Omaha Public Power District

1623 HARNEY · OMAHA, NEBRASKA 68102 · TELEPHONE 536-4000 AREA CODE 402

October 25, 1979

Mr. Darrell G. Eisenhut
Acting Director
Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Reference: Docket No. 50-285

Dear Mr. Eisenhut:

The Omaha Public Power District received the Commission's letter, dated September 13, 1979, on September 28, 1979, in regard to followup actions resulting from the NRC Staff reviews regarding the Three Mile Island Unit 2 accident. The District was requested to provide a commitment to meet the requirements of NUREG-0578, according to the implementation schedule of Enclosure 6 to the letter, and to provide a commitment to comply with the requirement of Enclosure 7, in accordance with the implementation schedules shown in Enclosure 8 in regard to emergency preparedness. Commitments to comply with Enclosure 6 are provided herein by attachment. Modifications, identified in the attachment, which require a shutdown to complete and are designated to be implemented by January 1, 1980, or January 1, 1981, in NUREG-0578, will be completed during the 1980 or 1981 refueling outage, as applicable. The 1980 refueling outage is scheduled to commence January, 1980, and the 1981 refueling outage is scheduled to commence March, 1981.

In regard to Enclosure 8 of the letter, the District will upgrade the Fort Calhoun Station Emergency Plan to meet the requirements of Regulatory Guide 1.101 prior to January 1, 1980. Items 2, 3, 4, 5, and 6 of the enclosure will be addressed in the revised Emergency Plan, as required by the Commission's acceptance criteria for revised Emergency Plans. The specific implementation plans and schedules will meet the requirements of NUREG-0578, where applicable, or will be resolved in reviews performed by the Staff's emergency preparedness task force.

As discussed at the various NRC topical meetings and NRC/industry users group meetings, the implementation schedules identified in NUREG-0578 may be difficult to meet, in some instances, because equipment for modifications is not immediately available. The attachment to this letter attempts to identify those tasks where equipment availability may cause scheduling delays. It must be recognized that equipment delivery times are difficult to predict and, therefore, problems may arise which are unforeseen at present. Notwithstanding

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these uncertainties, the District will continue to make every reasonable effort to comply with NUREG-0578 schedules.

Sincerely,

W. C. Jones
Division Manager
Production Operations

WCJ/KJM/BJH:jmm

Attach.

cc: LeBoeuf, Lamb, Leiby & MacRae
1333 New Hampshire Avenue, N. W.
Washington, D. C. 20036

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ATTACHMENT

2.1.1 EMERGENCY POWER SUPPLY REQUIREMENT

Pressurizer Heaters

In regard to providing emergency power for pressurizer heaters, the functional requirements for the specified pressurizer sub-systems needed in order to provide sufficient availability of the pressurizer for pressure control, in the event of a loss of offsite power, are being developed by the C-E Owners Group. These requirements will determine the additional load to be added to the emergency buses. If these loads are within the capability of the emergency buses, the redundant emergency power requirements for pressurizer heaters will be met at the Fort Calhoun Station, in conformance with safety grade requirements appropriate for this unit, no later than March 1980 (after our next refueling outage, scheduled to commence in January 1980). In addition, appropriate procedures will be developed and operator training initiated by then.

PORV and Block Valve Power Supply

The Fort Calhoun Station Power Operated Relief Valve and Block Valve System consists of two 480 V solenoid operated power operated relief valves and their associated 480 V motor operated block valves. As presently designed, PCV-102-1 (PORV) and HCV-151 (block valve) are powered from 480 V 3-phase 60 Hz motor control center (MCC) 3C1 and MCC 3B1, respectively; PCV-102-2 (PORV) and HCV-150 (block valve) are powered from MCC 4B1 and MCC 4A1, respectively. MCC 3C1 and MCC 3B1 are powered from emergency diesel generator No. 1 (Train A) and MCC 4B1 and MCC 4A1 are powered from emergency diesel generator No. 2 (Train B). The four MCC's discussed above are part of the station's Engineered Safety Features System.

As installed, the cabling for the PORV's is safety grade, meeting all separation and segregation criteria. PCV-102-1 is installed as safety related cable EA channel, and PCV-102-2 is installed as safety related cable EB. Therefore, no modifications are required for the PORV's. The block valves are presently installed as control grade cabling supplied from the Engineered Safety Features supplied MCC, previously listed. HCV-150 is installed as B channel and HCV 151 is installed as A channel. As installed, these cables are of the same material as the safety cables, but do not meet the more stringent separation requirements of the safety related installation criteria.

In order to meet the requirements of NUREG-0578, the following changes will be implemented during the 1980 refueling outage scheduled to commence January 1980.

1. In order to provide complete redundancy of the block valves to the PORV's, the power supplies will be changed as follows.

   HCV-151 to diesel generator No. 2
   HCV-150 to diesel generator No. 1

2.1.1 EMERGENCY POWER SUPPLY REQUIREMENT (Continued)

PORV and Block Valve Power Supply (Continued)

2. The cabling for the block valves will be upgraded to "E" (safety grade) criteria.

   HCV-150 to Train A (EA or EC) cable
   HCV-151 to Train B (EB or ED) cable

Pressurizer Level

The present pressurizer level system (calibrated for reactor coolant temperature and pressure) consists of two transmitters, one powered from each of the station batteries (via an instrument inverter). The cables to the transmitters are LOCA qualified; the transmitters are control grade.

NUREG-0578 requires that the pressurizer level measurement system be vital bus supplied and have safety grade interface. The power supply system at Fort Calhoun Station meets this requirement and, therefore, no modifications are required.

To insure availability of the level monitoring system, the transmitters will be upgraded to LOCA qualified transmitters.

This Modification will be completed during the 1980 refueling outage scheduled to commence in January 1980. It is, however, dependent on transmitter delivery (the manufacturer's present quotation is 18 weeks).

2.1.2 RELIEF AND SAFETY VALVE TESTING

A program for testing power operated relief valves (PORV's) and safety valves (SV's) used for primary system pressure control under design bases operating conditions is being developed by the C-E Owners Group. This program includes definition of test conditions and qualification requirements for all specified valves in operating reactors designed by Combustion Enginering (C-E). The results of this program will be made available to the generic efforts being undertaken by the industry (through, for example, the Electric Power Research Institute, EPRI, and the Nuclear Safety Analysis Center, NSAC) no later than Janaury 1, 1980. These results will also be available for discussion with the NRC staff to establish Generic Resolutions no later than January 1, 1980. The District will comply with the schedule for completion of the test program which is agreed to during these Generic Resolutions meetings.

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2.1.3.a DIRECT INDICATION OF VALVE POSITION CATEGORY A SAFETY GRADE

The Fort Calhoun primary system relief and safety valves will be provided with a position indication in the control room derived from a reliable valve position indication device or a reliable indication of flow in the discharge pipe no later than March 1980 (by the end of our 1980 refueling outage scheduled to commence January 1980). The functional requirements and conceptual design for this position indication are being developed by the District. Presently, two vendor proposals are being evaluated. The design will include specification of safety class, the type of sensor with method of indication, and logic diagrams.

2.1.3.b INSTRUMENTATION FOR INADEQUATE CORE COOLING

The procedures to be used by an operator to recognize inadequate core cooling will be developed based on analyses being performed as required by Item 2.1.9, Transient & Accident Analysis, Analysis of Inadequate Core Cooling. The guidelines for the procedures are being developed by the C-E Owners Group and will be available for discussions with the NRC staff to establish Generic Resolutions no later than January 1, 1980. If the analyses or the guidelines indicate the need for the design of new instrumentation, the design of such instrumentation will be made available for discussions with the NRC staff to establish Generic Resolutions on the schedule established by the discussions on the analyses and guidelines.

In addition, certain present non-LOCA qualified instrumentation at Fort Calhoun, which may be useful in detecting inadequate core cooling, will be replaced with LOCA qualified instrumentation during the next refueling outage, scheduled for January 1980, pending availability of components.

The functional requirements and a conceptual design for a subcooled margin monitor are being developed by the District. This effort includes evaluation of vendor proposals, selection of temperature and pressure inputs, requirements for safety grade, and assessment of delivery schedule of safety grade instrumentation. Vendors of safety grade instrumentation have indicated 18 to 20 weeks delivery time. The requirements for a subcooled margin monitor will be met at the Fort Calhoun Station no later than March 1980 (at the end of our next refueling outage, scheduled to commence January 1980), pending delivery of safety grade instrumentation. The functional requirements will be available for implementation review.

The functional requirements and a conceptual design for a reactor vessel level measurement device are being developed by the C-E Owners Group. This effort includes a survey of currently available technology and assessment of the feasibility of various alternatives. If required by the Generic Resolutions discussions with the NRC staff, the functional requirements, conceptual design, implementation procedures, and analyses used to derive procedures, will be submitted for Proposal Review by the NRC staff prior to implementation. The installation schedule for such a device, should it be deemed necessary, will be established during the Proposal Review.

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2.1.4 CONTAINMENT ISOLATION

The Fort Calhoun Station Unit No. 1 reactor containment building is provided with an automatic actuation system which operates containment penetration isolation valves (both to the open and to the closed position) to mitigate the consequences of an accident. The automatic systems are the Containment Isolation Actuation Signal (CIAS) and the Ventilation Isolation Actuation Signal (VIAS). It should be noted that VIAS is redundant to CIAS in that it also generates a valve closure signal for certain isolation valves closed by CIAS, specifically, the containment purge and relief lines.

The CIAS, as well as Safety Injection Actuation Signal (SIAS), is generated if either a Pressurizer Pressure Low Signal (PPLS) or a Containment Pressure High Signal (CPHS) is generated by an accident condition in the primary system and/or containment building. The actual mechanism of isolation is designed to meet single failure criteria. The CIAS system consists of two redundant isolation channels (or trains) which are actuated by sensors, in a two-out-of-four logic, monitoring the primary system and containment pressure. When the actuation logic is satisfied, a signal is then generated which actuates the PPLS or CPHS "86" lockout relay which mechanically seals in the accident signal. The accident signal in turn actuates the emergency core cooling system and CIAS via an "86" lockout relay which mechanically seals in the CIAS control function. In addition, the Fort Calhoun Station Engineered Safety Feature System is provided with an additional control relay whereby Channel A control may "reach across" and actuate Channel B equipment and vice versa. This provides increased availability of safety equipment. The CIAS train separation is maintained at the line isolation valves; channel A will operate the A valve and B the B valve (exceptions are dealt with later in the discussion).

When the "86" CIAS relay is actuated, relays in the CIAS panel AI-43A and AI-43B are de-energized, causing the isolation valves to assume their accident position. In general, this electrically deenergizes the solenoid and the valve diaphragm goes to zero pressure. In order to provide for special operation and maximum safety, exceptions to the above description are discussed as follows.

1. The instrument air header valve PCV-1849 closes on low air pressure (70 psig). With a normal operating pressure above accident pressure, the air header is left in service unless low air pressure is detected. The availability of air enhances reactor shutdown. It must be emphasized that air failure will in no way inhibit shutdown. PCV-1849 fails open.

2. The hydrogen purge system HCV-881, 882, 883A/B, 884A/B is required for operation in a post-accident condition. The supply and exhaust are equipped with automatic valves inside containment only. These valves are closed by CIAS, but fail open on loss of control power or air. The valve isolation function is backed up by redundant locked closed manual valves. The sample valves have a similar arrangement; however, the automatic valves outside containment fail closed and are closed by CIAS. Override switches are provided to override the CIAS function.

3. The auxiliary feedwater system (which is not directly connected to the RCS or containment atmosphere) HCV-1107A/B and 1108A/B are opened by CIAS and fail open. This permits decay heat removal. Override switches are provided for post-accident valve positioning.

4. The main feedwater penetration isolation valves HCV-1385 and 1386 are motor operated valves which close on CIAS. These valves fail "as is," but are connected to a vital 480 volt bus to ensure closing. In addition, there are check valves in the system as redundant valves. This system is not directly connected to the RCS or containment atmosphere.

5. The containment air cooling units' cooling water supply valves HCV-400A/C, B/D, 401A/C, B/D, 402A/C, B/D, and 403 A/C, B/D are opened by CIAS and fail open.

6. The reactor coolant pump lube oil cooler and seal cooler supply and return line HCV-438A, B, C, D close on CIAS. The valves inside containment (A and C) fail open. The valves have override switches to open on after CIAS to reinstate cooling water. The failure of valves or closure on CIAS will not trip the reactor coolant pump motors.

The remaining CIAS closed valves serve to isolate containment and prevent the release of any radioactive material to the atmosphere. The design of the individual valves control circuit is such that it "remembers" its accident position and remains in that position when the accident position signal is removed. The valve can then be positioned as desired by the operator. An additional design feature of the valve position circuit is that, if an accident signal is present, those circuits without an override feature cannot be repositioned by control switch.

The remaining penetrations not actuated by CIAS are those which are associated with shutdown or a different type accident. These systems are listed below.

1. Safety injection - opens under accident signal

2.1.4. CONTAINMENT ISOLATION (Continued)

2. Containment spray - opens on CPHS and PPLS

3. Shutdown cooling - locked closed

4. Containment sump recirculation - part of ECCS actuates on RAS

5. Main steam isolation valves - closed upon SGLS (Steam Generator Low Signal) and CPHS (Containment Pressure High Signal) part of closed system not in contact with RCS or containment atmosphere.

The design requirements of NUREG-0578 are presently satisfied, and no changes to the initiation scheme or to the circuit design are necessary. The deletion of HCV-438 A, B, C, and D from CIAS closure is under evaluation; but this will not affect the isolation requirements of NUREG-0578. Essential systems, identified above, are those which are not isolated on CIAS. All non-essential systems are isolated. An identification table of essential and non-essential systems will be submitted to the Commission by January 1, 1980.

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2.1.5.a DEDICATED PENETRATIONS FOR EXTERNAL RECOMBINERS OR POST-ACCIDENT PURGE SYSTEMS

The hydrogen purge system at the Fort Calhoun Station is designed to provide hydrogen removal capacity required by Reg. Guide 1.7, which defines a concentration limit for hydrogen accumulation following a loss-of-coolant accident; and to meet General Design Criterion 41 of Appendix A to 10 CFR 50. The performance characteristics of the filters are based upon the source terms defined under Table 1 of Reg. Guide 1.7.

The hydrogen purge system is part of the containment ventilation system, but it is completely independent of the containment purge system. Two independent penetrations are dedicated to the hydrogen purge system in order to satisfy redundancy requirements. The system consists of two purge units; each with its own 250 cfm positive displacement blower, inlet and outlet piping, and isolation valves. The hydrogen purge system filters are conservatively assumed to have 90% efficiency. A diagram of the system is attached.

The hydrogen purge system is designed to be used when the containment hydrogen concentration exceeds 3 volume percent. Prior to initiation of hydrogen purging, the containment air cooling and filtering units will have substantially reduced the activity and humidity levels in the containment atmosphere. The hydrogen purge system is manually operated and is normally isolated from the containment by locked-closed valves. A single failure analysis of all active components of the system shows that the failure of any component will not prevent the system from fulfilling its design function. The equipment inside the containment and outside the containment, up to and including the isolation valves, is Safety Class 2. The equipment downstream of the isolation valves is Safety Class 3.

Conclusions:

The post-accident hydrogen purge system at the Fort Calhoun Station is:

1. Designed to meet the redundancy and single failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR 50.

2. Sized to satisfy the flow requirements for the control of hydrogen gas concentrations inside the containment.

Therefore, the present system satisfies the requirements of NUREG-0578 and no modifications are planned.

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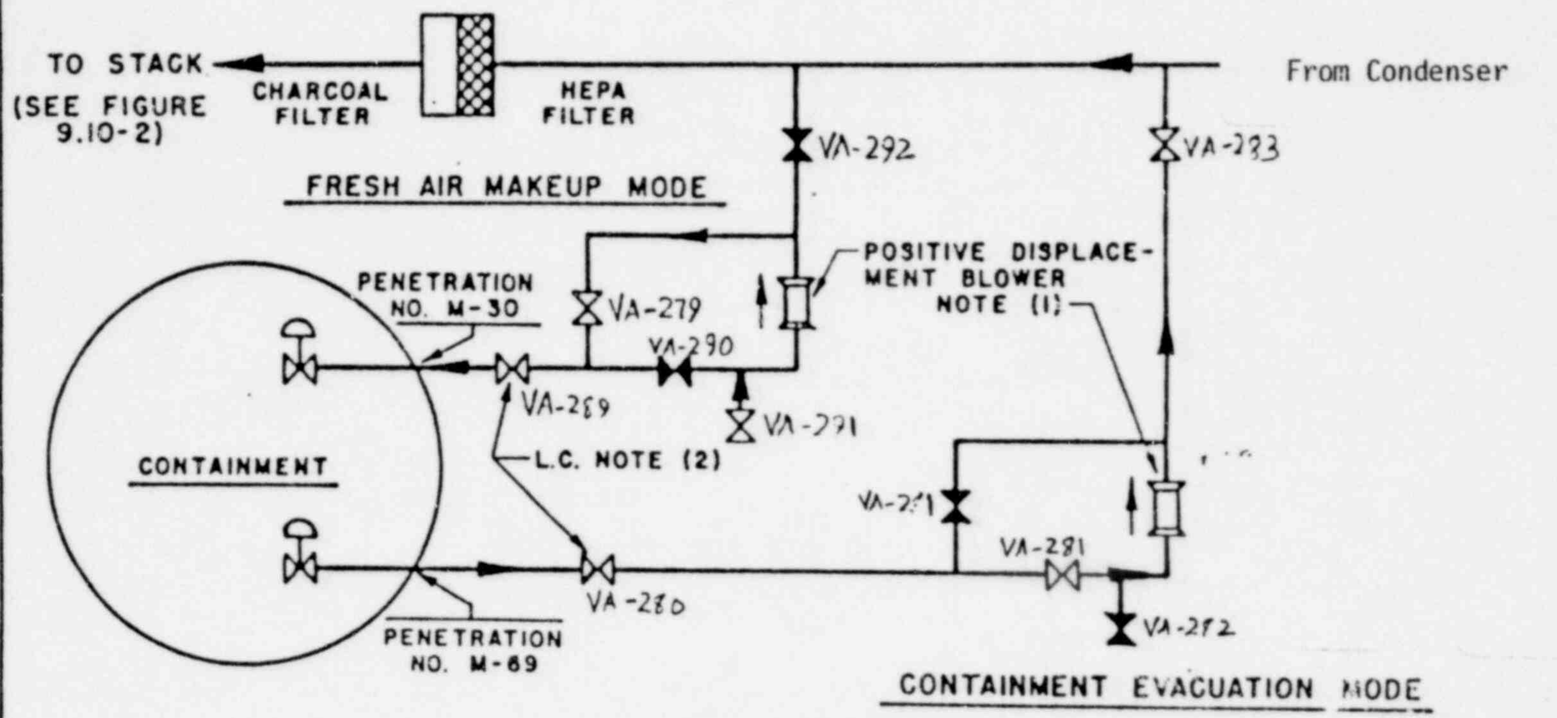
Omaha Public Power District
Fort Calhoun Station
Unit No. 1

Containment Hydrogen
Purge System



NOTES:

(1) EACH BLOWER IS CAPABLE OF TRANSPORTING 250 CFM IN EITHER THE FRESH AIR MAKE-UP MODE OR THE CONTAINMENT EVACUATION MODE

(2) NORMALLY LOCKED CLOSED, OPEN DURING PURGING ONLY

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

OPPD will implement a series of surveillance tests to inspect auxiliary systems for leakage, quantify and evaluate any leaks that are found to the extent possible, and initiate appropriate corrective actions. Results of the evaluation will be reported to the NRC. This surveillance program will generally be applied to the following systems.

1. CVCS charging and letdown system.

2. SI, RHR, and ECCS including LPSI, HPSI, and containment spray pumping systems, containment recirculation piping and related pumps, minimum recirculation piping, and shutdown cooling piping.

3. Portions of the waste disposal system including containment sump piping, spent regenerative tanks and piping, neutralization tank and piping, SI and CVCS valve leakoff piping, auxiliary building sump tank and piping, monitor tanks inlet piping, waste filters, gas stripper, waste evaporator and piping, monitor tanks and recirculation piping, waste evaporator to drumming piping, concentrate tanks, waste holdup tanks, spent resin storage and associated piping and pumps, and the gaseous waste disposal system.

4. Sample line piping designated Class 2; portions not designated Class 2, including and downstream of the total gas analyzer and reactor coolant loop sampling system, and the steam generator blowdown sampling system piping.

The surveillance tests will be performed under normal operating pressure once per fuel cycle. The basic acceptance criterion shall be that of the S75 Addenda, ASME XI, Paragraph IWA-5250, Corrective Measures, "If leakages (other than normal controlled leakages) are detected during the performance of a system pressure test, the source of the leakage shall be located, and the area shall be examined to the extent necessary to establish the requirements for corrective action. Repairs or replacement of components shall be performed in accordance with the rules of IWA-4000."

The procedures will be prepared and the program will be started prior to January 1, 1980. Full implementation will occur after startup from refueling for Cycle 6 in March 1980.

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2.1.6.b PLANT SHIELDING REVIEW

The District will perform a radiation and shielding design review of spaces around systems that may, as a result of an accident, contain highly radioactive materials. This design review will be performed in accordance with the requirements of NUREG-0578, and this review will provide information to be used in any modification required for access to equipment required for operation in the event of an accident. The ability to operate, not maintain, the equipment will be the basis for any modification.

The review will consider degradation of equipment due to radiation during the post-accident condition.

Implementation of plant modification will be completed prior to heatup following our 1981 refueling outage, pending availability of equipment. The design review will be completed by January 1, 1980.

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2.1.7.a AUTO INITIATION OF THE AUXILIARY FEEDWATER SYSTEM

The functional requirements and designs for both control and safety grade systems for automatic initiation of auxiliary feedwater, are being developed by the District. The functional requirements include consideration of power supply reliability, testability, failure modes, monitoring of operation, and automatic/manual initiation. The control requirements can be met for the steam turbine-driven pump, but the motor driven pump will require further investigation into onsite power supply capability. This effort will include a review of loads to be added to the emergency bus and additional loads to be shed so as to make capacity available in the automatic sequence. The design for the control grade system will be submitted for proposed review by December 17, 1979. The safety grade design will be submitted for proposal review no later than October 1980. Modifications which will provide safety grade auto initiation will be completed during the 1981 refueling outage, scheduled to commence March 1981.

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2.1.7.b AUXILIARY FEEDWATER FLOW INDICATION

Auxiliary feedwater flow can presently be detected by two independent means; control grade flow transmitters for the auxiliary feedwater lines to the two steam generators and steam generator level using the reactor protective system steam generator levels which meet Military Specification E5272. All cabling is LOCA qualified. The auxiliary feedwater flow transmitters are located in Room 81. No modification is required to meet the 1980 control grade requirements of NUREG-0578. To upgrade the system to safety grade, the following changes will be implemented.

1. Install LOCA qualified transmitters in all the areas discussed above.

2. Install safety routed cable to transmitters FT 1109 and 1110.

3. Insure power supply diversity.

Testability of the flow elements and transmitters will be by means of an inservice test when the auxiliary feedwater system is used for cooldown and heating. All other testing will be by calibration or channel comparison.

Modification for upgrading the control system to safety grade will be completed prior to heatup following our 1981 refueling outage.

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2.1.8.a POST-ACCIDENT SAMPLING

It is the District's intention to complete design reviews, by January 1, 1980, which would assure adequate capability to promptly obtain and analyze primary coolant and containment air samples without over-exposing personnel to radiation. However, the results of other design reviews (such as that performed in Section 2.1.6.b) could have an impact on the results of these design reviews and cannot be factored into the schedule at this time.

The District will revise sampling procedures by January 1, 1980, to ensure that personnel radiation doses are minimized during post-accident sampling.

The District will complete plant modifications identified by the design review prior to heatup following our 1981 refueling outage. However, it is currently expected that design modifications may be very extensive and require the use of equipment not commonly available, with procurement times of up to one year.

A description of plant modifications will be prepared by January 1, 1980.

2.1.8.b HIGH RANGE RADIATION MONITORS

The District will implement, by January 1, 1980, procedures for estimating noble gas and radioiodine release rates if the existing effluent instrumentation goes off-scale. Effluent and in-containment monitoring capabilities described in the NUREG will be achieved prior to heatup following our 1981 refueling outage, provided equipment is available commercially and can be delivered on schedule. Proposed monitoring system designs will be submitted to the Commission for review by October 1, 1981.

2.1.8.c IMPROVED IN-PLANT IODINE INSTRUMENTATION

The Fort Calhoun Station is presently equipped with the necessary equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas in the plant where personnel may be present under accident conditions. The station has two systems which will perform gamma energy spectral analysis.

1. System: TN-11
   Manufacturer: Tracor Northern

2. System: Jupiter
   Manufacturer: Canberra Industries, Inc.

These systems are located in the chemistry laboratory, an area which is presently expected to remain usable under accident conditions. Should the shielding evaluation performed pursuant to Section 2.1.6.b indicate that the chemistry laboratory is not usable under accident conditions, arrangements will be made to make acceptable counting equipment available in another location on an expedited schedule which would depend upon equipment availability. The chemistry personnel are thoroughly trained in sampling techniques and in the operation of the equipment.

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2.1.9 TRANSIENT AND ACCIDENT ANALYSIS

The response to Transient and Accident Analysis requirements is being developed by the C-E Owners Group in conjunction with Generic Resolution meetings with the NRC Bulletins and Orders Task Force. These responses will be submitted on the schedule required by that Task Force by the C-E Owners Group and will be referenced for specific application to the Fort Calhoun Station.

2.2.1.a SHIFT SUPERVISOR'S RESPONSIBILITY

The Shift Supervisor at Fort Calhoun Station is presently designated as the inidvidual in charge of the operation of the power station. The Shift Supervisor's authority is clearly established in standing orders at the plant and designated by letter from corporate management in certain areas such as nuclear material accountability.

The duties of the Shift Supervisor, as described in plant procedures have been reviewed with respect to NUREG-0578. These procedures will be revised as necessary, by January 1, 1980, to ensure that the duties, responsibilities, and authority of the Shift Supervisor, and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the Shift Supervisor, in the control room, relative to other plant management personnel. In addition, corporate management will periodically issue a management directive which emphasizes primary shift supervisor responsibility.

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2.2.2.a CONTROL ROOM ACCESS

Fort Calhoun Station presently has the policy of limiting control room access by use of the onsite security access control system. In addition, the Shift Supervisor may, at his descretion, limit control room access at any time. The District has reviewed the requirements of NUREG-0578, and in response will develop and implement procedures by January 1, 1980, which meet NUREG-0578 requirements.

2.2.1.b SHIFT TECHNICAL ADVISOR

In order to provide the control room with the immediate presence of an individual with technical and analytical capabilities, the District will institute shift technical advisors by January 1, 1980. In anticipation that qualified personnel cannot be hired to continuously fill this function, the Shift Technical Advisor program will require rotation of engineering personnel from other areas within OPPD which provide engineering support to the Fort Calhoun Station. The details of this program are currently under development, but generally incorporates the following features.

1. Personnel: Shift Technical Advisors will initially be technically qualified college graduates who have completed the District's training program. In the long term, personnel with equivalent qualifications will be considered for this position. Because of the diversity of background and experience of these individuals, the operating experience assessment function would also be performed.

2. Training: During the year 1980, a sufficient number of Shift Technical Advisors will be trained to provide quality individuals to all shifts and also allow for rotation of these individuals through their normal engineering functions within OPPD. The initial training program will be completed by January 1, 1981.

3. Schedule: Shift Technical Advisors will be available to the control room whenever the reactor coolant system is heated above 210°F. Shift Technical Advisor responsibilities will be assigned to an individual who may have other assignments, but who will assume the duties and responsibilities of the Shift Technical Advisor in the event of an accident.

It is anticipated that eventually Shift Technical Advisors will reach a level of competence which will permit them to return to their normal job functions for a period of several weeks and then return to the position of Shift Technical Advisor after ensuring that they are sufficiently aware of any safety-related changes in plant status or plant design since their last Shift Technical Advisor assignment. The length of intervals an individual will spend as a Shift Technical Advisor and as an engineer in another OPPD support function is yet to be determined and is dependent upon the requirements of the Shift Technical Advisory Program.

Managerial Comment:

The District believes that benefits will be achieved from the Shift Technical Advisory Program when the individuals who are trained to be qualified advisors return to their normal job function. These benefits will be manifested in more knowledgeable engineering support and a greater understanding of the needs of the Fort Calhoun Station. Continuous repetition of shift work by a single individual would

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2.2.1.b SHIFT TECHNICAL ADVISOR (Continued)

Managerial Comment (Continued)

result in a loss of this individual's input to the offsite engineering support and is therefore not proposed. Further, existing personnel have expressed interest in the program on the condition that it not involve continuous shift work and take them from their respective careers in which they have already started.

2.2.1.c SHIFT TURNOVER PROCEDURES

Fort Calhoun Station presently uses a shift turnover sheet for the control room operators on every shift. This checklist includes safety related system status and a review of alarm status. In addition, critical parameters are recorded and compared with the allowable limits on every shift. The District has reviewed its shift turnover procedures with respect to NUREG-0578. Revisions will be made by January 1, 1980, to bring current procedures into compliance with the NUREG requirements. A checklist for on-coming and off-going control room operators and auxiliary operators and designated technicians will be established as described in the NUREG.

2.2.2.b ONSITE TECHNICAL SUPPORT CENTER

A temporary technical support center will be established and operable at Fort Calhoun Station prior to January 1, 1980. The onsite technical support center will reflect as many as possible of the recommendations presented at the Staff Topical Meetings. The communication between the onsite technical support center and the NRC incident response center will require use of the presently installed system. An attempt will be made to supply parametric information from the control room via a computer link to the onsite technical support center.

The District will submit a description of the permanent onsite technical support center by January 1, 1980, as part of our revised emergency plans. The permanent center, to be established and operable prior to heatup following our 1981 refueling outage, will be in close proximity to the control room, be habitable to the same degree as the control room, and contain communications capability and records as required by NUREG-0578.

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2.2.2.c ONSITE OPERATIONAL CENTER

The present emergency plan at Fort Calhoun provides for the assembly of non-operations personnel at a common location near the emergency control center. They remain available and in communication with the control room and the emergency control center. Any extra operations staff presently report to the control room Shift Supervisor's office and are available to the Shift Supervisor as required.

By January 1, 1980, an onsite operations support center will be established and operable. The center will be separate from the control room, and communications with the control room will be provided. The emergency plan will be revised to reflect the existence of the center and to establish the methods and lines of communication and management.

CONTAINMENT WATER LEVEL INDICATION

The existing containment sump level monitoring system consists of a normal operating level monitor and control system, LC-504 and LIC-505; and a post-accident level monitor LIC-384. The normal level instruments monitor and control containment sump level in the range of 0" to 32". LC-504 and LIC-505 are not LOCA qualified.

LIC-384 is the post-accident monitoring instrument loop. This is a LOCA qualified transmitter with a range 0 to 330 inches which is equivalent to 550,000 gallons of water in the containment basement.

In order to meet the narrow range requirements of Regulatory Guide 1.89 and wide range requirements of Regulatory Guide 1.97, the following modifications and additions are required.

1. Upgrade LIC-384 cable routing to "E" safety grade.

2. Add a level channel redundant to LIC-384, using the same design.

3. Add two narrow range transmitters 0" to 32" (to be redundant and safety grade) of similar design to LIC-384; these are the range from the bottom of the sump to the 976'6" level. The cabling for these is to be a safety grade installation.

These modifications will be completed prior to heatup following our 1981 refueling outage.

CONTAINMENT HYDROGEN MONITOR

The hydrogen detection system, located in the auxiliary building (Room 59) can sample the containment atmosphere at various levels via six connections. Each sample line is provided with a normally closed, remotely operated valve which passes the sample to a common manifold header. The header then passes through the containment via penetration M-57. The sample is measured with a hydrogen analyzer and returned to the containment via another pentration, M-58.

The existing system does not conform with certain design and qualification provisions of Regulatory Guide 1.97.

The District will replace the existing hydrogen detection system with a new dual-channel system which will meet the requirements of Regulatory Guide 1.97, provide continuous indication in control room, and have a range of 0 to 10% concentration. A diagram of the proposed system is attached. This system is of the type used in plants which became operational after the effective date of Reg. Guide 1.97.

It is planned that this system will be installed prior to heatup following our 1981 refueling outage; but this date may be affected by equipment availability.

CONTAINMENT PRESSURE MONITOR

The functional requirements and a conceptual design for containment pressure indication are being developed by the District. Modifications will be completed prior to heatup following our 1981 refueling outage, provided equipment is available. The modifications will provide for continuous indication in control room, a range of -5 psig to at least 180 psig, and meet the requirements of Regulatory Guide 1.97.

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RCS VENTING

The functional requirements and conceptual design of a system for remote venting of the RCS are being developed by the C-E Owners Group. These requirements and a conceptual design will be available for discussions with the NRC staff, to establish Generic Resolutions, no later than January 1, 1980.

The installation of the system for remote venting of the RCS will be completed in accordance with the required schedule pending availability of components.