from a redundant safety related power source which is normally powered by offsite power. Offsite power as stated in the FSAR is the preferred power source. Upon a L.O.P. the heater groups are automatically tripped. The heaters can be manually reloaded onto the emergency onsite power source through a switch on the main control board or if the main control room is uninhabitable, by a switch on the Auxiliary Shutdown Panel in the Emergency Switchgear Room.

5(a) Plant operating procedures will be prepared by January 1, 1980 to provide operating personnel with required knowledge of loads and plant conditions that may occur.

5(b) Pressurizer backup heater groups will not be tripped by an S.I. Signal, only by a L.O.P. signal. See Clarification Item #7 for further delineation.

5(c) Instrumentation to monitor diesel loading is located on the Emergency Diesel Generator panels in the Main Control Room and includes a watt-meter, VAR meter, ammeter, and voltmeter for each diesel generator set. Procedures for operator use to prevent overloading of the diesel generator sets are being developed per Clarification Item 5(a) above.

6. Interfaces for main and control power are protected by Class 1E safety-grade breakers and/or fuses.

7. The pressurizer heater emergency backup groups are non-Class 1E loads being fed from an emergency source. They are automatically shed from the bus for a loss of power accident, but not for a safety injection actuation (SI) signal. When the heaters are shed by a Loss of Power Accident signal they must be reconnected to the bus by manual operator action. When a SI signal is received and offsite power is available, there is no reason to trip the heaters since offsite power has the capability to supply all the bus loads. (See Clarification Item #3.) If there is a SI signal and then a L.O.P. signal, the heater loads are automatically tripped and cannot be re-energized until operator action is taken as described above. In the event a L.O.P. signal has been received and the heaters have been manually reconnected (See Clarification Item #4) and then a SI signal is received, there is no reason to trip the heaters since the Emergency Diesel Generator has the capability to supply all the bus loads.

2.1.1.3.2 POWER SUPPLY FOR PRESSURIZER RELIEF AND BLOCK VALVES AND PRESSURIZER LEVEL INDICATORS

The qualification review of the motive and control power connections to the emergency buses for the PORVs and block valves will be completed by the required date of January 1, 1980, rather than our previous commitment date of March 31, 1980.

Response to Clarifications of October 30, 1979 Letter

1. During reactor power operation, each PORV is opened by energizing two solenoid valves which admit nitrogen to the PORV actuator to open the valve. De-energizing or loss of power to the solenoid valves causes the PORV to close. The respective PORV solenoid valves are powered from separate 125V DC emergency buses.

   Each block valve is motor operated and requires power to open and close. The block valves are powered from separate 480V AC emergency buses. During normal shutdown water solid operation, each PORV is opened by energizing a third solenoid valve which admits nitrogen to the PORV actuator. The PORV closes on de-energizing the third solenoid valve. The PORV third solenoid valve is powered from separate 125V DC emergency buses.

2. The motive and control power for the block valve is being supplied by an emergency power bus that is different from that which supplies the PORV solenoid for water solid operation and the same as the emergency power for the PORV solenoid valves used during reactor power operation. The solenoids controlling the PORV cause the PORV to close on loss of power to the solenoids, therefore, the closure of the PORV paths is assured even if a power source is not available.

3. PORV solenoid valve control power is from 125V DC emergency battery buses and thus will not shift on loss of normal offsite power. Block valves are connected to the emergency buses which are normally fed from the offsite power supply. Failure of the offsite power supply will cause the motive and control power for the block valves to shift automatically to their respective emergency on-site power supplies.

2.1.3.a. DIRECT INDICATION OF VALVE POSITION

In our initial response of October 24, 1979, we committed to install acoustic monitors on the reactor coolant system safety valves. These monitors will be installed on Units 1 and 2 by January 1, 1980. We were able to meet this date by having Babcock and Wilcox Corp. fabricate the system from components available from a loose parts monitoring system originally purchased for North Anna Units 3 & 4.

The alarms on the limit switches of the PORVs required by clarification 2 on page 7 of your October 30, 1979 letter will be installed on North Anna Units 1 and 2 by January 1, 1980.

Response to Clarifications

The direct position indication of each of the two Power Operated Relief Valves (PORV) is presently derived from limit switches. The existing limit switches will be replaced by environmentally qualified limit switches and a common control room alarm for each unit will be added to alarm when the PORVs or the pressurizer safety valves are not closed. The environmentally qualified limit switches will be installed during the first scheduled outage of sufficient duration after material delivery.

The valve acoustic monitoring system will provide two channels for each pressurizer safety valve, one for indication, the other for backup. One channel is actively connected to the monitoring equipment while the second channel is terminated in a plug passively mounted in the valve monitor cabinet. The active channel may be unplugged and replaced with the passive channel at the valve monitor cabinet mounted in the control room should the need arise. Additionally, backup indication of valve position is provided by temperature detectors in the piping downstream of the safety valves.

Each Pressurizer Safety Valve will have a valve position alarm indicator mounted on the main control board. In addition, a common control room alarm for each unit will sound when any Pressurizer Safety Valve or PORV is not in the fully closed position.

The power supply to the valve monitoring system will be derived from vital buses in Units 1 & 2 with an automatic transfer on the loss of one vital bus.

The valve monitoring cabinet will be seismically supported in the control room. The valve acoustical monitoring system will be seismically and environmentally qualified by the vendor (Babcock and Wilcox Co.). The Vendor is establishing a test program to bound the worst case environmental and seismic conditions of its application. The purchasing utilities will submit the worst case conditions early in December and a test program will be established early in 1980. The completion date for the qualification will be provided when the extent of the test program has been established.

2.1.3.b INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

Subcooling Meter

The subcooling meter may not be installed by January 1, 1980. We have committed to a fully qualified system that meets the requirements of NUREG 0578 and are making a concentrated effort to have it installed and operational by January 1, 1980.

In an effort to ensure that the system fully meets the NUREG 0578 requirements, and to ensure we obtain the best system currently available, a purchase order was not placed until after a thorough review of vendor proposals and the clarifications in your letter of October 30, 1979. An order was placed with Westinghouse Electric Corporation for their subcooling meter on November 16, 1979. Their estimated equipment delivery would support installation by January 1, 1980. However, Westinghouse will not be able to finalize their interface requirements until the first week of December. The engineering and design required to ensure that the installation of the system does not adversely impact the reactor protection or engineered safety system was initiated in October but will extend beyond January 1, 1980. In addition, until particular interface requirements are identified by Westinghouse, we are unable to purchase some long lead time items such as isolation devices.

The subcooling meter will be installed on North Anna Unit 1 following completion of design and receipt of all materials. Installation of the subcooling meter on Unit 2 will be prior to initial criticality.

In an effort to install the subcooling meter as soon as possible, the system may not be fully qualified to Class 1E application upon installation. However, Westinghouse has assured us that based on previous testing, the equipment can be qualified to these standards. In the short term the hot leg and cold leg temperatures, twenty existing incore thermocouples and the Reactor Coolant System pressure and pressurizer pressure inputs may be provided by paralleling existing computer input terminations for the control loops and wiring them to each subcooling meter electronics.

It is the opinion of Westinghouse that the system as permanently installed will meet all the requirements of NUREG 0578, item 2.1.3.b.1. The detailed information requested in your October 30, 1979 letter will be presented as it is available. A brief technical description of the system is presented below.

> The core subcooling monitor utilizes inputs from the existing hot leg RTDs, reactor coolant system pressures, and 20 selected incore thermocouples. A microprocessor is employed to calculate saturation temperature for the existing reactor coolant system pressure and determine the margin to saturation based on the various temperature inputs. The information display consists of a main control board indication of the margin to saturation and expanded information at the electronics drawer which will be located with the existing plant protection and control equipment.

The system is divided into two redundant channels, each accepting inputs from two hot leg RTDs and 10 thermocouples. The redundant analog meters mounted on the main control board have nonlinear scales, expanded near saturation, with the range extended to the limits of the incore thermocouples. In addition to the numerical indication, the meter faces are color coded to increase visual acuity for the operators.

The auctioneered low reactor coolant system pressure is utilized by the microprocessor to calculate the saturation temperature for the existing system pressure. By subtracting the auctioneered high incore thermocouple signal from the calculated saturation temperature, the current margin to saturation is calculated. This information is then displayed on the main control board meters.

The margin to saturation from the auctioneered high hot leg RTD is also available on the main control board meters by means of a two position switch to be located on the main board.

Two levels of alarm are provided; the first to indicate the development of off-normal conditions, the second the approach to loss-of-core subcooling. The actual set point for either alarm is controlled by the microprocessor and is easily modified by keyboard entries at the main processing unit.

Expanded information is displayed on the front of the electronics drawer to enable the operator to determine core cooling conditions. Each of the redundant microprocessors will utilize the computed saturation temperature for comparison with the following plant parameters: individual loop wide range T_{hot}, auctioneered high range T_{hot}, average incore thermocouple, auctioneered high incore thermocouple. In addition, the following information will be available for display: the temperature difference between the average of the incore thermocouples and individual loop T_{hot} and time to a loss of subcooling based on the rate of change of reactor coolant pressure and temperature.

In addition to the above digital displays, a graphic, color, 15 lamp display of the incore thermocouples is provided on the electronics drawer. The core is divided into five regions, four peripheral and one central, with inputs from two thermocouples used to generate the display for each region. Each region has three lamps, green, yellow and red. Two sets of three lamps are also provided for the hot leg RTDs input into each drawer. If all temperatures are sufficiently below saturation, the lamps will be green. If any individual temperature reaches an off-normal condition, the lamp display changes to yellow and alarm contacts for the

plant annunciator are closed. Should conditions continue to degrade to the preset margin to saturation, the lamp display changes to red and an additional set of alarm contacts are closed for the plant annunciator.

Until this system is operational other mechanisms have been provided to the operators to enable them to immediately assess the primary coolant margin to saturation conditions.

Changes to emergency procedures have been made to emphasize the need to insure adequate coolant flow and to insure that the reactor coolant temperature and pressure is maintained or immediately restored to achieve an appropriate margin to saturation. All licensed reactor operators and current trainees have received special instruction in the TMI type incident with particular emphasis on the use of existing instrumentation to determine core conditions.

The plant computer will be used to monitor plant saturation conditions. The operator can initiate a trend printout of the system saturation temperature and one or more of the critical reactor coolant system temperatures.

A saturation curve has been posted on the control board and provided in current procedures. This curve, combined with the nearby indications of reactor coolant system temperatures and pressures, enables the operator to quickly determine the systems margin to saturation.

As explained in our earlier response of October 24, 1979, Westinghouse has developed a listing of all instruments required for an operator to diagnose the type of plant event, take the necessary manual actions, and to monitor critical plant conditions. North Anna currently has all of the recommended instrumentation.

We believe that the above listed instrumentation, procedures and operator training will provide assurance that the operator can adequately assess plant saturation conditions until installation of the subcooling meter is completed.

Additional Instrumentation

Regarding instrumentation to monitor reactor vessel water level, we have not at this time identified a reactor vessel level measurement technology which is unambiguous under all conditions. We are continuing our evaluation of potential reactor vessel level measurement techniques. As soon as a suitable technology is identified, we will initiate design and procurement activities. We will make a concentrated effort to have the selection and installation of the level instruments completed by January 1, 1981.

2.1.4 DIVERSE CONTAINMENT ISOLATION

Our initial response of October 24, 1979 provided a brief description of the North Anna containment isolation scheme and a listing of the "essential" and "non-essential" containment penetrations. In a conference call on November 16, 1979, members of the NRC staff requested a brief summary of the bases for designation of a system as "essential" or "non-essential". These bases are as follows:

Essential systems are divided into two categories: 1) systems required to mitigate the consequences of various types of accidents, and 2) systems which are required to maintain the operation of critical systems and functions. Included in category 1 are the engineered safeguards features including the containment and recirculation spray systems, the safety injection systems, and the service water system which provides cooling to the recirculation spray heat exchangers. Category 2 includes auxiliary feedwater, and component cooling required for reactor coolant pump operation, containment air cooling, and residual heat removal.

Non-essential systems include all other systems not required for the above functions.

2.1.5.a DEDICATED HYDROGEN CONTROL PENETRATIONS

Response to Clarifications

The post-accident hydrogen recombiners take a suction on the containment through the same containment penetration which is used for the suction of the containment vacuum pumps (See FSAR Figure 6.2-98). The recombiner discharges back to the containment through its own dedicated penetration. The containment isolation for these penetrations meets the redundancy and single failure criteria. Deviations from General Design Criteria 54 and 56 are documented in Section 6 of the FSAR (use of check valves and no valve inside containment on the suction line).

The two inch hydrogen recombiner lines tie into the two inch containment vacuum lines downstream of the containment isolation valves. The containment vacuum system is isolated from the containment atmosphere by a manual valve (not subject to single active failure by spurious motor-operator movement).

The hydrogen recombiner discharges to the containment through a two inch manual valve (normally locked shut and not subject to a single failure by spurious motor-operator movement). The plant shielding review will determine whether modifications are required to allow personnel access for operation of the valve.

The combined containment suction line and the dedicated hydrogen recombiner discharge line is sized such that the flow requirements for the use of the recombiner system are satisfied. The system size calculations and the Units 1 and 2 preoperational test results confirmed that the flow requirements of the hydrogen recombiner are satisfied.

2.1.5.c CAPABILITY TO INSTALL HYDROGEN RECOMBINER AT EACH LIGHT WATER NUCLEAR POWER PLANT

In the conference call of November 16, 1979, members of the NRC staff requested an estimate of the time required to place the hydrogen recombiners in operation. The recombiners can be aligned and operable within approximately 8 hours, the majority of this time being for recombiner warm-up.

2.1.6.a INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIALS

In our response of October 24, 1979, we stated that a review was in progress to identify those systems outside containment which, in addition to mitigation systems, are likely to contain highly radioactive fluids during or following an accident. This review is now complete and forms the basis for our immediate and continuing leak reduction programs. By January 1, 1980 on Unit 1 and prior to initial criticality on Unit 2, we will complete the following:

1. All outside containment systems which are likely to contain highly radioactive fluids during or following an accident will

be inspected and tested to identify any leaks. All practical leak reduction measures will be applied to reduce leakage from these systems.

2. Leakage rates from all such systems will be measured or estimated during operation or during a simulation of operating conditions. A summary of the findings will be forwarded.

3. A long term leakage measurement and preventive maintenance program will be developed. The leak test interval will be no greater than a refueling cycle. Details of the program including a listing of the systems covered and systems not covered will be forwarded.

2.1.6.b PLANT SHIELDING REVIEW

Shielding reviews for Surry Units 1 and 2 and North Anna Units 1 and 2 are in progress. Both reviews are being performed by the same review team. Due to similarities in design between the two stations, a large portion of the review work has been applicable to both stations. As a result, we have been able to improve on our original schedule. The radiation and shielding design review for North Anna Units 1 and 2 will be completed by January 1, 1980.

2.1.7.a AUTO INITIATION OF THE AUXILIARY FEEDWATER SYSTEM

The modifications on North Anna Units 1 and 2 to install alarms on the pressure control valves position indication identified in our responses to your letter of September 13, 1979 will be installed by January 1, 1980.

Response to Clarifications

Instrument air is required for the operation of the control valves in the auxiliary feedwater system. The instrument air compressor is presently powered from an emergency bus. Additionally air bottles are provided at each valve to provide motive power in the event of loss of instrument air.

2.1.7.b AUXILIARY FEEDWATER FLOW INDICATION TO STEAM GENERATORS

Response to Clarification of October 30, 1979 Letter

The Auxiliary Feedwater Flow indication for each steam generator is safety grade and powered from vital buses. At present, the Auxiliary Feedwater Flow indication for each steam generator is not designed to accommodate a single failure. However, safety grade steam generator level is provided as a backup indication.

As part of the modifications identified in our response to your September 13, 1979 letter, the power supplies for auxiliary feedwater flow transmitters (FT-FW100A, B, C and FT-FW200A, B, C) were moved from a semi-vital power supply to three separate vital power supplies. Steam generators B and C each have different vital power supplies for their respective auxiliary feedwater flow and steam generator level indication loops. Steam generator A has the identical vital power supply for both auxiliary feedwater flow and steam generator level indication loops. Therefore, an additional modification is required and will be made by January 1, 1980 to place the steam generator A indication loops and auxiliary feedwater indication on separate vital power supplies for North Anna Units 1 and 2.

The auxiliary feedwater flow indication is testable from the transmitter back to the indicator.

The Auxiliary Feedwater Flow indication satisfies the requirement of ±10 percent accuracy.

2.1.8.a IMPROVED POST-ACCIDENT SAMPLING CAPABILITY

The clarifications presented in your letter of October 30, 1979 resolved many of our questions concerning specific sampling requirements. Therefore, a review of our sampling capability as well as a conceptual design of an improved sampling capability will be provided for North Anna Units 1 and 2 by January 1, 1980.

2.1.8.b INCREASED RANGE OF RADIATION MONITORS

Response to Clarifications

An evaluation of interim methods of estimating high level noble gas releases which meets the requirements of table 2.1.8.b is in progress for monitored effluent locations and other potential release points. The interim methods will employ the use of dedicated portable and/or rigidly mounted detectors at potential release points which will be sufficiently shielded from background and will be collimated, if required, to obtain readings within the range required. Conversion of the measured dose rate into a concentration will be performed using standard volume source calculations and predetermined conversion factors. The capability will exist to obtain a radiation reading every 15 minutes, either by remote read-out or by phone communication with an operator. The power source will be arranged such that alternate AC power will be available, if required, or a DC source will be available for seven consecutive days. Procedures will be developed for the removal and analysis of the radioiodine particulate media. Procedures also will be developed for dissemination of information, calibration of equipment and minimizing occupational exposures.

Since these short term requirements were not identified until your October 30, 1979 letter, a firm schedule for implementation is not yet available. However, the majority of the methods will probably use currently available equipment and any delay in installation due to equipment delivery will be a minimum. Procedures will be developed, approved and implemented as soon as the methods are finalized. Therefore, some of the procedures may not be in effect on January 1, 1980. It is expected that all procedures will be in effect by February 1, 1980.

In our initial response to the long term requirements, we identified our concerns that commercially available monitors may not be available for the extended ranges required. We still believe this to be the case but will make every effort to meet the requirements for inplant monitor capabilities by January 1, 1981.

2.1.8.c IMPROVED IN-PLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

Provisions to determine radioiodine concentration in the presence of noble gas using a single channel analyzer currently exists at North Anna.

The adequacy of the silver zeolite sampling cartridges under accident conditions will be documented by January 1, 1980. If necessary provisions to flush the cartridge with clean gas will be provided by January 1, 1981. Provisions for removal of the cartridge to a low background, low contamination area for analysis will be provided by January 1, 1981.

2.1.9 TRANSIENT AND ACCIDENT ANALYSIS

Containment Pressure Indication
Response to Clarification of October 30, 1979 Letter

The present design for containment pressure indication has four channels of protection grade transmitters with continuous indication in the control room. Their power supply is from Vital Buses I, II, III and IV. The lower range requirement (minus 6 psig) is met. The upper range of indication will be increased to approximately 180 psia. The containment pressure indications will be upgraded to meet the intent of the design and qualification of Regulatory Guide 1.97 including qualifications, redundancy and testability. These modifications will be accomplished by the required date of January 1, 1981.

Our method of meeting the intent of Regulatory Guide (RG) 1.97 for this and subsequent items utilizes plant licensing criteria as follows:

A. RG 1.97 position C4 imposes RG 1.80 Qualification Requirements. The qualification of existing instrumentation is in compliance with existing plant licensing/FSAR commitments. Qualification requirements of new equipment will address the requirements of RG 1.89. As a minimum, the new equipment will meet IEEE 323-1971 and IEEE 344-1976.

B. RG 1.97 position C7 cites RG 1.75 criteria. Our redundant monitoring channels will be fed from Class 1E power and separated in accordance with our licensed plant separation criteria.

Containment Hydrogen Indication
Response to Clarification of October 30, 1979 Letter

A continuous indication of hydrogen concentration in the containment atmosphere will be provided in the control room. Modifications are required to provide remote isolation valves and control room readout. Measurement capability is presently provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

The existing hydrogen analyzers are being evaluated to determine if they need to be modified to meet the intent of Regulatory Guide 1.97. One method being evaluated is to remove the electronics from the hydrogen analyzer package and mount it in an environmentally less severe location (away from the sample line and sensor). The sensor could be upgraded to qualify it to radiation requirements, if required. If the evaluation concludes that the existing hydrogen analyzers could not meet the intent of Regulatory Guide 1.97, new hydrogen analyzers will be purchased which will meet the requirements of Regulatory Guide 1.97. These modifications will be accomplished by January 1, 1981.

Containment Water Level Indication
Response to Clarification of October 30, 1979 Letter

The existing instrumentation meets the requirements for continuous monitoring of containment water levels. Narrow range indication is

provided by redundant level transmitters located in the small containment sump next to the primary shield wall. These instruments have a 12 inch range. The wide range indication is provided by redundant level transmitters located in the containment safeguards sump. These transmitters have a range of 10 feet above the containment floor which is equivalent to about 620,000 gallons and encompass the maximum calculated water level during an accident. Indication from all four transmitters is continuously displayed in the control room.

The transmitters have been environmentally qualified to North Anna Unit 1 and 2 LOCA conditions, as well as seismically qualified. The indication channels meet the intent of the design and qualification provisions of Regulatory Guide 1.97 for the wide range and 1.89 for the narrow range. Functioning of the narrow range transmitters may be observed during plant operation as the containment sump fills and is pumped out.

Reactor Coolant System Vent

Design of reactor coolant system and reactor vessel high point vents is underway. However, there are many questions concerning proper design and operation that remain unresolved. By Januay 1, 1980, we will present a conceptual design of the RCS Vent System. We will make every effort to install the system by January 1, 1981. However, the system cannot be installed until a proper and thorough design review has been performed and adequate procedures are developed for its operation.

2.2.1.b SHIFT TECHNICAL ADVISOR

In our initial response of October 24, 1979 we indicated that the operating experience assessment function would be performed by the Operating Supervisor. In discussions with the staff on November 16, 1979, we agreed to designate an individual other than the Operating Supervisor, whose primary responsibility would be the operating experience assessment function. An individual with the appropriate operating experience and technical knowledge will be assigned to this position by January 1, 1980.

2.2.2.b ONSITE TECHNICAL SUPPORT CENTER

Items 1.A through 1.G of your October 30, 1979 clarification will be completed by January 1, 1980. Our schedule for upgrading the Technical Support Center, as required by items 2-10 of your clarification, will be provided in our response to item 1.G.