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

May 5, 1980

Docket No. 50-302

Mr. J. A. Hancock
Director, Nuclear Operations
Florida Power Corporation
P. O. Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

Dear Mr. Hancock:

Enclosed is the staff's evaluation of the implementation of Category "A" Lessons Learned requirements (excluding 2.1.7.a) at Crystal River Unit 3. This evaluation is based on your submitted documentation and the discussions between our staffs at a site visit on January 17, 1980.

Based on our evaluation, we conclude that the implementation of the Category "A" requirements at Crystal River is acceptable. Certain items, identified in the evaluation, will be verified by the Office of Inspection and Enforcement.

This evaluation does 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.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
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Enclosure:
Category "A" Evaluation

cc w/enclosure:
See next page

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

FLORDIA POWER CORPORATION
CRYSTAL RIVER UNIT 3 NUCLEAR PLANT

DOCKET NO. 50-302

Date: May 5, 1980

I. INTRODUCTION

By letter to Florida Power Corporation 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 Crystal River Unit 3. 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 Florida Power Corporation 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. Category A requirements were to be implemented by January 1980 and Category B requirements were to be implemented by January 1981.

By letters dated October 17,[3] November 17,[4] 1979, January 11,[5] 1980 February 6, [6], 8 [7], 11 [8], 12 [9], 15 [10]. 1980 and April 11[11], 15[12] and 20 [13], 1980 Florida Power Corporation (licensee) submitted commitments and documentation of actions taken at Crystal River Unit 3 to implement our requirements. To expedite our review of the licensee's actions members of the staff visited the licensee's facility on January 17 and 18, 1980. This report is an evaluation of the licensee's efforts to implement each Category A item.

Implementation of our requirements at Crystal River Unit 3 will be complete prior to start up from the current outage which started February 26, 1980. The majority of the Category A requirements were completed by the licensee prior to February 15, 1980 which was the date the NRC required completion for this plant, except for delayed equipment delivery related items.

II. EVALUATION

Each of the Category A requirements applicable to PWR's is identified below. The numbered designation of each item is consistant with the identification used in NUREG-0578.

2.1.1 EMERGENCY POWER SUPPLY REQUIREMENTS PRESSURIZER HEATERS

Florida Power Corporation has determined based on B&W calculation and startup testing experience, that a conservative value for the total heat loss from the primary coolant system, under hot standby conditions, is 107 kilowatts. On this basis, it has determined that a minimum of 126 kilowatts of pressurizer heaters, which according to Florida Power Corporation corresponds to a typical 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. We have reviewed this information and note this calculated 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. This is in accordance with our position.

Each redundant group of heaters are supplied from non-safety related motor control centers during all modes of operation. The design provides the capability to realign one of the heater groups to one emergency power train and another heater group to the redundant and independent emergency power train. A procedure has been written and is available to the operator which will allow the connection of the preselected heaters to the respective emergency power supply during a loss of offsite power. This will be accomplished by utilizing the existing cross-tie breakers and assuring that all non-essential loads are disconnected from appropriate buses. This transfer of the heaters from the normal power source to the emergency power source can not be accomplished completely within the control room and some of the disconnections of the non-essential loads may have to be accomplished at the local power center. We find this method to be acceptable. The bases for acceptability of this operational mode is that there is sufficient time, up to two hours, for operator action outside the control room.

We conclude that the licensee has met the emergency power supply requirements of this item. Verification of the adequacy of licensee's procedures will be performed by the Office of Inspection and Enforcement and documented in an appropriate inspection report.

EMERGENCY POWER SUPPLY REQUIREMENTS (PRESSURIZER LEVEL, POWER OPERATED RELIEF VALVE (PORV) AND BLOCK VALVE)

The pressurizer level indication instrument channels are powered from the vital instrument buses. The motive and control components for the PORV are powered from the on-site DC power systems. The motive and control power for the block valve are powered from the AC emergency power supply. The power supplies for the PORV and the block valve are therefore independent and diverse. The power supplies for PORV, pressurizer level indicators and block valve are capable of being powered from both offsite power and the onsite emergency power system. Any changeover of the power supplies from the offsite power to the emergency power source will be automatic. We have been informed that, due to the February 26, 1980 event at the plant, the pressurizer level indication will be provided with a different power supply which will be capable of being supplied by emergency power.

We have reviewed the above design and conclude it is in accordance with the requirements of this item.

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 its applicability to the Crystal River Plant. Completion of the test program is on a schedule different from Category A items. We conclude that the licensee's commitment to participate in this EPRI test program 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 has purchased an acoustical monitoring system from B&W to monitor the position of PORV and safety valves. This system will be installed during the present outage. This acoustical monitoring system is similar to those submitted on other pressurized water reactors. Valve position will be monitored by a reliable, single channel system powered from a battery backed vital bus. This system will provide the operator with positive indication of valve position and an annunciation of an open valve in the control room. The valve position indication components will be seismically and environmentally qualified as appropriate for conditions applicable to their location. The licensee is in the process of determining the seismic and environmental conditions to which this equipment must be qualified as well as the test methods to be used. This effort will be completed in October 1980.

Back up valve position indication is provided in terms of quench tank level and pressure instrumentation which are indicated and alarmed in the control room. The licensee has stated that there has been extensive operator training at Crystal River to ensure operator awareness of the indications of stuck open PORV or safety valves.

Based on our review of the licensee's submittal, we conclude that prior to startup from the current outage the licensee will meet the requirements of this item. The Office of Inspection and Enforcement will verify (1) the adequacy of the installation of the above design and, (2) 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 FOR PWR'S AND BWR'S

The licensee in response to IE Bulletin 79-05C submitted guidelines that will allow the reactor operator to recognize and respond to inadequate core cooling conditions. In addition, it has revised its plant procedures to incorporate these new guidelines and has implemented operator training related to the inadequate core cooling. Our generic review of this area is not yet complete and our evaluation of this item will be reported separately.

ADDITIONAL INSTRUMENTATION

The licensee has submitted a conceptual design to the staff and has reviewed several conceptual designs for reactor vessel water level indication. FPC has informed the staff that it does not consider any of these designs that it has reviewed 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 has installed two primary coolant saturation meters on temporary mountings. These meters will be installed in the permanent cabinets prior to startup from the current outage. These saturation meters will continuously display saturation margin conditions. Degrees of superheat are displayed as negative margin should temperature exceed Tsat. The two saturation meters are located in the 4160 volt switchgear room B. Originally, the licensee included four temperature inputs for each saturation meter. Two wide range hot leg temperatures and two cold leg temperatures (from each loop) with a range of 120°F to 920°F and two reactor coolant pressure inputs (one per loop) with a range of 0 to 2500 psig. These signals are taken from the Non-Nuclear Instrumentation (NNI) system.

Since their February 26, 1980 transient, the licensee has modified the design of the saturation meters. This includes use of auctioneered in-core thermocouples as primary temperature inputs to each saturation meter. Five thermocouples (one from each quadarant and one from the center of the core) provide inputs to each saturation meter, i.e. a total of ten. The operator also has the capability to use the back up temperature inputs from hot leg and cold leg as described above, by a selector switch on the control panel. The power to the saturation meters is provided from vital sources.

Each meter has a remote digital indicator/selector, mounted on the main control board, and a low margin to saturation alarm is also provided in the control room.

In addition to the saturation meters, backup capability already exist to detect inadequate core cooling conditions. This includes the plant computer using auctioneered reactor system temperature and selected in-core thermocouples and reactor system pressure. This system provides a computer alarm on decreasing saturation margin. A plant fabricated pressure vs temperature plot, utilizing a scope with overlays, representing the saturation curve and allowable margin to saturation, also provides a visual indication of the saturation margin.

Based on our review of the licensee's design, we conclude that the design of the subcooling meters will be in compliance with our requirements for two reliable systems for indicating subcooling of the reactor system. This modification will be completed prior to startup from the current outage. Verification of the adequacy of installation of the above design and procedures to use the subcooling meters will be performed by the Office of Inspection and Enforcement and be 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 whether systems penetrating containment are considered essential or non-essential to safety; b) modify containment isolation circuitry to automatically isolate all non-essential systems by diverse parameters; and c) modify containment isolation circuitry to assure that resetting of the containment isolation signals does not cause the inadvertent opening of containment isolation valves. In addition, the isolation system was reviewed to assure that certain systems which are isolated but might be desirable for 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 identified essential and non-essential systems with regard to containment isolation and core cooling in the April 12, 1979 response to Item 6 of IE Bulletin 79-05A. Those systems required for core cooling are defined as essential. This includes ECCS lines, RCP seal and cooling lines, containment sample and pressure sensing lines, and closed loop cooling to the letdown coolers, reactor cavity and the CRDs. All of these, with the exception of the ECCS lines and the RCP seal water supply, isolate on high reactor building pressure. Thus, many of the systems, which the licensee designates as essential, do isolate, but only on high reactor building pressure.

The licensee states that all non-essential valves which receive an isolation signal on high reactor building pressure will be provided with a diverse isolation signal based on HPI actuation prior to startup from the current outage. Also, as described in the February 11, 1980 FPC letter, the letdown system has been reclassified as a non-essential system and is provided with diverse isolation signals. The licensee states that the diverse signals satisfy safety grade requirements. Resetting of containment isolation signals will not result in automatic loss of containment isolation. Also, two independent operator actions are required to re-open any isolated system.

We conclude that prior to startup from the current outage the licensee will have satisfied the requirements of this item. Verification of the adequacy of the installation of the above design will be performed by the Office of Inspection and Enforcement and be documented in an appropriate inspection report.

2.1.5.a DEDICATED H_2 CONTROL PENETRATIONS

Crystal River 3 was licensed to use a hydrogen purge system for post-accident combustible gas control of the containment atmosphere. Therefore, Crystal River 3 was required to have a redundant, safety-grade and dedicated hydrogen purge system. Each dedicated containment penetration for the exhaust and the air supply of the hydrogen purge systems have redundant isolation valves in series outside containment. Therefore, these containment penetrations meet the single-failure criteria for containment isolation. There are redundant containment penetrations for hydrogen purge exhaust and two separate air supply systems each with a containment penetration for hydrogen purge air supply. Therefore, these containment penetrations meet the single-failure criteria for operation of the hydrogen purge system. The lines have been sized for the flow requirements of the purge system. This has been verified by the staff.

Based on the above considerations, we conclude that the licensee has met the requirements for this item.

2.1.5.b INERTING BWR CONTAINMENTS

This item does not apply to CR-3 which is a PWR.

2.1.5.c H_2 PURGE PROCEDURES

The licensee has reviewed the procedures and shielding for operating the hydrogen purge system which provides post-accident combustible gas control of the containment atmosphere during an accident. The licensee states that no changes to shielding for this system are needed.

After an accident, a minimum of three portable air compressors will be delivered to the plant site and located on the berm adjacent to the Intermediate Building. After connecting these compressors to the integrated leak rate test connection, personnel must enter the Intermediate Building to align valves to operate the purge system. These operations can be carried out within the exposure limits of General Design Criteria 19, Appendix A to 10 CFR Part 50. In addition, the station air system has the capacity to be used to supply air for the hydrogen purge air supply through a separate penetration.

The licensee has identified two potential changes to the hydrogen purge system. These include the addition of a small exhaust air flow filter and water drain lines. This modification will be completed as a Category B item.

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

2.1.6.a <u>SYSTEM INTEGRITY</u>

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 following: Reactor Coolant Bleed line, Waste Gas Disposal System, Decay Heat Removal System (low pressure safety injection), Reactor Building Spray System, Makeup and Purification System, High Pressure Safety Injection System and Reactor Coolant Sampling System. Prior to issuance of NUREG-0578 the licensee had implemented a leak reduction program for the above systems which had leakage requirements identified in the plant Technical Specifications. By 1980, this program was expanded to include the requirements of this item for all the systems listed above. The licensee has reported the measured "as found" leakage for these systems to NRC except for the Decay Heat Removal System, Reactor Building spray system and Reactor Coolant Sampling System. All these systems will be leak tested before startup from the present outage.

The licensee's leak reduction program should keep future leakage from these systems to low-as-practical levels. The program includes checks for leakage from periodic integrated leak rate tests; daily identification of leakage from reactor coolant system leakage test, the total water inventory program and visual surveillance by plant personnel; area radiation monitors and the unit ventilation effluent monitors; and the existing plant preventative maintenance program.

The licensee has reviewed the plant design for potential leakage release paths from the above systems due to design and operator deficiencies as discussed in an NRR letter to the licensee regarding North Anna and Related Incidents dated October 17, 1979. The licensee has concluded that no changes to the plant are needed. The licensee is reviewing two unplanned gas releases from the plant which are similar to the North Anna Incident to determine if any plant changes are needed. Any modifications identified from this review will be completed promptly.

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

2.1.6.b PLANT SHIELDING REVIEW

The licensee has performed a radiation and 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 listed in the evaluation of item 2.1.6.a. This design review has been provided to NRC. The radioactive source terms assumed for the design review are consistent with the source terms given in the NRC clarification letter dated October 30, 1979. The licensee has identified a modification based on this design review, that of changing manual operated valves to motor operated valves to reduce personnel exposure during an accident.

The licensee did identify the location of vital areas and equipment and instrument areas in which personnel occupancy may be limited. The areas with limited access are those used for reactor coolant and containment air sampling and analysis and the area around the radioactive waste disposal control board. The licensee plans to relocate this control board to a lower radiation area. The areas for sampling and analysis are discussed in Item 2.1.8.a.

The licensee is continuing its review of plant ESF equipment outside containment to determine if any of this equipment will be unduly degraded by radiation fields during post-accident operations. The licensee has identified the approximate location of ESF equipment, the radiation sensitive material in each piece of equipment and the post-accident integrated radiation exposure calculated for the six month period following a serious accident for each location. The licensee must still determine the period of time that the plant ESF equipment will have to function during post-accident operations.

The licensee will complete its review to determine the needed modifications to the plant to provide access to vital areas and protect vital mechanical equipment prior to June 1, 1980. The licensee will complete this review for electrical equipment as part of its response to IE Bulletin 79-01B. Modifications to the plant and/or ESF equipment is a Category B item and is required to be completed by January 1, 1981.

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

2.1.7.b AUXILIARY FEEDWATER FLOW INDICATION

The licensee has installed control grade emergency feedwater flow indication for each steam generator and it meets our vital power requirements. It is based on an acoustic system which 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 one control grade level indicator for each steam generator. Flow and level instrumentation are available in the control room. We conclude that the system meets the short term requirements of this item. The licensee has committed to upgrade the system to safety grade by January 1981.

2.1.8.a POST-ACCIDENT SAMPLING

The licensee has performed a design and operational review of the reactor coolant and containment atmosphere sampling. The licensee has procedures and an interim modified sampling system to provide the capability in 1980 to promptly take a reactor coolant sample during a serious transient or accident and to minimize personnel radiation exposure. To obtain reactor coolant samples during an accident, temporary sample lines have been extended from the sample room through a common wall into the hood in the radiochemistry laboratory. A single entry into the nuclear sample room will be required to align valves for post-accident sampling. Subsequent sampling will be controlled from the radiochemistry laboratory. Portable shadow shielding will be provided to shield the person from the makeup and purification system piping. The sample will be collected in a shielded container.

The licensee is evaluating several alternatives to improve their capability to take post-accident reactor coolant samples: a permanent post-accident reactor coolant sampling facility in the radiochemistry laboratory, redesign of the nuclear sample room and the installation of a new post-accident reactor coolant sample station. This evaluation includes use of on-line monitors to analyze samples to further reduce radiation exposure.

The licensee has provided procedures to take a containment air sample in an area which will minimize personnel exposure. Post-accident containment atmosphere sampling can be obtained utilizing the plant air and integrated leak rate test penetrations through containment. These penetrations are located in the Intermediate Building. A pump for taking samples is permanently installed. Portable lead shielding is available to further reduce exposure if it is needed.

New coolant system sampling points have been proposed and are being reviewed for detailed design to allow collection of a representative sample during a serious transient or accident without having to operate the reactor coolant letdown.

The licensee is doing a design and operational review of the plant radiological analysis facility and the plant chemical analysis facility. Permanent modifications to this facility including modifications to seal openings and/or penetrations between this facility and the Auxiliary Building to preclude airborne contamination and to provide additional shielding are under review. If this facility cannot be used during an accident, the facility can be relocated in an onsite warehouse and comparable equipment in the health physics laboratory can be used to analyze samples.

For the short term requirements, procedures are available to provide the capability to promptly quantify radionuclides and certain chemical analyses in a highly radioactive sample during a serious transient or accident.

Based on the above considerations, we conclude that the licensee has met our Category A requirements. Permanent modifications of the plant radiological analysis facility, chemical analysis facility and coolant sampling facility are Category B requirements and will be completed by 1/1/81. Verification of the procedures for taking and analyzing a reactor coolant and containment samples will be performed by the Office of Inspection and Enforcement and be documented in an appropriate inspection report.

2.1.8.b INCREASED RANGE OF RADIATION MONITORS

The licensee has provided an interim method for quantifying high level noble gas effluent from the Auxiliary Building ventilation line and the Reactor Building vent. These lines and the main steam system are the only ones used during a serious transient or accident involving the reactor. The hydrogen purge is used to control the hydrogen concentration inside containment after an accident and utilizes the reactor building vent. The licensee is studying means to monitor noble gas radioactivity releases from the main steam system. The licensee will install a radiation monitoring system on this line and provide interim procedures to quantify releases from this line prior to June 1980.

The licensee has installed a new noble gas monitoring and iodine/particulate sampling system for the Auxiliary Building vent and Reactor Building vent lines to provide monitoring and sampling at a more accessible location. 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 our Category A requirements. Verification of the adequacy of the procedures for quantifying high-level radioactive noble gas and iodine/particulate effluents from the plant will be performed by the Office of Inspection and Enforcement and be documented in an appropriate inspection report.

2.1.8.c IMPROVED IODINE INSTRUMENTATION

The licensee has GM monitors and silver zeolite cartridges which can analyze air samples for radioiodine concentrations during an accident. This equipment will be located onsite: one set in the control room area and one set in the Technical Support Center where plant personnel will be stationed during an accident. The licensee has also ordered a single channel analyzer which can be used to promptly and accurately analyze air samples for airborne radioiodine during an accident: it should be available prior to June 1980.

Based on the above considerations, we conclude that the licensee has met our Category A requirements. There are no Category B requirements. Verification of the procedures that state the licensee has equipment dedicated to analyzing air samples during an accident, that the above equipment is in place and is periodically checked and calibrated and verification of the adequacy of the procedures and training of plant personnel for operation of the equipment 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 Supervisor so that he can provide direct management of ongoing safety related operations and not be distracted by administrative details. This revised responsibility has been set forth in plant documents.

We conclude that the licensee has satisfied the requirements for this item. Verification of the licensee's procedures will be performed by the Office of Inspection and Enforcement and be documented by 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 the shift supervisor to fulfill the function of accident assessment. Selected plant personnel who currently hold an SRO will be used for this function. They will have received a minimum of 2 weeks of additional training at the B&W technical advisor classroom and simulator training course. They will serve a 24-hour duty day on a rotating basis and will be on site at all times during their duty. We believe they will be available to report to the control room within 10 minutes of being called by the shift supervisor.

The operating experience assessment function will be performed by contract consultants.

For the long term, the licensee will establish a separate organization to fulfill the functions of the STA. The position of Operations Engineer will be established to supervise a complement of Operations Technical Advisors who will combine the functions of accident assessment and operating experience assessment. The Operations Engineer will have a bachelor's degree and five years nuclear experience, or sixty hours of college level technical training and 10 years nuclear experience. The operations technical advisors will hold an RO license or be given equivalent training.

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 be documented in an appropriate inspection report.

2.2.1.c SHIFT AND RELIEF TURNOVER PROCEDURES

The licensee has revised plant procedures to assure that 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.

We conclude that the licensee has met our requirements for this item. Verification of the licensee's procedures for implementation of this item and for establishing a system for evaluating the effectiveness of the procedures will be performed by the Office of Inspection and Enforcement and 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. The shift supervisor is responsible for maintaining control of personnel entering the control room. He is authorized to refuse entry or direct personnel to leave the control room. During an accident the shift supervisor or assistant nuclear shift supervisor will limit the control room access to only those personnel who are essential for the direct operation of the plant and to technical advisors required to support the particular operating condition.

On the basis of our review, we conclude that the licensee has satisfied our requirements for this item. The Office of Inspection and Enforcement will verify the adequacy of the implemented procedures. This will be documented in an appropriate inspection report.

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

The licensee has established a temporary onsite technical support center in the office building located on the northwest corner of the turbine building. The TSC will provide assistance to the operating personnel in evaluating the course of an incident or accident. Direct communication between TSC and the control room, and the NRC have been established. Portable monitoring equipment for measuring radiation levels in the TSC is provided. Plant parameters necessary for assessment have been provided by a computer printer located in the TSC and in parallel with the control room computer printer. The licensee has proposed to build a separate center located near the administrative building within close proximity to the control room. This will be reviewed as a Category B item.

Based on our review of licensee's submittal and our site visit, we have concluded that TSC at the Crystal River 3 satisfies our Category A requirements for this item. The Office of Inspection and Enforcement will verify that the procedures are in effect directing the operation of the TSC. This will be documented in an appropriate inspection report.

2.2.2.c OPERATIONAL SUPPORT CENTER

The licensee has established a on-site operational support center at the north end of the shop facilities building located northeast of the control complex. Operations Support Personnel will be located in the OSC for response to control room and/or TSC needs.

Based on our review of licensee's submittal and our site visit, we conclude that the licensee has satisfied our requirements for this item. The Office of Inspection and Enforcement will verify that the licensee has revised his procedures to include this center and its use is included in the licensee's emergency plan. This will be documented in an appropriate inspection report.

RCS HIGH POINT VENTING

The licensee has proposed installation of remotely operated vents for the high point of each hot leg, for the top of the pressurizer and for the reactor vessel head. Each vent path has two valves in series so a single failure will not result in reactor coolant leakage. All valves fail closed on loss of power. Both series vent valves at a single venting location are energized by a power supply different from that which powers the valves at the other vent paths to assure that failure of a power supply will not disable the entire venting capability. We conclude that the licensee has met our Category A requirements for this item.

REFERENCES

1. Ltr. NRC (Eisenhut) to ALL OPERATING NUCLEAR POWER PLANTS, dtd. September 13, 1979.
2. Ltr. (Denton) to ALL OPERATING NUCLEAR POWER PLANTS, dtd. October 30, 1979.
3. Ltr. FPC (Stewart) to NRC (E/NRR) October 17, 1979.
4. Ltr. FPC (Baynard) to NRC (D/NRR) November 17, 1979.
5. Ltr. FPC (Baynard) to NRC (D/NRR) January 11, 1980.
6. Ltr. FPC (Baynard) to NRC (D/NRR) February 6, 1980.
7. Ltr. FPC (Baynard) to NRC (D/NRR) February 8, 1980.
8. Ltr. FPC (Baynard) to NRC (D/NRR) February 11, 1980.
9. Ltr. FPC (Hancock) to NRC (D/NRR) February 12, 1980.
10. Ltr. FPC (Baynard) to NRC (D/NRR) February 15, 1980.
11. Ltr. FPC (Moore) to NRC (D/NRR) April 11, 1980.
12. Ltr. FPC (Bright) to NRC (D/NRR) April 15, 1980.
13. Ltr. FPC (Bright) to NRC (D/NRR) April 20, 1980.