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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

May 6, 1980

Docket No. 50-346

Mr. Lowell E. Roe
Vice President, Facilities
Development
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

Dear Mr. Roe:

Enclosed for your information is the staff's evaluation for the Davis-Besse Nuclear Power Station of the actions you have taken to satisfy the Category "A" items of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations". This evaluation is based on your submitted documentation and the discussions between our staff at a site visit on February 25 and 26, 1980.

Based on our evaluation, we conclude that with the exception of Items 2.1.3.a, 2.1.3.b and 2.1.7.a, you have satisfactorily met all Category "A" requirements. In our letter of February 1, 1980, we concluded that equipment availability would preclude implementation of Items 2.1.3.a and 2.1.3.b by January 1980. These items should be completed prior to startup following your current refueling outage. Item 2.1.7.a has been the subject of a separate review and will be discussed in a future letter.

With respect to Category "A" items, our Office of Inspection and Enforcement will verify the adequacy of implemented procedures. These items are discussed in our evaluation.

These evaluations do not address the Technical Specifications necessary to ensure the limiting conditions for operation and the long-term operability surveillance requirements for the systems modified during the Category "A" review. You should be considering the proposal of such Technical Specifications. We will be in communication with you on this item in the near future.

Should you have any questions regarding our evaluation, please contact us.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosure: Category A Evaluation

cc w/enclosure: See next page

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Toledo Edison Company

cc w/enclosure(s):

Mr. Donald H. Hauser, Esq.
The Cleveland Electric
Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

Gerald Charnoff, Esq.
Shaw, Pittman, Potts
and Trowbridge
1800 M Street, N.W.
Washington, D.C. 20036

Leslie Henry, Esq.
Fuller, Seney, Henry and Hodge
300 Madison Avenue
Toledo, Ohio 43604

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 420, 7735 Old Georgetown Road
Bethesda, Maryland 20014

Ida Rupp Public Library
310 Madison Street
Port Clinton, Ohio 43452

President, Board of County
Commissioners of Ottawa County
Port Clinton, Ohio 43452

Attorney General
Department of Attorney General
30 East Broad Street
Columbus, Ohio 43215

Harold Kahn, Staff Scientist
Power Siting Commission
361 East Broad Street
Columbus, Ohio 43216

Mr. Rick Jagger
Industrial Commission
State of Ohio
2323 West 5th Avenue
Columbus, Ohio 43216

Mr. Ted Myers
Licensing Engineer
Toledo Edison Company
Edison Plaza
300 Madison Avenue
Toledo, Ohio 43652

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
5503 N. State Route 2
Oak Harbor, Ohio 43449

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

Ohio Department of Health
ATTN: Director of Health
450 East Town Street
Columbus, Ohio 43216

UNITED STATES
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EVALUATION OF LICENSEE'S COMPLIANCE WITH CATEGORY "A" ITEMS OF NRC RECOMMENDATIONS RESULTING FROM TMI-2 LESSONS LEARNED

TOLEDO EDISON COMPANY
DAVIS-BESSE NUCLEAR POWER STATION
UNIT NO. 1

DOCKET NO. 50-346

I. INTRODUCTION

By letter to Toledo Edison Company dated September 13, 1979(1), the NRC transmitted the short term requirements related to the lessons learned from the TMI-2 accident that must be met for the Davis Besse Nuclear Power Station, Unit No. 1. This letter clarified, augmented, corrected and invoked the staff positions presented in the NRC Report NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations." In a later letter to Toledo Edison Company dated October 30, 1979(2), the NRC provided additional guidance and clarification concerning the staff positions and requirements that were transmitted by the September 13, 1979 letter. The short term requirements were divided into two categories, "A" and "B". In accordance with the schedule given in the October 30, 1979, letter, Category "A" requirements were to be implemented by January 1980 and Category "A" requirements were to be implemented by January 1981.

By letters dated October 23(3), November 21(4), December 31(5), 1979 and January 18(6), 22(7), 30(8), 31(9), (10), (11), February 12(12), March 21(13) and April 1(14), 3(15), and 15(16), 1980, Toledo Edison (licensee) submitted commitments and documentation of actions taken at Davis-Besse Nuclear Station, Unit 1 to implement our requirements. To expedite our review of the licensee's actions, members of the staff visited the licensee's facility on February 25 and 26, 1980. This report is an evaluation of the licensee's efforts to implement each Category "A" item which was to have been completed by January 1980.

II. EVALUATION

Each of the Category "A" requirements applicable to PWRs is identified below. The numbered designation of each item is consistent with the identification used in NUREG-0578.

2.1.1 EMERGENCY POWER SUPPLY REQUIREMENTS (Pressurizer Heaters)

Toledo Edison Company has determined based on B&W calculation and startup testing experience that a minimum of 126 kilowatts of pressurizer heaters, which according to Toledo Edison corresponds to a single bank of pressurizer heaters, should be available from an assured power source within two hours after loss of offsite power to establish and maintain natural circulation at hot standby conditions. Startup testing demonstrated that the total heat loss was 82 kilowatts. We have reviewed this information and note this calculated and measured heat loss is similar to heat loss estimates that have been accepted for other pressurized water reactors. We conclude that sufficient heater capacity has been provided to maintain pressure control in the pressurizer during normal hot standby conditions.

The current Davis Besse design provides manual loading of 126 kilowatts of pressurizer heater capacity to each emergency diesel generator. All actions to energize these heaters from the emergency diesel generators can be accomplished from the control room. The licensee has stated that procedures are already in place covering the manual loading of these electrical loads. We find this design is in conformance with the requirements for this item. Therefore, no modifications are necessary to meet the requirements of this item. Verification of the adequacy of the licensee's procedures will be performed by the Office of Inspection and Enforcement and be documented in an appropriate report.

EMERGENCY POWER SUPPLY REQUIREMENTS (Pressurizer Level and Power Operated Relief Valve (PORV) and Block Valves)

The current design has its PORV powered from a 125 volt DC bus supplied by a safety grade battery system. The PORV block valve is an AC motor operated valve and its power supply has been modified to be powered from an Essential Motor Control Center. The power supplies for the PORV's and their associated block valves are therefore independent and diverse.

The current design has two channels of safety grade pressurizer level instrumentation. These channels are also powered from safety buses. Thus the power supplies for PORV, PORV block valve and pressurizer level instrumentation are capable of being powered from both offsite and onsite emergency sources.

We conclude that the licensee meets the emergency power supply requirements for PORV, PORV block valve and pressurizer level instrumentation for this item. Our Office of Inspection and Enforcement will verify the adequacy of the power supply modification to the PORV block valve and be documented in an appropriate inspection report.

2.1.2 PERFORMANCE TESTING FOR RELIEF AND SAFETY VALVES

The licensee has stated in its response to this item that it will participate in the Electric Power Research Institute (EPRI) program to conduct performance testing of PWR relief and safety valves. A description of the test program was provided by EPRI in December 1979. At present this program is under review to ensure that the NUREG-0578 requirements are met.

We will review the test program to confirm the applicability to Davis Besse plant. Completion of the test program is on a schedule different from Category "A" items. We conclude that the licensee has satisfied the Category "A" requirements of this item.

2.1.3.a DIRECT INDICATION OF POWER OPERATED RELIEF VALVE AND SAFETY VALVE POSITION

The licensee will install an acoustical monitoring system to monitor the position of PORV and safety valves. This will be implemented during the planned spring outage. This acoustical monitoring system is provided by Technology for Energy Corporation (TEC) and is similar to those found acceptable by the staff for this purpose for other pressurized water reactors. It is a reliable, single channel system, powered from a vital bus. It will provide the operator with positive indication of valve position and an annunciation of an open valve in the control room. The licensee has stated that the valve position indication components will be seismically and environmentally qualified as appropriate for conditions applicable to their location. This qualification will be completed in October 1980.

Backup valve position indication is provided from the quench tank level indicator, quench tank pressure. These backup methods are included in the plant emergency procedures.

Based on our review of the licensee's design, we conclude that the licensee will meet the requirements for direct indication of PORV and safety valve position for this item prior to start-up from the April 1980 refueling outage. Our Office of Inspection and Enforcement will verify (1) the adequacy of installation of the above design, (2) the adequacy of the qualification and documentation of the valve position indication components and, (3) that the procedures for backup valve position indication are included in the plant emergency procedures. This will be documented in an appropriate inspection report.

2.1.3.b INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

The licensee has provided a description of the existing instrumentation for detection of inadequate core cooling. Further, in its letter of January 8, 1980 the licensee stated that the procedures for use of existing instrumentation to determine the adequacy of inadequate core cooling have been implemented based on guidelines provided by B&W. Our generic review of this item is not yet complete and our evaluation thereof for this item will be reported separately.

ADDITIONAL INSTRUMENTATION

The licensee has stated that it has reviewed several conceptual designs for reactor vessel water level indication. It has informed the staff that it does not consider any of these designs that it has considered to date to be acceptable. The licensee is continuing its effort to provide an appropriate design. We conclude that the licensee has satisfied our short term requirement. However, the need to supplement existing instrumentation and to provide unambiguous indications of inadequate core cooling are still under review. We will complete this item during the review of Category "B" items.

SUBCOOLING METER

The licensee will install two primary coolant saturation meters during its spring refueling outage. These saturation meters will continuously display the margin between actual primary coolant temperature and the saturation temperature (the temperature at which boiling occurs). Each saturation meter will have one pressure input with a range of 0-2500 psig and two temperature inputs with a range of 120°-920°F from each hot leg. The temperature inputs to the saturation meters are not safety grade, however, they will be upgraded to safety grade requirements by January 1, 1981. The power to each saturation meter will be provided from vital sources.

In addition to the saturation meters, backup capability already exists to detect inadequate core cooling conditions. This includes primary coolant temperature and pressure which are directly available to the operator by means of existing console indicators. Steam tables are also provided in the control room for use by the operator to determine saturation margin manually.

Based on our review of the licensee's design, we conclude that the design of the subcooling meters will meet the requirements for this item prior to startup from the April 1980 refueling outage. Verification of the adequacy of the installation of this design and inclusion of the use of subcooling meter in the operating procedures will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

2.1.4 CONTAINMENT ISOLATION

The NRC lessons learned requirements concerning containment isolation direct the licensee to: a) determine which systems penetrating containment are considered essential or non-essential to safety; b) modify containment isolation circuitry to automatically isolate all non-essential system by diverse parameters; and c) modify containment isolation circuitry to assure that clearing of the containment isolation signals does not cause inadvertent opening of containment isolation valves. In addition, the isolation system was reviewed to assure that certain systems which are isolated but might be desireable to use following an accident or transient, can be reopened and to assure that operator controls of containment isolation are not ganged to reopen multiple systems with a single operator action.

The licensee states that the classification of systems penetrating containment as essential and non-essential was reconsidered in response to the NRC bulletins 79-05 and 79-05A. Those safety systems which are required following an accident are considered essential and would remain unisolated.

The licensee states that all non-essential systems are isolated by diverse safety grade signals. Component cooling water, main feedwater and main steam isolation valves are closed if containment pressure exceeds 24 psig. The steam and feedwater line are isolated by a diverse signal from the steam and feedwater line rupture detection system. The component cooling is isolated by a diverse signal of low level in component cooling surge tank. The component cooling system is a closed system not connected to the RCS. This is sufficient justification for not isolating due to the safety features actuation signal.

All the remaining non-essential systems isolate on the diverse parameters of containment vessel pressure greater than 4 psig or RCS pressure less than 1600 psig. A proposal has been submitted to the NRC to change the setpoint for isolation of RCS make up and RCP seal injection and seal return to RCS pressure of less than 400 psig. This would allow seal water to be available to assure operability of RCP during a wider range of plant conditions. Diverse isolation signals would be retained. This change is still under review by the NRC.

The licensee has stated that no containment isolation valves reopen automatically following reset of the containment isolation signal. Reopening of isolation valves is accomplished by manual blocking of the containment isolation signal followed by further manual action to move the valve control switch to the open position. Two manual actions are required to open the valve following isolation. The blocking function affects a group of valves but a single blocking action does not block the isolation signal to all valves.

We conclude that the licensee has satisfied the requirements for this item. Verification that the installed systems at Davis-Besse are characterized by the above design feature will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.1.5.a DEDICATED H_2 CONTROL PENETRATIONS

Davis Besse Unit 1 was licensed to use a hydrogen dilution system for post-accident combustible gas control of the containment atmopshere. Therefore, the plant has a redundant safety grade and dedicated hydrogen dilution system. The containment penetrations to the hydrogen dilution system meet the single-failure criteria for containment isolation and operation of the system. The system is sized to meet the flow requirements during an accident. This has been verified by the staff.

Based on the above considerations, we conclude that the licensee has met the Category "A" and Category "B" requirements of this item. No further action needs to be taken.

2.1.5.b INERTING BWR CONTAINMENTS

This item does not apply to Davis Besse Unit 1.

2.1.5.c H_2 DILUTION PROCEDURES

The licensee has reviewed the procedures and shielding for operating the hydrogen dilution system which provides post-accident combustible gas control of the containment atmopshere. The licensee states that no changes to the shielding or the operating procedures for this system are needed. This system is operated entirely from the control room.

Based on the above considerations, we conclude that the licensee has met the requirements for this item. Verification of the adequacy of the licensee's procedures will be performed by the Office of Inspection and Enforcement. These will be documented in appropriate inspection reports.

2.1.6.a SYSTEM INTEGRITY

The licensee has listed the plant systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident. These systems are the makeup and purification system, low pressure injection system, containment spray system, high pressure injection system, waste gas system, primary sample system and the part of the reactor coolant drain system outside containment up to and including the drain tank. The licensee has implemented an immediate leak reduction program for these systems to reduce their present leakage. The licensee has reported the measured "as found" leakage for these systems.

The licensee has established and implemented a permanent leak reduction program to keep future leakage from these systems to as-low-as-practical levels. This program includes integrated leak rate tests once per refueling cycle; identification of leakage from visual surveillance by plant personnel, area radiation monitors and effluent monitors and corrective actions taken; and the existing plant preventive maintenance program.

The licensee has also reviewed the plant design for potential leakage release paths from the above systems due to design of operator deficiencies as discussed in an NRR letter to the licensee regarding North Anna and Related Incidents dated October 17, 1979. No corrective actions by the licensee were needed.

Based on the above considerations, we conclude that the licensee has met the requirements for this item. There are no Category "B" requirements. Verification of the adequacy of the procedures which implement the licensee's permanent leak reduction and preventive maintenance programs will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.1.6.b PLANT SHIELDING REVIEW

The licensee has performed a radiation shielding design review of the spaces around the plant systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident. These systems are discussed in the evaluation of item 2.1.6.a. This design review has been provided to NRC which includes maps of the estimated dose rates in different areas of the plant. The radioactive source terms assumed for the design review are consistent with the source term given in the NRC clarification letter dated October 30, 1979.

The licensee identified the location of vital areas in which personnel occupancy may be limited. These areas are the reactor coolant sample room and the area near the decay heat pump rooms (i.e., valves there must be manually operated). The licensee will install a shielded reactor coolant sampling facility physically separated from the present facility to reduce the personnel exposure when taking a reactor coolant sample. The licensee will also replace certain manually operated valves for operation of the high pressure injection system with motor operated valves that will be controlled from the control room. These plant modifications are Category B requirements which should be completed by January 1981.

The licensee has identified areas in which safety equipment may be unduly degraded by radiation fields during post-accident operations. The licensee has committed to describe, in the response to IE Bulletin 79-01B, what changes, if any, will be needed to protect this safety equipment. The licensee has identified the type of changes that may be made. Plant modifications are a Catebory B requirement which should be completed by January 1981.

Based on the above considerations, we conclude that the licensee has met the Category "A" requirements for this item. An evaluation of the above design review and the licensee's corrective actions will be performed as part of the review of the Catebory "B" Lessons Learned requirements.

2.1.7 AUXILIARY FEEDWATER FLOW DEDICATION

The licensee currently has installed control grade auxiliary feedwater flow indicators on each steam generator. The sensor is an acoustical device and the indicators are in the control room. The system is testable and accurate to within 10 percent at high flow conditions. The single failure criteria is satisfied by one control grade flow indicator and a safety grade level indicator for each steam generator. We conclude that the system meets the requirements of this item. The licensee has committed to upgrade the system to safety grade flow indication with a system based on the use of orifice plates and pressure sensors. This design will be reviewed as a Category B item.

2.1.8.a POST-ACCIDENT SAMPLING

The licensee has performed a design and operational review of reactor coolant and containment atmopshere sampling systems. For the short term, the licensee has provided interim procedures and a modified sampling system to provide the capability in 1980 to promptly take a reactor coolant sample and a containment air sample during a serious transient or accident and minimize personnel radiation exposure. The containment hydrogen gas analyzer system has been modified for a containment air sample to be taken.

For the long term, the licensee will install a shielded post-accident reactor coolant sample facility physically separate from their present facility. The licensee will add a new sample line to sample reactor coolant from the reactor coolant loop drain. This will allow collection of a representative sample during an accident without operation of letdown into the makeup and purification system. The interim containment air sample facility will be made the permanent post accident facility by installing a containment atmosphere gross monitor with the capability to take a grab sample. Plant modifications are Category B requirements which should be completed by January 1981.

The licensee has performed a design and operational review of the plant radiological analysis facility. This facility can be moved to the water laboratory if it is necessary to reduce the radiation background in this facility. Procedures are available that set forth the methods to promptly quantify radionuclides in a highly radioactive sample during a serious transient or accident.

The licensee has performed a design and operational review of the plant chemical analysis facility. Procedures are available that set forth the methods to promptly quantify certain chemical analyses in a highly radioactive sample during a serious transient or accident.

The licensee has proposed to make changes to his sample analysis facilities to handle post-accident samples. These changes include the radionuclides, boron concentration and dissolved gas analyses. The plant modifications and procedures for these changes are Category "B" requirements which should be completed by January 1981.

Based on the above considerations, we conclude that the licensee has met the requirements for this item. Verification of the adequacy of the procedures for sampling and analyses of samples during post accident operations will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.1.8.b HIGH RANGE EFFLUENT MONITORS

The licensee has provided an interim method for quantifying high level noble gas effluent from the station ventilation line. This line and the main steam lines are the only ones used during a serious transient or accident. The licensee has committed to the development of a methodology to quantify high level radioactive releases from the main steam lines utilizing the existir main steam line radiation monitors. This methodology will be completed before the end of the first refueling outage which is currently in progress.

The licensee has a iodine/particulate sampling system for the station vent. The iodine/particulates are collected on cartridges and taken to the plant radiological counting facility for analysis. Procedures have been developed for collecting and analyzing these samples.

Based on the above considerations, we conclude that the licensee has met the requirements for this item. Verification of the adequacy of the procedures to quantify high-level effluents from the plant will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.1.8.c IN-PLANT IODINE INSTRUMENTATION

The licensee has a gross monitor with silver zeolite filter cartridges which can be dedicated to promptly analyze air samples for radiodine concentrations during an accident. Training of technicians for obtaining and counting air samples has been completed. These gross monitors and cartridges will be taken to the Control Room and to the interim Technical Support Center where plant personnel will be stationed during an accident. The equipment is stored in an area readily accessible to both the Control Room and the interim Technical Support Center.

Based on the above considerations, we conclude that the licensee has met the requirements of this item. Verification that the licensee has the above equipment dedicated to analyzing air samples during an accident, and that it is in place and is periodically checked and calibrated and verification of the adequacy of the procedures and training of plant personnel for operating of the equipment and to determine airborne iodine concentration will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.1.a SHIFT SUPERVISOR RESPONSIBILITIES

The licensee has revised the responsibilities of the Shift Foreman (Shift Supervisor) so that he can provide direct management of ongoing safety related operations and not be distracted from these primary responsibilities by administrative assignments. This revised responsibility has been set forth in plant documents.

Based on the above considerations we conclude that the licensee has satisfied the requirements of this item. Verification of the adequacy of the licensee's procedures will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.1.b SHIFT TECHNICAL ADVISOR

For the interim period of 1980, the licensee has provided an on-shift technical advisor (STA) to assist the shift foreman in the function of accident assessment. The current personnel in this position are degreed engineers with plant experience. They will serve a 24 hour duty day on a rotating basis and will be on-site at all times during their duty. The STA is provided with on-site living quarters. We believe the STA will be able to report to the control room within 10 minutes of notification. He will be called to the control room for any anticipated power change, for any reactor trip, for any loss of safety system function and loss of offsite power supply.

The operational experience function will be handled by the station Technical Section.

For the long term, the licensee will provide training in the areas of reactor operations and transient and plant accident response.

We conclude that the licensee has satisfied the Category A requirements for this item. Verification of the licensee's procedures for implementation of this item will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.1.c SHIFT AND RELIEF TURNOVER PROCEDURES

The licensee revised plant procedures to assure that the procedures are adequate to provide guidance for a complete and systematic turnover between the off-going and on-coming shift to assure that critical plant parameters are within limits and that the availability and alignment of safety systems are made known to the on-coming shift. The procedures also establish a system for evaluating the effectiveness of the shift and relief turnover procedures.

Based on the above, we conclude that the licensee has satisfied the requirements of this item. Verification of the adequacy of the procedures will be performed by the Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.2.a CONTROL ROOM ACCESS

The licensee has implemented procedures which will limit control room access during an emergency and establish the authority and responsibility of the person in charge of the control room to control access and establish the line of succession for the person in charge of the control room. We therefore conclude that the licensee has satisfied the requirements of this item. Verification of the adequacy of the licensee's procedures will be performed by our Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.2.b ON-SITE TECHNICAL SUPPORT CENTER (TSC)

The licensee has established an onsite technical support center on the fifth floor of the Davis Besse office building. Direct communication between the TSC and the control room and NRC have been established. The TSC will provide assistance to the operating personnel in evaluating the course of an incident or accident. The licensee has stated that plant data for the TSC will be provided by phone communication and hard copy computer data. Further, a communicator other than the plant operator or shift supervisor will be stationed in the control room upon activation of the interim TSC. The communicator or other personnel will hand carry the process computer hard copy information to the TSC.

The licensee has proposed to build a permanent center approximately 2100 feet away from the plant. This will be reviewed as a Category B item.

Based on our review of the licensee's submittal and our site visit, we conclude that the licensee has met our requirements for this item. Verification of the adequacy of the licensee's procedures for operation of this center will be performed by our Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

2.2.2.c OPERATIONAL SUPPORT CENTER

The licensee has estab ished the assembly room on the turbine floor as an on-site operational support center. Operations support personnel will be located in the OSC for support of the control room and/or TSC needs. Communication with the control room at the onsite operational support center is provided.

Based on our review of the licensee's submittal and our site visit, we conclude that the licensee has satisfied our requirements for this item. Verification of the licensee's procedures to include this center will be performed by our Office of Inspection and Enforcement and will be documented in an appropriate inspection report.

REACTOR COOLANT SYSTEM HIGH POINT VENTS

The licensee has proposed installation of remotely operated vents for the top of the pressurizer and for the high point of each hot leg. Each vent path will be controlled by two solenoid operated valves in series. The valves will be seismically and environmentally qualified and powered from Class IE power supplies. The valves fail closed on loss of power. The vents are sized such that the flow rate will not exceed the makeup system capacity. It has also committed to install a reactor vessel head vent which will be similar to the proposed hot leg vents. We conclude the licensee has met our requirements for this item.

REFERENCES

1. Letter, NRC (Eisenhut) to ALL OPERATING NUCLEAR POWER PLANTS, dated September 13, 1979.
2. Letter, NRC (Denton) to ALL OPERATING NUCLEAR POWER PLANTS, dated October 30, 1979.
3. Letter, TECO (Roe) to NRC (E/NRR) dated October 23, 1979.
4. Letter, TECO (Roe) to NRC (D/NRR) dated November 21, 1979.
5. Letter, TECO (Crouse) to NRC (D/NRR) dated December 31, 1979.
6. Letter, TECO (Crouse) to NRC (D/NRR) dated January 18, 1980.
7. Letter, TECO (Crouse) to NRC (D/NRR) dated January 22, 1980.
8. Letter, TECO (Crouse) to NRC (R/NRR) dated January 30, 1980.
9. Letter, TECO (Crouse) to NRC (R/NRR) dated January 31, 1980.
10. Letter, TECO (Crouse) to NRC (R/NRR) dated January 31, 1980.
11. Letter, TECO (Crouse) to NRC (R/NRR) dated January 31, 1980.
12. Letter, TECO (Crouse) to NRC (R/NRR) dated February 12, 1980.
13. Letter, TECO (Crouse) to NRC (R/NRR) dated March 21, 1980.
14. Letter, TECO (Crouse) to NRC (R/NRR) dated April 1, 1980.
15. Letter, TECO (Crouse) to NRC (R/NRR) dated April 3, 1980.
16. Letter, TECO (Crouse) to NRC (R/NRR) dated April 15, 1980.

Dated: May 6, 1980