Florida
Power
CORPORATION

January 11, 1980

File: 3-0-3-a-3

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

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

Dear Mr. Denton:

This letter supplements Florida Power Corporation's letters of October 17 and November 17, 1979, concerning implementation of the Short-term Lessons Learned Recommendations contained in NUREG-0578 by February 15, 1980, unless otherwise stated.

Information on system reliability for the Florida Subregion will be provided by January 15, 1980, as requested in your January 4, 1980, letter.

Florida Power Corporation's Supplemental Response, Enclosure 1, details the implementation of the Short-Term Lessons Learned Recommendations. The various attachments provide supporting documentation as follows:

Attachment I -- Information Required on the Subcooling Meter. This provides detailed technical and analytical capabilities of the subcooling meter.

Attachment II -- Babcock & Wilcox Saturation Meter - Implementation of NRC Requirements. The explanation of how the saturation meter meets the intent of the NRC recommendations for single failure criteria, testability, seismic criteria, and environmental qualification, is identified.

Advanced Cys To Telford ORB #4

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Florida Power
CORPORATION

January 11, 1980

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Attachment III -- Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post-accident Operations Outside Containment at Crystal River Unit 3 Nuclear Generating Station. The review of the post-accident radiation fields, based on guidelines provided in Darrell G. Eisenhut's September 13, 1979, letter, and your October 30, 1979, letter is summarized.

Attachment IV -- Venting Design Criteria. This is a summary of the venting design criteria utilized in venting noncondensible gases to aid in refilling the RCS and promoting natural circulation flow for core cooling.

Attachment V -- AI-500, Conduct of Operations. This in-plant implementing procedure identifies the method of operation for the Operations Staff.

Attachment VI -- Management Responsibility of Nuclear Shift Supervisor. This directive emphasizes the primary management responsibility of the Shift Supervisor and clearly establishes his command duties.

Attachment VII -- AI-200, Organization and Responsibility. This in-plant implementing procedure identifies the Crystal River Unit 3 organization and responsibilities.

Attachment VIII -- Long-range Plan for Upgrading the On-site Technical Support Center. Preliminary facility plans are provided.

Based on system reliability considerations, and the necessary equipment for implementation of Recommendation 2.1 3.a - Response to Direct Indication of Power-operated Relief Valves and Safety Valve Position for PWRs and BWRs, Florida Power Corporation will implement the PORV and Safety Valve Position Indication modification within 30 days of receipt of all equipment. The mounting cabinets are presently scheduled to be shipped from B&W to FPC on February 22, 1980.

As a result, the required compliance and maintenance outages of other FPC generating units will be scheduled in accordance with an anticipated implementation deadline for this modification of late March, 1980.

Should any unforseen circumstances arise that preclude Florida Power Corporation's implementation of the Short-term Lessons Learned Recommendations are identified in this submittal, we will inform you within 48 hours of the circumstances involved and any changes in commitments.

Very truly yours,

FLORIDA POWER CORPORATION

Patsy Y. Baynard

Dr. P. Y. Baynard
Manager
Nuclear Support Services

NUREG-0578(TxmtlLtrl)DN-94

Attachments

STATE OF FLORIDA

COUNTY OF PINELLAS

Dr. P. Y. Baynard states that she is the Manager, Nuclear Support Services, of Florida Power Corporation; that she is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of her knowledge, information, and belief.

Patsy Y. Baynard
Dr. P. Y. Baynard

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 11th day of January, 1980.

Catherine S. Cameron
Notary Public

Notary Public, State of Florida at Large,
My Commission Expires: August 8, 1983

CameronNotary 3(D12)

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FLORIDA POWER CORPORATION

RESPONSE TO NUREG-0578
SHORT-TERM RECOMMENDATIONS
FOR
CRYSTAL RIVER UNIT 3

Supplemental Response
January 11, 1980

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FLORIDA POWER CORPORATION'S RESPONSE TO NUREG-0578

Recommendation 2.1.1 Emergency Power Supply

Pressurizer Heater Emergency Power Supply

After further review of the NUREG-0578 recommendations and subsequent clarification for Section 2.1.1, Florida Power Corporation has determined that the existing design satisfies the NUREG recommendations with only one exception. This being that the transfer of the heaters from the normal power source to the emergency power source can not be accomplished completely within the control room. This exception is justified with the fact that after the accident, B&W has determined that the operator has 2 hours to accomplish the connection of the preselected pressurizer heaters of sufficient capacity (126 kW) to initiate and maintain natural circulation.

A procedure (EP-101, Unit Blackout) is available to the operators which will allow the connection of the preselected heaters to the Engineered Safeguards (Safety-related) Bus during a loss of offsite power. This will be accomplished by utilizing the existing cross-tie breakers and assuring that all nonessential loads are disconnected from the respective buses. This method meets the intent of the NUREG requirements with the exception that the manual transfer of heater breakers is not entirely accomplished in the Control Room. Some of the disconnections of the nonessential loads may have to be accomplished at the local power center. Load consideration is given in this procedure, to prevent overloading a diesel generator.

Pressurizer Level and Pressurizer Relief and Block Valves Emergency Power Supplies

The existing design satisfies the requirements of NUREG-0578 for the power supplies for the Pressurizer Level Indicators and Pressurizer Relief and Block Valves, as follows:

1. The motive and control components for the Relief Valve are powered from the on-site DC power system.

2. The motive and control components for the Block Valve are powered from the AC emergency power supply (Engineered Safeguards Bus).

3. The pressurizer level indication instrument channels are powered from the vital instrument buses (Inverters).

4. As noted in 1 and 2 above, the power for the Block Valve is supplied from a different bus from that which supplies the Relief Valve.

5. The motive and control power connections to the emergency buses are through safety-grade devices.

6. The manual transfer of power from the normal power to the emergency power is not applicable to the design. As noted in 1 above, the Relief Valve is normally powered from the on-site DC power system, therefore, no transfer is required. As noted in 2 above, the Block Valve is normally powered from the Engineered Safeguards Bus, which is normally powered from an offsite source. On a Loss of Offsite Power Event, the Emergency Diesel Generators will automatically pick up the Engineered Safeguards Bus and the safety-related loads connected to it. This includes the ES 3AB MCC, which feeds the Block Valve, therefore, a manual transfer is not applicable.

Recommendation 2.1.2 PWR Relief and Safety Valve Testing

By letter dated December 17, 1979, Mr. William J. Cahill, Jr., Chairman of the EPRI Safety and Analysis Task Force submitted "Program Safety/Relief Valves and Systems," December 13, 1979.

Florida Power Corporation considers the program to be responsive to the requirements presented in Section 2.1.2 of NUREG-0578. The EPRI Program Plan provides for a completion of the essential portions of the test program by July 1981. Florida Power Corporation will be participating in the EPRI program to provide technical review and to supply plant specific data as required.

Recommendation 2.1.3.a Response to Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWRs and BWRs

In direct response to NUREG-0578, Item 2.1.3.a, Florida Power Corporation has purchased from Babcock & Wilcox a Valve Monitoring System. This system incorporates acoustical monitoring techniques to provide the reactor operator with indication of valve open/closed position. The equipment is very similar to the existing Loose Parts Monitoring System supplied by Babcock & Wilcox. The engineering design for installation of this equipment is proceeding on an expedited basis to meet the specified inservice date.

This design provides for two transducers mounted on each safety valve and the PORV. Each of these transducers will be wired out of the containment to the PORV/T_{sat} monitoring cabinet, to be located in the 4160 V ESB SWGR Room. Within this cabinet will be three channels (one for each valve) of signal conditioning with local indication, alarm (high and low), and selectable audio monitor. Only one transducer will be normally monitored on each valve. The other is manually selectable for comparison of performance or in the event of transducer failure. Each channel will also provide remote analog indication and annunciator events recorder high alarm functions. This analog indicator for each channel will be mounted on the PSA section of the main control board. A common annunciator window will be located on the ICS section of the main control board. The events recorder will provide CRT and hard copy indication of valves that actuate.

The valve monitoring T_{sat} cabinet will be powered from a vital source with all cable routing meeting seismic requirements. Seismic testing

of equipment identical to that used in the Babcock & Wilcox Valve Monitoring System has been performed. Environmental qualification of in-containment equipment has been satisfied by similar equipment test and survival of TMI-2 equipment.

The present equipment delivery schedule for the above-described modification will not permit complete installation by February 15, 1980. The accelerometers, which were manufactured by N. Devco, were received on-site on January [illegible] 1980. The visual indicators (edge meters) for the main control board are being manufactured by International Instruments, and are presently scheduled to be shipped from Babcock & Wilcox to Crystal River Unit 3 on January 25, 1980. The mounting cabinets are being manufactured by Hoffman Engineering Company, and are presently scheduled to be shipped from Babcock & Wilcox to Crystal River Unit 3 on February 22, 1980.

In accordance with Section III of the Commission's Show Cause Order for Crystal River Unit 3, dated January 2, 1980, the above modification to provide direct position indication in the Control Room for the PORV and safety valves at CR-3, will be completely installed within 30 days after receipt of all necessary equipment at CR-3. This installation will be completed no later than June 1, 1980. The installation of this equipment will require a unit shutdown of approximately 10 days.

Recommendation 2.1.3.b Instrumentation for Inadequate Core Cooling

In response to NUREG-0578, the Babcock & Wilcox Owners' Group has developed an extensive program for inadequate core cooling which has been discussed with the Bulletins and Orders Task Force. In addition, at the request of the Bulletins and Orders Task Force, the program has been expanded beyond the requirements of NUREG-0578. The objectives of this program are as follows:

1. Develop operating guidelines that will allow the reactor operator to recognize and respond to conditions of inadequate core cooling under the following conditions:

    a. Power Operation with portions of the core in DNB.

    b. Loss of RCS Inventory without the reactor coolant pumps operating.

    c. Loss of RCS Inventory with the reactor coolant pumps operating.

    d. Loss of the Decay Heat Removal System and Loss of RCS Inventory During Refueling Operations.

    e. Loss of natural circulation due to loss of heat sink.

2. Provide recommendations for any additional instrumentation required to indicate inadequate core cooling under the conditions listed above. Included with the recommendations will be:

   a. A description of the functional design requirements for the additional instrumentation.

   b. A description of the Operating Guidelines to be used with the proposed equipment.

   c. A description of the analyses used in developing these guidelines.

   d. Installation schedules for additional instrumentation.

To-date, Operating Guidelines and supportive analyses are complete for the following conditions within the scope of the Inadequate Core Cooling Program:

1. Loss of RCS Inventory without the reactor coolant mps operating.

2. Loss of RCS Inventory with the reactor coolant pumps operating.

3. Loss of natural circulation due to a loss of heat sink.

These guidelines and supportive analyses have been submitted to the NRC by Florida Power Corporation in response to IE Bulletin 79-05C, dated November 14, 1979. Florida Power Corporation has revised Plant Procedures to incorporate these new guidelines and has implemented operator training related to the inadequate core cooling. This activity is complete.

Additional guidelines/support analyses for refueling operations have been performed by Babcock & Wilcox and are presently being reviewed by Florida Power Corporation. Copies of the B&W guidelines and supportive analyses will be submitted to the Bulletins and Orders Task Force. Following our review, we will develop the necessary procedures and implement operator training. This effort will be completed by April 1, 1980.

On January 4, 1980, Florida Power Corporation received from B&W their analyses and guideline recommendations for inadequate core cooling due to DNB at power. As a result of these evaluations, no substantive changes to existing operating procedures are necessary. The investigations indicated that to obtain inadequate core cooling at power, the operators would need to ignore numerous existing alarms or major non-mechanistic damage to reactor internals needs to occur. Copies of these analyses and guideline recommendations will be submitted to the Bulletins and Orders Task Force.

Babcock & Wilcox is scheduled to submit to Florida Power Corporation a final report containing recommendations for additional instrumentation in late January, 1980. FPC will submit this information as soon as

possible to you, following our review. Every effort will be made to install this new instrumentation by January 1, 1981, subject to equipment availablity and NRC review.

Subcooling Meter

In direct response to NUREG-0578, Item 2.1.3.b, Florida Power Corporation has purchased from Babcock & Wilcox two saturation meters and a field change package to provide wide-range T_{hot} input to these devices; T_{cold} wide-range is currently available. This meter was designed by Babcock & Wilcox to monitor plant temperature and pressure and implement, with hard wire logic, the determination of margin to saturation for present plant conditions and indicate this to the plant operator. The engineering design for installation of this equipment is proceeding on an expedited basis to meet the specified inservice date.

This design provides for two T_{sat} meters to be mounted in the PORV/T_{sat} monitoring cabinet that will be located in the 4160 V ESB SWGR Room. Each meter will receive the following inputs:

4 hot leg temperature (2 per loop) 120°F to 920°F
2 RC pressure (1 per loop) 0 to 2500 psig

These signals will be taken from the Non-Nuclear Instrumentation (NNI) System, with individual buffers to preclude interaction between T_{sat} meters or NNI/ICS. The temperature inputs are not qualified safety-grade, however, they are reliable in that this NNI provides two vital sources and signal cables are routed in seismic instrument trays.

Each meter will have a remote digital indicator/selector, mounted on the PSA section of the main control board, and a low margin to saturation alarm to the annunciator events recorder. The low margin to saturation alarms will light a common window on the PSA section of the control board with CRT and hard copy events recorder identification of loop indicating the condition. The digital indicator on the control board will have a spring return selector switch such that one meter is normally looking at Loop A, and the other is looking at Loop B, with the capability to switch for checking performance and, in the event of meter failure, the power to each meter will be from different vital sources.

Attachments I and II provide additional information concerning the design of the subcooling meters.

The Construction Work Package has been issued to the site for the T_{sat} Meter modification, and the equipment is on-site, with the exception of the PORV/T_{sat} monitoring cabinet (see Item 2.1.3.a).

Recommendation 2.1.4 Diverse Conta t Isolation

Florida Power Corporation, in its April 12, 1979 response to Item 6 of IE Bulletin 79-05A, identified essential and nonessential systems with regard to containment isolation and core cooling. Essential systems were defined as those systems which are required for core cooling

capability and, therefore, should not be isolated on automatic HPI actuation. For the valves listed in our April 12 response, which receive no ES signal and are normally closed and remain closed following the accident conditions, no further action is required.

The nonessential valves, listed in our response, which receive a containment isolation signal (4 psig RB pressure) will be provided with a diverse containment isolation parameter with the addition of an auto-close isolation signal, based on automatic HPI actuation. These diverse containment isolation signals will satisfy safety-grade requirements and resetting of these signals shall not result in the automatic loss of containment isolation.

The Construction Work Package was issued to the site on January 11, 1980. Installation of this modification will require approximately 4-5 weeks, as it will be accomplished with the unit on-line. This modification will be installed and tested on or before February 15, 1980.

Recommendation 2.1.5.a Dedicated Penetrations for External Recombiners or Post-Accident Purge Systems

The present CR-3 design has installed a redundant, dedicated, hydrogen purge system. The CR-3 system uses two penetrations, dedicated to hydrogen purge only, which are sized consistent with the flow requirements of the purge system. The CR-3 purge system is single failure proof. Therefore, we conclude that the existing hydrogen purge system satisfies the requirements of Section 2.1.5.a of NUREG-0578.

The hydrogen purge system is described in Section 14B of the CR-3 FSAR and shown on Flow Diagram FD-302-722.

Recommendation 2.1.5.c Recombiner Procedure

CR-3 does not have a requirement for hydrogen recombiners as a design basis for licensing. Therefore, this requirement does not apply to CR-3.

Recommendation 2.1.6.a Integrity of Systems Outside Containment Likely to Contain Radioactive Material for PWRs and BWRs

Prior to issuance of the NUREG-0578 requirements, FPC had a leak reduction program implemented to satisfy the leakage rate requirement identified in the CR-3 Technical Specifications. This program is described and implemented by SP-317--RC System Water Inventory Balance. Since receipt of NUREG-0578, the CR-3 program has been expanded to meet these new requirements beyond the CR-3 Technical Specifications.

The present program includes the following systems:

1. RC Bleed Line
2. Waste Gas Disposal System
3. Decay Heat
4. Building Spray
5. Make Up

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6. High Pressure Injection
7. RCS Sample Lines.

This leak reduction program is described in SP-716--Sample Line Leak Rate Test, SP-412--ECCS and Containment Spray System Leak Rate Test, SP-429--Waste Gas System Leak Rate Test, PT-108--Decay Heat Removal and Reactor Building Spray System Leak Rate Test, and PT-10_--RC Bleed Line Leak Rate Test. Copies of these procedures were submitted on November 17, 1979, for your review, except for SP-716, which is being written to include some additional sample lines in the program. A copy of SP-716 is being written in place of SP-317 for this program, and will be submitted as soon as it is issued. To-date, all of the above systems have been leak tested except for the Decay Heat and Building Spray Systems and the Waste Gas Storage Tanks. We are presently repairing leaks in the Decay Heat System at CR-3. Upon receipt of parts to complete this effort and the leaks are repaired, the Decay Heat and Building Spray Systems will be leak tested together. The leak test of the Waste Gas Storage Tanks will require a 2-week outage, and will be performed during our April, 1980, refueling outage. Results of these additional leak tests will be submitted upon completion of this effort.

Recommendation 2.1.6.b Design Review of Plant Shielding and Environmental Qualifications of Equipment for Spaces/Systems Which May Be Used in Post-accident Operation

Enclosed as Attachment III is the GAI Report prepared for Florida Power Corporation, entitled "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post-accident Operations Outside Containment at Crystal River Unit 3 Nuclear Generating Station". This Report, which is under review by Florida Power Corporation, is being submitted in response to the September 13, 1979, letter from Darrel G. Eisenhut, requesting implementation of NUREG-0578, Item 2.1.6.b, and clarified in Mr. Harold R. Denton's letter of October 30, 1979. Following our review, any modifications to this report will be submitted as soon as possible.

Recommendation 2.1.7.a Auto-Initiation of Auxiliary Feedwater System (AFWS)

This recommendation has been excluded from NUREG-0578 and will be addressed by the Bulletins and Orders Task Force.

Recommendation 2.1.7.b Auxiliary Feedwater Flow Indication to Steam Generators

1. Short-term Control Grade:

   a. Emergency feedwater flow indication to each steam generator satisfies the single failure criteria because there are ultrasonic flow indications on each steam generator with a backup steam generator level indication on each steam generator.

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b. The present ultrasonic flow indication channels are testable by electronically verifying the zero and circuit fault conditions for each unit.

c. The present emergency feedwater flow indicating devices are powered from vital buses with a battery-backed inverter.

2. Long-term Safety-Grade:

a. We are in the process of evaluating the present equipment for upgrading to safety-grade, as well as evaluating other alternate methods of emergency feedwater flow measurement.

3. Other:

a. The short-term control grade flow indication channels satisfy the single failure criteria because each steam generator has an ultrasonic flow indicator and steam generator level indication.

b. Ultrasonic flow indicators were factory-calibrated as a matched system (transducers and flow display computers) at 740 gpm and has an accuracy of about 2%.

Recommendation 2.1.8.a Improved Post-accident Sampling Capability

We are currently conducting a design and procedure review regarding post-accident sampling at CR-3. Florida Power Corporation has hired Applied Physics Technology (APT) and GAI to assist us with this effort. These reviews and a report describing the review and corrective actions will be submitted to you as soon as possible, but no later than Feburary 15, 1980.

Recommendation 2.1.8.b Increased Range of Radiation Monitors

We are presently developing interim procedures for the estimation of high level accidential radioactive releases, if instrumentation goes off-scale. Additional information concerning this effort will be submitted as soon as possible. We are utilizing APT and GAI for this item also. Completion of this effort is scheduled for on or before February 15, 1980.

Recommendation 2.1.8.c Improved In-Plant Iodine Instrumentation

CR-3 presently has six portable air samplers and procedures in place for obtaining and determining airborne iodine concentrations using spectral analyses. Therefore, we currently satisfy the requirements of this section.

Recommendation 2.1.9 Transient and Accident Analyses

	Task Description	Status
1.	Small Break LOCA analysis and preparation of emergency procedure guidelines	Complete

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Task Description	Status
2. Implementation of Small Break LOCA emergency procedures and retraining of operators	Complete
3. Analysis of inadequate core cooling and preparation of emergency procedure guidelines	See response to Recommendation 2.1.3.b
4. Implementation of emergency procedures and retraining related to inadequate core cooling	See response to Recommendation 2.1.3.b
5. Analysis of accidents and transients and preparation of emergency procedure guidelines	Late 1980
6. Implementation of emergency procedures and retraining related to accidents and transients	6 months after guidelines established
7. Analysis of LOFT small break tests	*

*By letter dated December 31, 1979, J. H. Taylor, Manager, Licensing, Babcock & Wilcox, provided the "B&W LOFT L31 Pretest Prediction Report" to the NRC Staff.

RCS Venting

Enclosed as Attachment IV, is the B&W Report, entitled "177 High Point Vent Design Criteria".

This Report provides the design criteria and other design input which will be used by Florida Power Corporation and our A/E to complete the design. Included are:

1. Venting System Design Criteria: This includes the requirement that the vent piping and valving be sized (approximately 1/2") such that failure of a line does not cause coolant loss in excess of normal makeup system capacity. The actual sizing requirements will be a direct function of the CR-3 actual makeup rate.

2. Venting System Schematic.

3. Vent Flow Rate Curves, showing mass and volume flow rates as a function of system resistance, since the actual resistance will be determined by the hardware installed.

4. Anchor Motions, Seismic Response Spectra, and Allowable Nozzle Loads.

The B&W requirements for installation of high point vents on the Reactor Coolant System are summarized as follows:

1. Remotely operable high point vents are required at the top of each hot leg and on top of the pressurizer.

2. The vent line size (approximately 1/2") will be such that the flow rate will be less than the capacity of a makeup pump, thus eliminating a LOCA analysis.

3. The discharge will be to the containment.

4. Each vent line will have two isolation valves powered from the same IE power supply and operated remotely by separate switches.

The Design Criteria are such that a LOCA analysis is not required. The criteria limit the size of the vent piping so normal makeup capability can accommodate the outflow of a break in the vent line. This is consistent with the B&W position on breaks in instrument lines.

In addition to the two new high point reactor coolant vents, the gases that accumulate in the pressurizer can be vented by use of the PORV presently installed on the pressurizer. We do not propose to add a remotely operated vent to the reactor head, since any accumulation of gases sufficient to fill the reactor vessel volume will be vented via the hot leg vents, due to the free path available.

Florida Power Corporation is presently reviewing this B&W report, and will, with the assistance of GAI, complete the final design of this modification. This final design work will include the designation of appropriate power supplies to the vent valves, piping routing, and hydrogen gas concentration limits. Additional information concerning the detailed design and schedule for procurement and installation will be submitted as soon as possible.

Recommendation 2.2.1.a Shift Supervisor Responsibilties

The responsibilities of the Shift Supervisor have been defined in Administrative Instruction AI-500, Conduct of Operations and Management Directives. Copies of AI-500 and our Management Directive concerning the Shift Supervisor are enclosed in Attachment V and VI, respectively. Florida Power Corporation considers this item completed.

Recommendation 2.2.1.b Shift Technical Advisor

The on-shift Technical Advisor to the Shift Supervisor will be provided as follows:

1. Short-term Plan:

By January 1, 1980, Shift Technical Advisors (STAs) are on-shift to provide accident assessment. The compliment is provided by current plant personnel who meet the requirements identified in

paragraphs A.1, 2, and 3 of Enclosure 2 of Darrel G. Eisenhut's letter of September 13, 1979.

The Shift Technical Advisors are assigned for 24-hour periods, are on the plant site, and will remain within 10 minutes of the Control Center. The STAs will be in the Control Center for planned major plant evaluations. They are independent from supervision of the manipulation of plant controls. In addition to performing the function of the STA, they will be performing the functions of their normal positions. To provide the operating experience assessment function, contract personnel are located on-site, dedicated to evaluation of plant operations for potential safety implications. The STAs and contract personnel will provide feedback to one another, on a current basis, on the nature and results of their assessments. This plan will be in effect until the necessary personnel can be hired and trained.

2. Long-term Plan:

Establish Nuclear Operations Technical Advisor positions to provide both the accident assessment and operating experience functions. These Advisors would meet the intent of the requirements identified in Enclosure 2 of Darrel G. Eisenhut's letter of September 13, 1979.

As of October 29, 1979, the Nuclear Operations Technical Advisor positions were approved. The position descriptions are developed and recruitment of personnel is underway. The Nuclear Operations Technical Advisors will undergo training to bring them up to the level of expertise required prior to being placed on-shift to perform the STA function.

Recommendation 2.2.1.c Shift and Relief Turnover Procedures

The shift and relief turnover procedures are defined in Enclosures 9, 10, and 11 of Administrative Instructions AI-500, Conduct of Operations (copy enclosed). Florida Power Corporation considers this item completed.

Recommendation 2.2.2.a Control Room Access

Access to the CR-3 Control Room is handled in accordance with AI-500 and AI-200, Organization and Responsibility (Attachment VII). Florida Power Corporation considers this item completed.

Recommendation 2.2.2.b On-Site Technical Support Center

A Technical Support Center (TSC) is established in the office building located on the northwest corner of the turbine building. This is the normal storage and retrieval area for those drawings and records described in ANSI N45.2.9-1974. The TSC will provide assistance to the operating personnel in evaluating the course of an incident or accident and will also be the designated point of contact with offsite agencies (after activation), in providing advice on the expected

course of the accident. This area cannot be designated as the permanent TSC, due to the requirement for the TSC to be habitable to the same degree as the Control Room for postulated accident conditions. The designated temporary location is habitable, provided with two conference rooms, capable of supporting 15 - 20 assigned personnel. Portable monitoring equipment for measuring radiation levels in the TSC is provided. Action level criteria (EM-102) have been developed to define when protective measures (evacuation to the Control Room) should be taken.

Dedicated telephone communications have been provided to allow reliable communications between the TSC and Control Room, and the NRC. Dedicated telephone communications will be provided between the TSC and the Emergency Operations Center (EOC), once the EOC is established, prior to mid-1980. In addition, nondedicated telephone lines and interplant communication systems are available for additional communications.

Plant parameters necessary for assessment have been provided by a computer printout, located in the TSC and paralleled with the control room printer.

Plans for staffing the TSC during emergency situations, and for performing this accident assessment function from the control room should the TSC become uninhabitable, are developed and will be implemented by January 18, 1980. These plans are covered by EM-102.

Attachment VIII to this letter provides Florida Power Corporation's general long-range plant to upgrade the TSC at CR-3. Additional details will be submitted at a later date.

Recommendation 2.2.2.c On-site Operational Support Center

In order to provide an on-site assembly area where assigned support personnel will report in the event of an accident or emergency situation the following actions have been taken by CR-3:

a. The north end of the shop facilities building, located northeast of the control complex, has been designated as the Operational Support Center (OSC). This choice of locations allows ready access to the control complex and utilizes the existing control complex personnel radiation shielding to reduce potential radiation exposures during accident conditions which require manning of the OSC.

b. The designated location is habitable and provided with washroom and facilities to support 25 assigned personnel.

c. The interplant communication system is provided to allow communication between the OSC, Control Room, and Technical Support Center.

ATTACHMENT I

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	Normal-T-Tsat Manual-P-Psat
Display Type (Analog, Digital, CRT)	Digital
Continuous or On Demand	Continuous
Single or Redundant Display	Redundant 1/Loop A 1/Loop B
Location of Display	Main Control Board
Alarms (include setpoints)	T-Tsat (selectable)
Overall uncertainty (°F, PSI)	±4°F, --
Range of Display	±4096
Qualifications (seismic, environmental, IEEE323)	See Attachment

Calculator

Type (process computer, dedicated digital, or analog calc.)	Dedicated Digital
If process computer is used specify availabilty (% of time)	---
Single or redundant calculators	Single meter
Selection Logic (highest T., lowest press)	Manual
Qualifications (seismic, environmental, IEEE323)	See Attachment
Calculational Technique (Steam Tables, Functional Fit, ranges)	Steam Tables

Input

Temperature (RTD's or T/C's)	RTD's
Temperature (number of sensors and locations)	8, $2R_x$ Out/Loop+$2R_x$ In/Loop
Range of temperature sensors	R_x Out 120°-190°, R_x In 50° to 650°

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Uncertainty* of temperature sensors (°F at 1)	
Qualifications (seismic, environmental, IEEE323)	See Attachment
Pressure (specify instrument used)	RC-3A + 3B - PT3
Pressure (number of sensors and locations)	2, one R_x Outlet/Loop
Range of Pressure sensors	0-2500 psig
Uncertainty* of pressure sensors (PSI at 1)	
Qualifications (seismic, environmental, IEEE323)	PT qualified to Engineered safeguards system requirements described in the FSAR

Backup Capability

Availability of Temp & Press		Two meters, loop A&B, with manual selection opposite loop.
Availability of Steam Tables, etc.	Covered by small break operating procedures & training	
Training of operators		
Procedures		

*Uncertainties must address conditions of forced flow and natural circulation.

1755 023

ATTACHMENT II.

BABCOCK & WILCOX SATURATION METER - IMPLEMENTATION OF NRC REQUIREMENTS

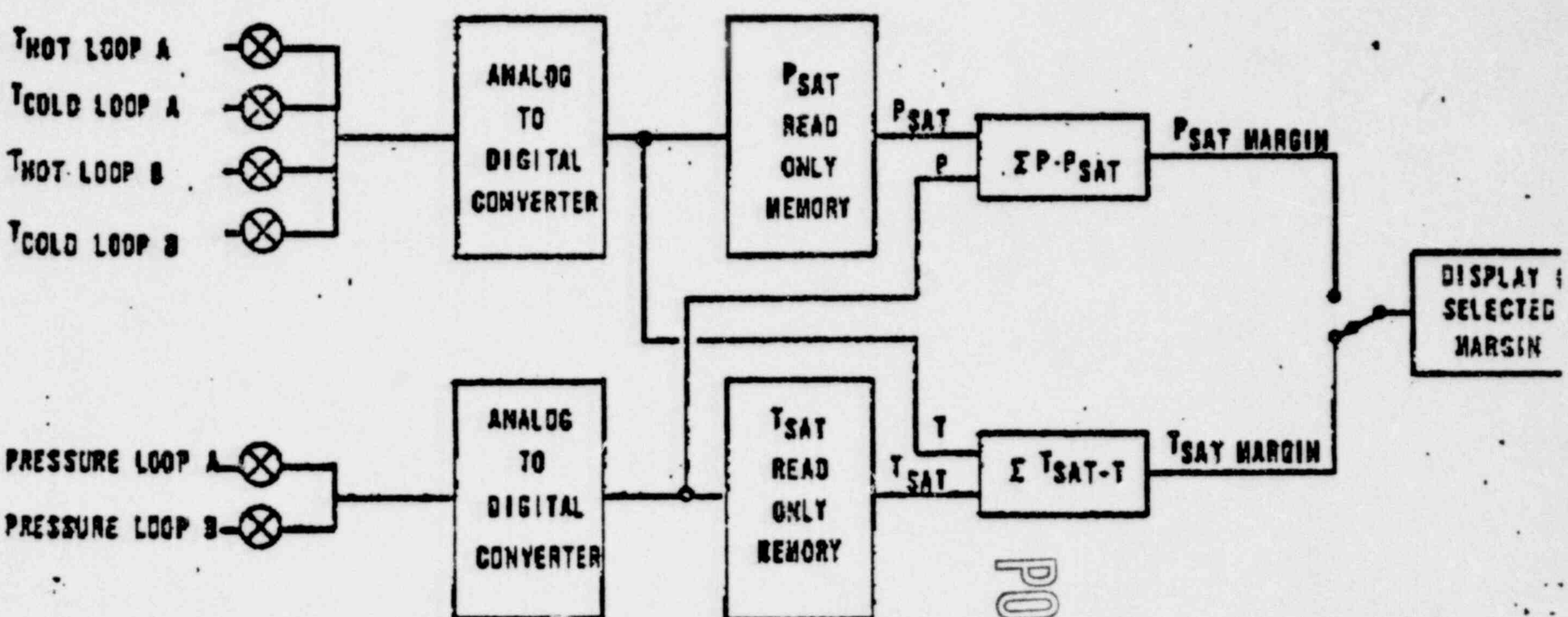
The Saturation Meter, designed by B&W to meet the requirements of NUREG-0578, accepts existing inputs of plant temperature and pressure, looks up in a stored steam table the values of saturation temperature and pressure and determines margin to saturation conditions from present plant conditions. This margin is displayed to the power plant operator on a digital panel meter. Saturation conditions are indicated by the meter displaying zero (00) margin. Degrees of superheat are displayed as a negative margin should temperature exceed T_{sat}.

Method Of Implementation

A block diagram of B&W's Saturation Meter is shown in figure 1. The values of T_{sat} and P_{sat} from the 1967 ASME steam tables have been stored in non-volatile semiconductor read only memory. Incoming analog pressure and temperature signals are converted to digital signals using single chip analog to digital (A/D) converters. The output or digital signal from the A/D converters addresses a read only memory in which the Saturation values from the steam tables are stored. Thus, the incoming signals from the plant "look up" a Saturation Valve from a stored table. The current temperature is then digitally subtracted from the value of Saturation temperature to give T_{sat} margin and P_{sat} is subtracted from the current pressure to give P_{sat} margin. Either P_{sat} margin or T_{sat} margin is displayed on a digital meter depending on operator selection.

It should be noted that B&W's implementation of the Saturation Meter does not use a microporcessor or other programmable logic device. It is implemented with hard wired logic. The only "calculation" done is the subtraction to determine margin. This is also done with hard wired logic.

Since the meter uses hard wired logic, its operation is easy to confirm that it is operating properly. In addition, every signal in the meter can be tested as discussed further below.



POOR ORIGINAL

FIGURE 1
BABCOCK & WILCOX
SATURATION METER
BLOCK DIAGRAM

1755 025

Compliance with Requirements

In the regional NRC meeting in Atlanta, Georgia, on September 28, 1979, discussions were held concerning NRC requirements for implementation of NUREG-0578, and particularly, the NRC definition of safety grade as related to certain control room instrumentation such as the Saturation Meter. The NRC requirements as stated in this meeting and B&W's response to meet these is summarized below:

A. NRC Requirement: Single Failure Proof

- The design of the T_{sat} Meter considered the most likely failure to be loss of the internal clock. A circuit was incorporated which will detect loss of the clock at the remote display and will blank the display.
- The capability was incorporated in the design to switch to alternate input sensors should a sensor fail. In addition, isolation resistors in series with each input prevent a failure in one input from affecting another.
- B&W is recommending that redundant power sources be provided to the Saturation Meter.
- B&W is recommending that redundant meters be installed, or that operator consultation of steam tables be used as redundancy.

B. NRC Requirement: System Shall be Testable

Since the B&W T_{sat} Meter does not use a microcomputer or any other programmable calculating machine. Input signals simply "look-up" saturation values in a read only memory, and margin is calculated in a hard wired logic adder. A test display module is included in the equipment rack with each T_{sat} meter. The test module has a digital display which allows an operator or technician to check every signal internal to the meter. These include the input temperatures and pressures in °F and Psig, the saturation pressure

and the temperature that the input signals look up, and the margin to T_{sat} and margin to P_{sat} as determined by the meter. Since the meter is implemented with hard wired logic, there are no other intermediate answers calculated internal to the machine. Thus, the capability to check every signal in the meter has been provided.

C. Qualification - Seismic

A shaker test was performed on the Saturation Meter. Each circuit board was instrumented with an accelerometer and the entire assembly was excited at a level of 1 g, sinusoidal. The frequency of excitation was increased from 5 Hz to 33 Hz at a rate of 1 octave per minute to search for resonances. None were found. The excitation was held at 33 Hz for 5 minutes. The assembly was then excited at 10 g peak acceleration at 20 Hz for 5 minutes duration.

No damage was done to the meter, and testing subsequent to the shaker test confirmed that it functioned properly.

D. Qualification - Environmental

Radiation - The T_{sat} meter was designed to accept input from existing plant instrumentation. Qualification of these instruments to 1E requirements including radiation has already been performed. The equipment supplied by B&W is for installation in an electronic equipment room and the control room. No radiation testing has been performed. 1755 027

Temperature - Analog circuitry in the T_{sat} meter has been tested to $60^{\circ}C$ with results as shown in figure 2. An error analysis performed using worst case errors in the analog circuitry plus quantitation errors in the digital circuitry showed a worst case error of $\pm 4^{\circ}F$. That is, the value of T_{sat}

margin displayed to the operator could be in error $\pm 4^{\circ}F$ from a value he would obtain by looking up T_{sat} for the indicated control room pressure and subtracting the indicated temperature.

For the digital circuitry all components used are rated at $60^{\circ}C$ or higher.

__Humitity__ - No humidity tests were performed. All components used are hermetically sealed. Specific care was taken in the design of the meter to eliminate components such as carbon resistors whose values are sensitive to humidity The only resistors used are used for digital logic pull up where exact values are not critical.

ATTACHMENT III

DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS OUTSIDE CONTAINMENT AT CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING STATION

PREPARED FOR

FLORIDA POWER CORPORATION

PREPARED BY

GILBERT/COMMONWEALTH
GILBERT ASSOCIATES, INC.
READING, PENNSYLVANIA
DECEMBER 1979

1755 029

1.0 INTRODUCTION

This report has been prepared in response to the September 13, 1979, letter from Darrel G. Eisenhut of the NRC to all operating nuclear power plants. This letter presented an implementation schedule for the recommendations presented in "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" (NUREG-0578). The requirements defined in the September 13th letter were subsequently clarified in a letter from Harold R. Denton to all operating nuclear power plants dated October 30, 1979.

Among the requirements defined in the two NRC letters is a review to determine whether post-accident radiation fields unduly limit personnel access to areas necessary for mitigation of or recovery from an accident or unduly degrade the proper operation of safety equipment. Corrective actions for problems identified as a result of the review are also to be determined. This report presents the results of such a review for the Crystal River Unit 3 Nuclear Generating Station.

The review was based on the following guidelines:

a) The post accident dose rate in areas requiring continuous occupancy should not exceed 15 mr/hr.

b) The post accident dose rate in areas which do not require continuous occupancy should be such that the dose to an individual during a required access period is less than 5 Rem whole body or its equivalent.

c) The integrated dose to safety equipment should be less than the dose that the equipment has been qualified for.

d) The minimum radioactive term used in the evaluation should be equivalent to the source terms recommended in Regulatory Guides 1.3 and 1.4.

2.0 SUMMARY

Reviews have been performed of: a) the areas requiring access for post-accident operations; and b) the radiation qualification of safety equipment outside containment.

The areas identified as requiring access are:

nuclear sample room
hydrogen purge equipment
containment air monitor RM-A6
diesel generator room
radioactive waste disposal control board
control room
radiochemistry laboratory
count room

Considerations associated with the nuclear sample room, radiochemistry laboratory, and count room are addressed in the response to NRC Lessons Learned Task Force Short Term Recommendation 2.1.8.a (Improved Post-Accident Sampling Capability). The evaluations performed for the other areas indicate that with the exception of the radioactive waste disposal control board access is possible for the estimated occupancy. The radioactive waste disposal control board will be moved to a low radiation area to assure access to it.

An additional result of the review is that certain modifications are planned which will make it unnecessary to have post-accident access to certain areas. These modifications include:

- Changing valves DHV-7, 8, 39 and 40 to motor operated valves.

- Adding a motor operated bypass valve for the Makeup and Purification System prefilter.

It is not possible to reach conclusions as to the acceptablility of the integrated doses calculated during the review of the radiation qualification of safety equipment. This is due to the lack of available data on: 1) qualification doses; and 2) the period of time post accident that safety equipment must be functional. Work is continuing to resolve these concerns. The period of time post accident that each item of safety equipment must be functional will be defined making it possible to calculate an integrated dose for the time period appropriate for the function of the equipment. These doses will then be compared to qualification doses where available. For safety equipment with no available qualification data, the feasibility of developing the data or providing appropriate dose mitigating measures (shielding, equipment relocation, etc.) will be pursued.

3.0 METHODS

3.1 SOURCE TERMS

The activity assumed for liquid source term calculation is based on 100% of the noble gas inventory, 50% of the halogen core inventory and 1% of all other nuclides in the core inventory. The activity assumed for gaseous source term calculation is based on 100% of the noble gas core inventory and 25% of the halogen core inventory.

Two liquid source terms were used in the evaluation. For systems which contain post accident recirculation fluid, the source term was based on diluting the liquid inventory discussed in the previous paragraph with the expected volume of fluid in the bottom of the reactor building post accident. For systems which can contain fluid from the reactor coolant system but do not take suction on the recirculation sump, the source term was based on diluting the liquid inventory discussed in the previous paragraph with the volume of fluid in the reactor coolant system.

Gaseous source terms were determined for containment and for the waste gas system. The containment airborne source term was based on diluting the gaseous inventory discussed previously with the air contained in the containment free volume. The waste gas system source term was determined by taking 100% of the noble gases and 50% of the halogens in the makeup tank, assuming it to be full of reactor coolant as discussed above, and diluting that inventory in one-half the makeup tank volume.

Table 3-1 presents the inventories and source terms discussed above for the time period immediately after the postulated accident (T=0). For other time periods, the decay parameters given in Refs. 1 and 2 were used to adjust the source terms for radioactive decay.

3.2 CALCULATION OF DOSE RATES

Dose rates for the areas of interest in this review were calculated by determining the potential contributing sources at a representative location. The appropriate source term data for these sources was selected from Table 3-1 and adjusted for decay as required. The dose rate at the representative location was used as the general area dose rate. Both the SDC Code (Ref. 3) and the Gilbert/Commonwealth developed SPOT1 Code were used in performing the dose rate calculations. The SPOT1 Code uses the methodology originally presented by Ōno and Tsuruo (Ref. 4) and developed by Shure and Wallace (Ref. 5). Energy groups required as input to the codes were determined using the gamma ray energy and intensity data in Refs. 1 and 2 for the nuclides in Table 3-1.

3.3 CALCULATION OF DOSES TO PERSONNEL DURING POST ACCIDENT ACCESS TO VITAL AREAS

Personnel doses received in performing a given operation in a given vital area are calculated as the sum of the doses received during travel to and from the vital area and the dose received while performing the given operation in the vital area.

The doses received during travel are determined by calculating dose rates at selected locations (or at a single location if the dose rate along the travel route is relatively uniform) along the travel route using the methodology discussed in Section 3.2. and multiplying the dose rates by The appropriate travel time for each selected location along the travel route.

Doses received while performing a given operation are determined by multiplying the dose rate for the given area by the time required to perform the operation. Dose rates for the given vital area are determined using the methodology discussed in Section 3.2.

3.4 CALCULATION OF INTEGRATED DOSES TO SAFETY EQUIPMENT

The integrated dose to a given item of safety equipment is determined by integrating the dose rate appropriate for the given item over the time period that it is required to be available to perform its safety function. Dose rates are calculated using the methodology disucssed in Section 3.2.

Table 3-1

Shielding Source Terms (T=0)

Isotope	Liquid[1] Source Activity (Ci)	Gaseous[2] Source Activity (Ci)	Containment Sump Concentration (µCi/cc)	Reactor Coolant Concentration (µCi/cc)	Containment Airborne Concentration (µCi/cc)	Waste Gas Concentration (µCi/cc)
Br-84	7.85 + 6	3.93 + 6	4.24 + 3	2.43 + 4	6.94 + 1	2.44 + 4
Kr-83m	9.25 + 6	9.25 + 6	4.99 + 3	2.87 + 4	1.63 + 2	5.74 + 4
Kr-85m	2.19 + 7	2.19 + 7	1.18 + 4	6.79 + 4	3.87 + 2	1.35 + 5
Kr-85	5.30 + 5	5.30 + 5	2.86 + 2	1.64 + 3	9.36 + 0	3.28 + 3
Kr-87	4.00 + 7	4.00 + 7	2.16 + 4	1.24 + 5	7.06 + 2	2.48 + 5
Kr-88	5.60 + 7	5.60 + 7	3.02 + 4	1.74 + 5	9.89 + 2	3.48 + 5
Rb-88	5.64 + 5	-	3.04 + 2	1.75 + 3	-	-
Sr-89	7.42 + 5	-	4.00 + 2	2.30 + 3	-	-
Sr-90	3.99 + 4	-	2.15 + 1	1.24 + 2	-	-
Sr-91	9.72 + 5	-	5.25 + 2	3.01 + 3	-	-
Sr-92	9.50 + 5	-	5.13 + 2	2.94 + 3	-	-
Y-90	3.96 + 4	-	2.14 + 1	1.23 + 2	-	-
Y-91	9.85 + 5	-	5.32 + 2	3.05 + 3	-	-
Mo-99	1.28 + 6	-	6.91 + 2	3.97 + 3	-	-
Ru-106	2.29 + 5	-	1.24 + 2	7.10 + 2	-	-
Xe-131m	4.38 + 5	4.38 + 5	2.36 + 2	1.36 + 3	7.73 + 0	2.72 + 3
Xe-133m	3.07 + 6	3.07 + 6	1.66 + 3	9.51 + 3	5.42 + 1	1.91 + 4
Xe-133	1.27 + 8	1.27 + 8	6.85 + 4	3.93 + 5	2.24 + 3	7.87 + 5
Xe-135m	3.26 + 7	3.26 + 7	1.76 + 4	1.01 + 5	5.76 + 2	2.02 + 5
Xe-135	2.09 + 7	2.09 + 7	1.13 + 4	6.48 + 4	3.69 + 2	1.29 + 5
Xe-138	1.17 + 8	1.17 + 8	6.31 + 4	3.63 + 5	2.07 + 3	7.26 + 5

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Isotope	Liquid[1] Source Activity (Ci)	Gaseous[2] Source Activity (Ci)	Containment Sump Concentration (µCi/cc)	Reactor Coolant Concentration (µCi/cc)	Containment Airborne Concentration (µCi/cc)	Waste Gas Concentration (µCi/cc)
I-131	3.68 + 7	1.84 + 7	1.99 + 4	1.44 + 5	3.25 + 2	1.14 + 5
I-132	4.31 + 7	2.16 + 7	2.33 + 4	1.34 + 5	3.81 + 2	1.34 + 5
I-133	6.40 + 7	3.20 + 7	3.45 + 4	1.98 + 5	5.65 + 2	1.98 + 5
I-134	8.00 + 7	4.00 + 7	4.32 + 4	2.48 + 5	7.06 + 2	2.48 + 5
I-135	6.35 + 7	3.18 + 7	3.43 + 4	1.97 + 5	5.62 + 2	1.98 + 5
Cs-134	1.27 + 4	-	6.85 + 0	3.93 + 1	-	-
Cs-136	8.02 + 3	-	4.33 + 0	2.48 + 1	-	-
Cs-137	4.99 + 4	-	2.69 + 1	1.55 + 2	-	-
Cs-138	1.23 + 6	-	6.64 + 2	3.81 + 3	-	-
Ba-137m	4.67 + 4	-	2.52 + 1	1.45 + 2	-	-
Ba-140	1.25 + 6	-	6.75 + 2	3.87 + 3	-	-
La-140	1.27 + 6	-	6.85 + 2	3.93 + 3	-	-
Ce-144	7.50 + 5	-	4.05 + 2	2.32 + 3	-	-
Cr-51	-	-	9.06 - 4	5.20 - 3	-	-
Mn-54	-	-	1.01 - 4	5.80 - 4	-	-
Mn-56	-	-	2.96 - 3	1.70 - 2	-	-
Fe-59	-	-	1.01 - 4	5.80 - 4	-	-
Co-58	-	-	5.23 - 3	3.00 - 2	-	-
Co-60	-	-	6.97 - 4	4.00 - 3	-	-
Zr-95	-	-	8.71 - 5	5.00 - 4	-	-

(1) Based on 100% noble gas core inventory, 50% halogen core inventory, and 1% of all others core inventory.

(2) Based on 100% noble gas core inventory and 25% halogen core inventory.

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4.0 REVIEW OF AREAS REQUIRING ACCESS FOR POST-ACCIDENT OPERATIONS

A review has been performed to identify areas requiring access for post-accident operations. Dose evaluations for the required access to these areas were performed, and the results are documented below.

An additional result of the review is that certain modifications are planned which will make it unnecessary to have post-accident access to certain areas. These modifications include:

a) Changing valves DHV-7, 8, 39, and 40 to motor operated valves.

b) Adding a motor operated bypass valve for the Makeup and Purification System prefilter.

4.1 AREA A

Area A is the designation for the nuclear sample room located on the 95' elevation of the auxiliary building (see Figure 4-1). For the evaluations associated with this area, refer to the responses for NRC Lessons Learned Task Force Short Term Recommendation 2.1.8.a (Improved Post-Accident Sampling Capability).

4.2 AREAS B AND C

Areas B and C are the general location of the hydrogen purge equipment which could be used for controlling hydrogen buildup in containment. These areas are located on the 119' elevation of the Intermediate Building (see Figure 4-2). Access could be required 48 hours post-accident to line up and monitor hydrogen purge equipment.

The estimated occupancy requirements to perform the necessary operations is 10 minutes. The activity inside containment is the predominant source of direct radiation for these areas. The dose rates at T=48 hours are estimated to be 350 rem/hr and 10 mrem/hr for Areas B and C, respectively. The

resulting doses are 60 and 2 mrem, respectively. Access to Areas B and C is via Travel Route 1 shown on Table 4-1. The dose for this travel route is negligible. The total doses for the required access to these areas is within the guidelines discussed in Section 1.0.

4.3 AREA D

Area D is the general location of containment air monitor RM-A6. For the evaluations associated with this area refer to the response for NRC Lessons Learned Task Force Short Term Recommendation 2.1.8.a (Improved Post-Accident Sampling Capability).

4.4 AREA E

Area E as identified on Figure 4-2 is the diesel generator room. Continuous access to this area following an accident would be required if offsite power is lost. Access to Area E is via Travel Route 2 presented on Table 4-1. The dose rate in the diesel generator room from direct radiqtion following an accident is negligible. The dose for Travel Route 2 is also negligible. Area E could thus be continuously occupied if required.

4.5 RADIOACTIVE WASTE DISPOSAL CONTROL BOARD

The radioactive waste disposal control board is presently located on the 95' elevation of the auxiliary building. Access to operate waste systems post-accident is required. This access will be provided by relocating the control board to a low radiation area.

4.6 CONTROL ROOM

The control room is located on the 145' elevation of the control complex. Continuous and immediate access is required post accident. Access to the control complex is via Travel Route 3 shown on Table 4-1. The dose rate in the control room from direct radiation from containment is estimated to be 11 mr/hr at T=0. The dose for Travel Route 2 is negligible. These values would thus allow continuous occupancy of the control room.

4.7 RADIOCHEMISTRY LABORATORY AND COUNT ROOM

The radiochemistry laboratory and count room are located on the 95' elevation of the control complex. Access is required for analysis of nuclear samples. Access is via Travel Route 3 shown on Table 4-1. The dose rate in these areas from sources other than those associated with sampling is negligible. The dose for Travel Route 3 is also negligible. For the evaluation of the sources associated with sampling refer to the response for NRC Lessons Learned Task Force Short Term Recommendation 2.1.8.a (Improved Post-Accident Sampling Capability).

TABLE 4-1

Travel Routes

Travel Route	Description
1	From the control room in the control complex at elevation 145 down to elevation 124 and north into the turbine building, down to elevation 119 and west in the turbine buildint to access the intermediate building at elevation 119.
2	Access from the southeast corner of the auxiliary building.
3	Access from office building to turbine building to control complex.

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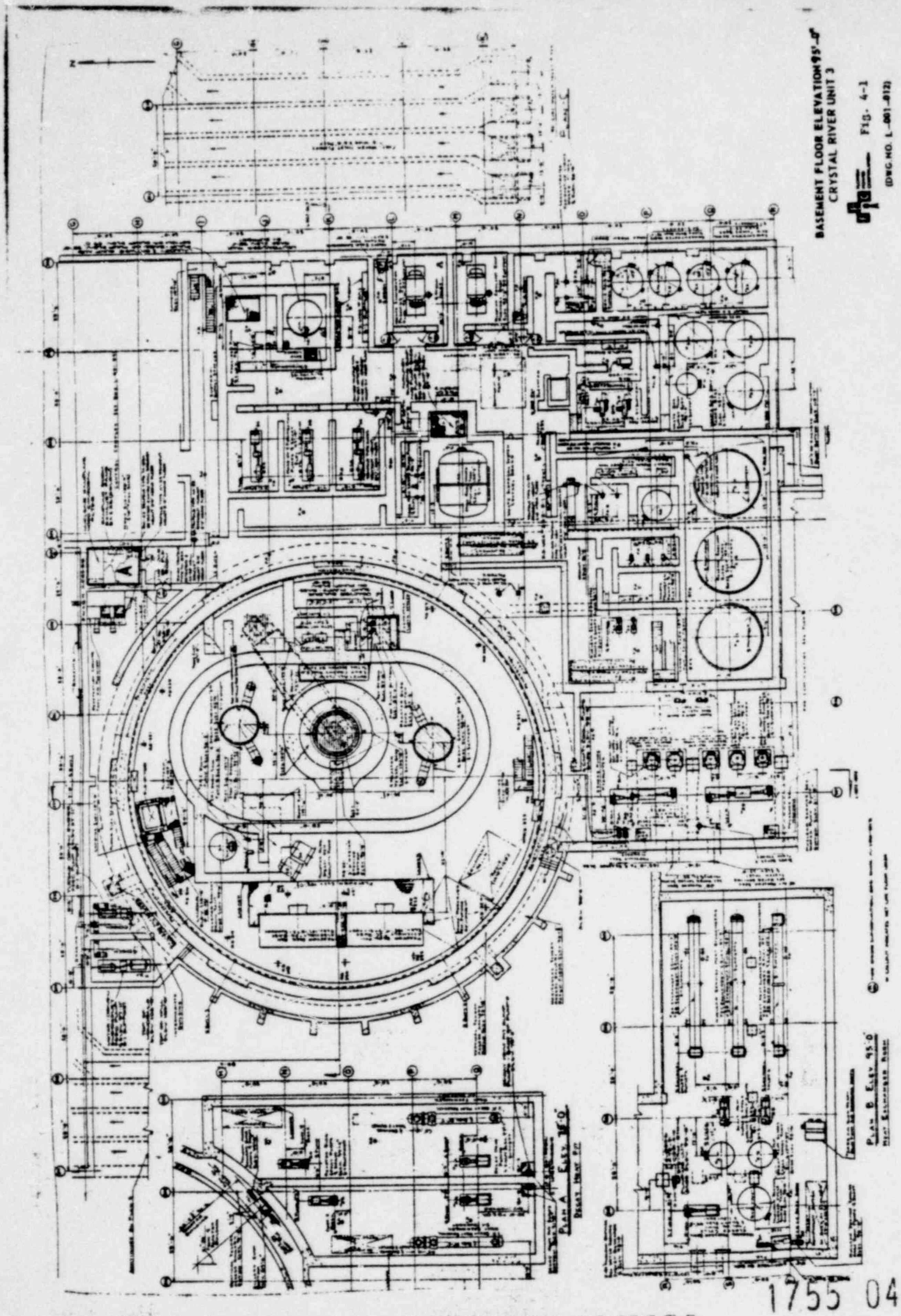

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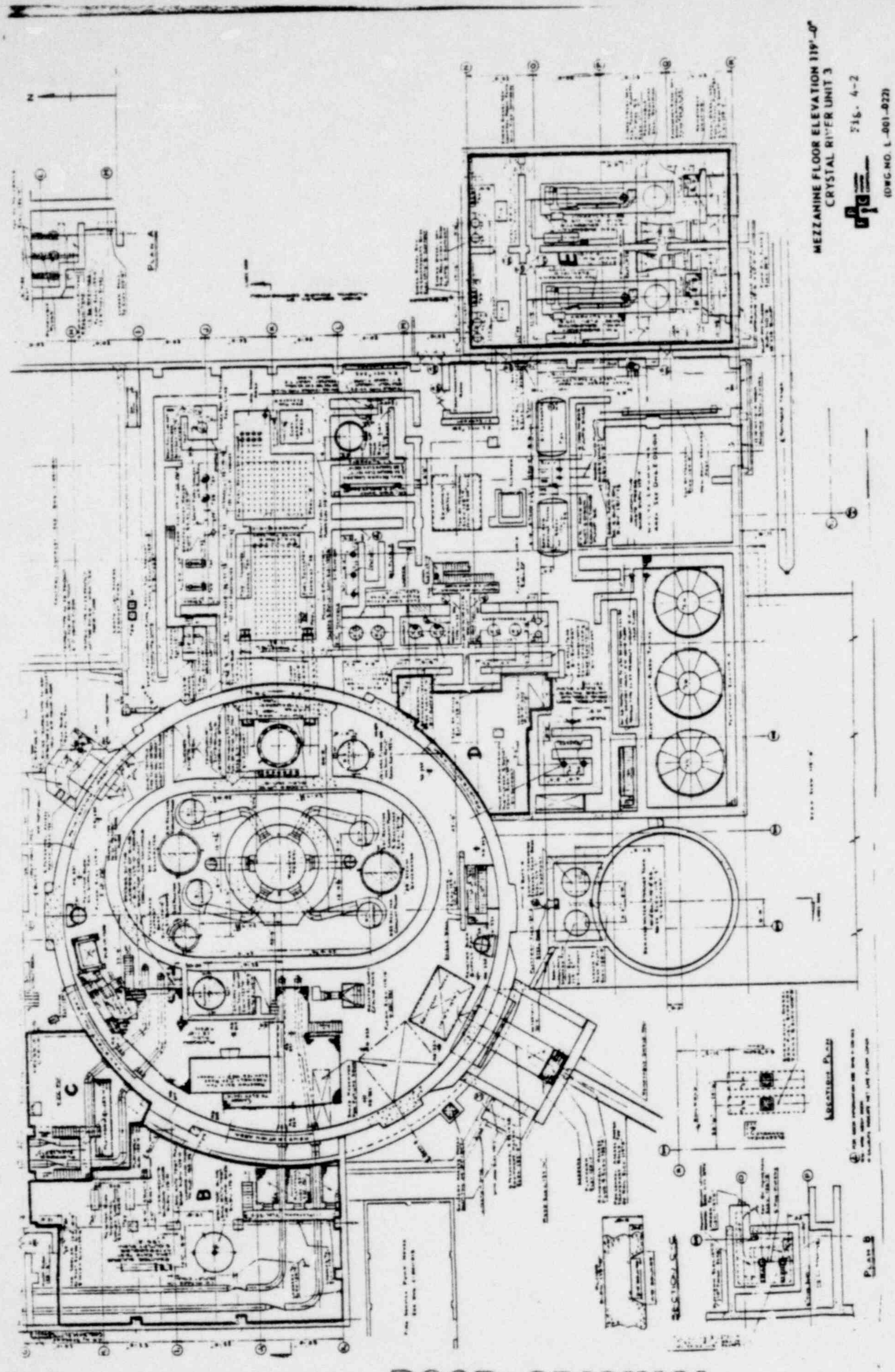

MEZZANINE FLOOR ELEVATION 119'-0"
CRYSTAL RIVER UNIT 3
Fig. 4-2
(DWG-NO. L-001-022)

POOR ORIGINAL

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5.0 REVIEW OF RADIATION QUALIFICATION OF SAFETY EQUIPMENT

Table 5-1 identifies the approximate location of the safety equipment considered in the review. The table also presents the radiation sensitive materials in each item and the integrated dose calculated for the six month period following the accident at the given location.

Table 5-2 lists the radiation sensitive materials identified in Table 5-1 and radiation damage information for the stated reference materials of similar composition.

It is not possible to reach conclusions as to the acceptability of the integrated doses calculated during the review of the radiation qualification of safety equipment. This is due to the lack of available data on: 1) qualification doses; 2) the period of time post accident that safety equipment must be functional.

TABLE 5-1

Equipment Name	Identification Number	Location Building	Location Elevation	Location Column	Calculated Integrated Dose (6 mon) RADS	Radiation Sensitive Material
MUV 23/24 MCC		Aux	95	304/J	5.3 + 5	
MUV 25/26 MCC		Aux	95	304/L	5.3 + 5	
Sample Room		Interm	95	304/H	5.3 + 3	
MUP Pump Filter (2)	MU-FL3A,B					Ethyl-PPRO Seals, Epoxy Impreg Cellulose Fib. Filt.
Seal Return Cooler (2)	MU-HE2A,B	Aux	119	302/M	7.528 + 4	
Makeup Pump (3)	MU-P1A,B,C	Aux	95	303/J,K	1.475 + 5	
Letdown Cooler Isolation Valve	MU-V49F	Aux	95	304/K	3.687 + 6	316+Asbestos Gasket, JC1871CR Packing
Letdown Cooler Shutoff Valve	MU-V50F	Aux	119	304/L	1.835 + 6	Evalpak 187-1X Packing + Pist
Bypass Control Valve to MUDIII-1A	MU-V194	Aux	119	305/D	1.144 + 6	Asbestos
MU-FL2B Inlet Isolation Valve	MU-V243	Aux	119	305/M	1.373 + 6	Asbestos + Pist
MU-FL2B Discharge Isolation Valve	MU-V244	Aux	119	305/M	1.373 + 6	Asbestos + Pist
Cation Demin Cross Tie Isolation Valve	MU-V144	Aux	119	304/N	8.728 + 5	Asbestos + Pist
Bypass Isolation Valve to MU-F	MU-V126	Aux	119	304/M	8.728 + 5	Asbestos + Pist
Isolation Valve to MU-F	MU-V116F	Aux	119	304/L	1.835 + 6	JC-187-I-CR Asbestos Packing, S/S 316 + Asb Gask, Rubber Pist Gromm
Demineralizer to Bleed Holdup	MU-V112F	Aux	119	304/M	8.728 + 5	Ethylene Propylene, Asbestos, TFE, SAE F-3 Felt
Purification MU Inlet to MU-F2B	MU-V97F	Aux	119	303/L	1.144 + 6	JC-187-I-CR Asbestos Packing, S/S 316 + Asb Gask, Rubb.+ Pist. Gromm
MU Filter Bypass to MU-T1	MU-V100	Aux	119	303/L	1.144 + 6	JC-187-I-CR Asbestos Packing, S/S 316 + Asb. Gask, Rubb.+Pist.
Seal Return Isolation Valve	MU-V357	Aux	119	302/L	7.528 + 5	JC-187-I-CR Asbestos Packing, S/S 316 + Asb. Gask, Rubb.+ Pist
MU-T1 Isolation Valve	MU-V64	Aux	95	302/L	5.299 + 5	Braided Asbestos Packing + Oper.
MUP Suction Isolation Valve (4)	MU-V68,69,62,63	Aux	95	304/J,L	3.687 + 6	Asbestos Packing
MUP Discharge Stop Check Valve (3)	MU-V2,6,11	Aux	95	304/L,K,J	3.687 + 6	Asbestos Packing
MUP Discharge Isolation Valve (4)	MU-V3,4,8,9	Aux	95	304/L,K,J	3.687 + 6	Braided Asbestos Packing JG-187
HPI Control Valve (4)	MU-V23F,24F,25F,26F	Aux	95	304/K,L	3.687 + 6	316 + Asbestos Gasket, JC1871CR Packing
RC Pump Seal Control Valve	MU-V16F	Aux	95	304/K	3.687 + 6	TFE Packing, Nitrile Diaph.
MUP Bypass Control Valve	MU-V17	Aux	95	304/K	3.687 + 6	Braided Asbestos Packing JC-187, SS316 + Asbestos Gask
RC Pump Seal Isolation Valve	MU-V18F	Aux	119	305/N	1.373 + 6	316 + Asbestos Gasket, JC187CR Packing
Seal Injection Bypass Valve	MU-V452	Aux	95	305/N	1.144 + 5	Braided Asbestos Packing JC-187, SS316 + Asbestos
RB Spray Pump (2)	BS-P1A,B	Aux	75	305/N,O	1.373 + 6	Garlock #7022 Gasket, A-108 + Rubber Seal
RB Spray Pump Isolation Valve (2)	BS-V16F,17F	Aux	75	305/P,O	1.373 + 6	JC-187-I-CR Asbestos Pack, S/S316 +ASB Gask, Rubb. Gromm
RB Spray Header Isolation Valve (2)	BS-V3F,4F	Aux	95	305/N	1.144 + 5	Asbestos Packing JC-187-I-CR

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TABLE 5-1 (Cont.)

Equipment Name	Identification Number	Location: Building	Location: Elevation	Location: Column	Calculated Integrated Dose (6 mon) RADS	Radiation Sensitive Material
BS-T2 Isolation Valve (2)	BS-V36F,37F	Aux	75	305/P	1.373 + 6	JC-187-I-CR Asbestos Pack, S/S 316 + Asb Gask, Rubber Gromm
Aux. Bldg. Air Handlitg Cooling Coil (2)	AH-HE30A,B	Aux	95	306/T	(1)	Red Rubber or Neoprene Gaskets 1/16" Tack
RB Spray Pump Cooling Coil (2)	BS-P1A,B	Aux	75	305/N,O	1.373 + 6	Red Rubber or Neoprene Gaskets 1/16" Tack
DH Closed Cycle Heat Exchanger (2)	DC-HE1A,B	Aux	95	304/T	(1)	1/8" Tack Neoprene Gaskets
DH CC Cooling Pump (2)	DC-P1A,B	Aux	95	306/S,T	(1)	
DH Heat Exchanger (2)	DH-HE1A,B	Aux	75	304,306/P	1.373 + 6	
DH Pump (2)	DH-P1A,B	Aux	75	305/Q	1.373 + 6	
MUP Cooling Coil Pump (2)	MU-P1A,C	Aux	95	303/J,K	1.475 + 5	
DH Seawater Cooling Coil Pump (2)	RW-P3A,B	Aux	95	307/P,R	5.299 + 4	Garlock 7021 Gaskets, Carbon Bushings (Cutless Rubber) O-Ring BunaN, Crane Superseal #1
Demin Water to DHCC Surge (2)	DGV-10,12	Aux	95	306/S	(1)	TFE Packing, Nitrile Diaphram
DHCC Tank Isolation Valve Pump Isolation Valve (4)	DCV-1,2,3,4	Aux	95	306/S,T	(1)	Buna N (Hycar) Seat and Seals
DH Air Unit Inlet Isolation Valve (2)	DC-V31,32	Aux	95	306/T	(1)	JC-187-I Asbestos Packing
DH Air Unit Discharge Isolation Valve (2)	DC-V37,38	Aux	95	306/T	(1)	Asbestos
DH CC Heat Exchanger (4)	DC-V13,14,15,16	Aux	95	305/T	(1)	Buna N (Hycar) Seal and Seals
MUP Pump Motor (2)	DC-V21,22	Aux	95	304/J,K	1.373 +	Asbestos Packing
DH Shutoff Valve (2)	DC-V43,44	Aux	95	307/P	5.299 + 4	Braided Asbestos Packing - JC-187
RB Spray Pump Inlet Isolation Valve (2)	DC-V27,28	Aux	75	305/N,O	1.373 + 6	JC-187-I Asbestos Packing
RB Spray Pump Stuffing Box Bearing (2)	DC-V115,117	Aux	75	305/N,O	1.373 + 6	Braided Asbestos Packing JC-187, S/S 316 + Asbestos
RB Spray Pump Discharge Isolation Valve	DC-V33	Aux	75	305/N	1.373 + 6	Asbestos
DCH Pump Inlet Isolation Valve (2)	DC-V29,30	Aux	75	305/Q	1.373 + 6	JC-187-I Asbestos Packing
DCH Heat Exchanger Inlet Isolation Valve (2)	DC DC-V5,6	Aux	75	304/R	1.373 + 6	Buna N (Hycar) Seat and Seals
DCH Cooler Control Valve (2)	DC-V17,18	Aux	75	304,306/R	1.373 + 6	Carbon Filled TFE Bushings, TFE Packing, Nitrile Diaph.
DCJ Jeat Excjamger Discharge Isolatioc Valve (2)	DC-V7,8	Aux	75	304,305/R	1.373 + 6	Buna N (Hycar) Seat and Seals
DCH Cooler Control Valve (2)	DC-V177,178	Aux	75	304,305/R	1.373 + 6	Carbon Filled TFE Bushings, TFE Packing, Nitrile Diaph.
Low Pressure Injection Isolation Valve (2)	DH-V5F,6F	Aux	95	305,306/N	1.144 + 5	316 + Asbestos Gasket, JC1871CR Packing
DH Discharge Crosstie (2)	DH-V7,8	Aux	95	305/O	2.709 + 5	JC-187-I Asbestos Packing
DH Pump Discharge	DH-V12	Aux	95	306/P	5.299 + 4	JC-187-I-CR Asbestos Pack, S/S 316 + Asb Gask, Rubber Gromm Motor

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TABLE 5-1 (Cont.)

Equipment Name	Identification Number	Location: Building	Location: Elevation	Location: Column	Calculated Integrated Dose (6 mon) RADS	Radiation Sensitive Material
DH Heat Exchanger	DH-V144	Aux	75	306/P	1.373 + 6	Asbestos Packing
DH Pump Discharge Isolation Valve	DH-V106	Aux	95	306/Q	5.299 + 4	Braided Asbestos Packing JC-187 S/S 316 + Asb. Gas + Motor
DH Pump Suction to DH-T1 Isolation (2)	DH-V34F,35F	Aux	75	305/P	1.373 + 6	JC-187-1-CR Asbestos Pack, S/S 316 + Asb Gas, Rubber Gromm + Motor
DH Pump Isolation Valve (2)	DH-V39F,40F	Aux	95	305/O	2.709 + 5	Asbestos Packing
DH Containment Isolation	DH-V41F	Aux	95	305/N	1.144 + 5	JC 187-1-CR Asbestos Pack, S/S 316 + Asb Gask, Rubber Gromm + Motor
RB Sump to DH Pump Isolation (2)	DH-V42F,43F	Aux	75	305/M,N	1.373 + 6	316 + Asb Gask, Asb Packing JC-187-I-CR. Once opened must remain operator recirc.
DH Exchanger Discharge Isolation (2)	DH-V110,111	Aux.	75	304/P	1.373 + 6	JC-187-I-CR Asbestos Pack, S/S 316 + Asb Gask, Rubber Gromm + Motor
DH Pressurizer Spray Isolation	DH-V91	Aux	119	305/N	1.373 + 6	Braided Asbestos Packing JC-187, S/S 316 + Asb. Gask + Motor
RB Vent Header Isolation Valve	WD-V405	Aux	119	305/N	1.373 + 6	Braided Asbestos Packing JC-187, S/S 316 + Asb. Gask + Motor
Waste Gas Decay Tank to Recycle Valves (3)	WD-V393,394,395	Aux	95	303/P	3.703 + 6	Vinyl Coated Mylar Label Grade M EPT Nordel Diaph., Neoprene Gask Elastomer Diaph, Dash #114 EPR O Ring
WG DT Outlet Valve to Vent System (3)	WD-V436,437,438	Aux	95	303/P	3.703 + 6	Vinyl Coated Mylar Label Grade M EPT Nordel Diaph., Neoprene Gask Elastomer Diaph, #112 O-Ring
WG Surge Tank Bypass Isolation	WD-V382	Aux	95	303/P	3.703 + 6	Grade M EPT Nordel Diaph, O-Ring - EPR Dash #112&222, Plastic Shim W..her
WG Surge Tank Inlet Isolation	WD-V380	Aux	95	303/P	3.703 + 6	Grade M EPT Nordel Diaph, O-Ring - EPR Dash #112&222, Plastic Shim Washer
WG Surge Tank Outlet Isolation	WD-V381	Aux	95	303/P	3.703 + 6	Grade M EPT Nordel Diaph, O-Ring - EPR Dash #112&222, Plastic Shim Washer
WG Compressor Control Valve	WD-V465F	Aux	95	303/P	3.703 + 6	TFE Packing
WGC Suction Isolation Valve (2)	WD-V384,385	Aux	95	303/P	3.703 + 6	Grade M EPT Nordel Diaph., Neoprene Gask, Vinyl Coated Mylar Label, Elastomer Diaph, O-Ring Dash #112 EPR
Borated Water Tank Level Trans	DH-7,37-LT	Yard	119	307/P	(1)	
Makeup Tank Level Trans	MU-14-LT	Aux	119	302/L	3.7 + 7	
RC Letdown Monitor	RM-L-01	Aux			3.7 + 6	
RB Air Sample	RM-A-06	Aux	119	305/N	1.4 + 6	
NaOH Tank Level Trans.	BS-5-LT	Yard	119	307/P	(1)	
Gas Decay Tank Level Trans.	WD-16,17,18-LT	Aux	95	302/P	3.7 + 6	

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TABLE 5-1 (Cont.)

Equipment Name	Identification Number	Location: Building	Location: Elevation	Location: Column	Calculated Integrated Dose (6 mon) RADS	Radiation Sensitive Material
Pressurizer Steam Space Sample Isolation	CA-V2F	Aux	95	305/H	4.441 + 3	Evalpak 187-1X Packing + Pist.
Steam Generator Sample Isolation (2)	CA-V6F,7F	Aux	95	305/H	4.441 + 3	Evalpak 187-1X Packing + Pist.
Pressurizer Sample Inlet Isolation	CA-V331	Aux	95	304/H	5.299 + 3	Braided Asbestos Packing
Pressurizer Sample Discharge Isolation	CA-V332	Aux	95	304/H	5.299 + 3	Braided asbestos Packing
Steam Generator Sample Cooler Metering (2)	CA-V312,314	Aux	95	304/H	5.299 + 3	Viton O-Ring, TFE Packing
Sample Metering Valve	CA-V313	Aux	95	304/H	5.299 + 3	Viton O-Ring, TFE Packing
H_2 Purge Containment Isolation (4)	LRV-35,38,49,51	Int	119	308/H1,I_1,J_2	2.689 + 2	Asbestos Packing
H_2 Purge Crossover (2)	LRV-47,57	Int	119	309/I1,J2	2.689 + 2	Asbestos Packing
Spent Fuel Cooler (2)	SF-HE1A,B	Aux	143	302/J	(1)	1/8" Thck. Armco-1 JAF Gasket
SF Pump Air Handling Unit (2)	AH-HE29A,B	Aux	119	304/J	(1)	
Control Complex Water Chiller (2)	CH-HE1A,B	Con Complex			(1)	3/4" Thck. Closed Cell Plastic Insul.
Sample Cooler (3)	CA-HE1,2A,2B	Aux	95	305/G	5.299 + 3	
Seal Return Cooler (2)	MU-HE2A,B	Aux	119	302/L	7.528 + 5	See Sheet 1 #5
RC Evaporator	WDEV-1	Aux	95	301/M	1.732 + 5	
Waste Evaporator	WDEV-2	Aux	95	301/L	1.732 + 5	
Waste Gas Compressor (2)	WD-P1A,B	Aux	95	302/O,P	3.703 + 6	Asb Polyacrylate, Silicon, Sponge, Synth Rubber; Neop., Invar, TFE, EPR, Plastic
Motor Driven Emergency Feed Pump	EF-P1	Aux	95	309/H	(1)	
Turbine Driven EFP	EF-P2	Aux	95	308/H	(1)	
Emergency Nuclear Service Closed Cycle Cooling Pump (2)	SWP 1A,B	Aux	95	307/P/Q	5.299 + 4	
Normal NS Seawater Pump	RW-P1	Aux	95	307/P	5.299 + 4	Garlock 7021 Gaskets, Buna N-Rings Carbon Bushings, Crane Super Seal #1
Emergency NS Seawater Pump (2)	RW-P2A,B	Aux	95	307/Q	5.299 + 4	Garlock 7021 Gaskets, Buna N-Rings Carbon Bushings, Carne Super Seal #1
Closed Cycle Pump Isolation	SW-V1	Aux	95	307/S	5.299 + 4	EPDM Nordel Seat and Seals (Sulphur Free)
Control Rod Drive Inlet Isolation	SW-V201	Aux	119	305/O	1.144 + 6	Braided Asbestos JC-187-I
CRD Discharge Isolation (2)	SW-V203,381	Aux	119	305/O	1.144 + 6	Braided Asbestos JC-187-I FOR-203/FOR-381 Teflan Seal & Seal
CRD Bypass Isolation	SW-V283	Aux	119	305/O	1.144 + 6	Braided Asbestos JC-187-I
NS Water Inlet Isolation	SW-V109	Aux	119	306/N	1.373 + 6	Braided Asbestos Packing + Pist.
Letdown Cooler Inlet Isolation	SW-V47,48	Aux	95	305,306/N	1.144 + 5	EPDM Nordel Seat & Seals + Pist.
Letdown Cooler Discharge Isolation	SW-V49,50	Aux	95	306/N	1.144 + 5	EPDM Nordel Seat & Seals + Pist.
RB Fan Outlet Isolation (3)	SW-V,41,43,45	Aux	95	306/N	1.144 + 5	EPDM Nordel Seat & Seals + Pist.
RB Fan Inlet Isolation (3)	SW-V35,37,39	Aux	95	306/N	1.144 + 5	EPDM Nordel Seat & Seals + Pist.
RB Fan Isolation (2)	SW-V353,354	Aux	95	307/O	5.299 + 4	Buna N (Hycar) Seat & Seals + Pist
RC Pump Temperature Control Valve (4)	SW-V87,88,89,90	Aux	119	305,306/O	1.144 + 6	TFE Packing, Nitrile Diaphram
RC Pump Discharge Isolation (4)	SWV-83,84,85,86	Aux	119	306/N	1.373 + 6	EPDM Nordel Seat & Seals (Sulfur Free)

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TABLE 5-1 (Cont.)

Equipment Name	Identification Number	Location: Building	Location: Elevation	Location: Column	Calculated Integrated Dose (6 mon) RADS	Radiation Sensitive Material
RB Pump Inlet Isolation (4)	SWV-78,80,81,82	Aux	119	305,306/N	1.144 + 6	EPDM Nordel Seat & Seals (Sulfur Free)
NS Isolation to RB	SWV-11D	Aux	119	305/N	1.373 + 6	Braided Asbestos Packing + Pist.
Evaporator Package Isolation	SWV-12	Aux	119	304/P	1.732 + 5	EPDM Nordel Seat & Seals (Sulfur Free) Pist. Oper.
Misc. Evap. Outlet Isolation	SWV-57	Aux	95	302/L	1.715 + 3	EPDM Nordel Seat & Seals (Sulfur Free) + Pist. Oper.
FC Evap. Outlet Isolation	SWV-58	Aux	95	302/L	1.715 + 3	EPDM Nordel Seat & Seals (Sulfur Free) + Pist. Oper.
Nuclear Service Heat Exchanger (4)	SWHE-1A,B,C,D	Aux	95	305/S	(1)	
DH Seawater Pump Discharge (2)	RWV-17,18	Aux	95	306/R,P	5.299 + 4	Buna N (Hycar) Seat & Seals
Emergency NS SWP Discharge Isolation	RWV-21	Aux	95	306/Q	5.299 + 4	Buna N (Hycar) Seat & Seals
Normal NS SWP Discharge Iso. (2)	RWV-22,24	Aux	95	306/Q,P	5.299 + 4	Buna N (Hycar) Seat & Seals
NS Heat Exchanger Inlet Isolation (4)	RWV-5,6,7,8	Aux	95	305/R,S	(1)	Buna N (Hycar) Seat & Seals
NS Heat Exch Discharge Isolation (4)	RWV-13,14,15,16	Aux	95	305/S	(1)	Buna N (Hycar) Seat & Seals
NS Heat Exch Isolation (2)	RWV-32,33	Aux	95	307/S,R	5.299 + 4	Buna N (Hycar) Seat & Seals
DH CC Heat Exch Isolation (2)	RWV-40,41	Aux	95	307/S,R	5.299 + 4	Buna N (Hycar) Seat & Seals
Hotwell Isolation to EFP (2)	EFV-1,2	Aux	95	308,309/G	(1)	Asbestos Packing + Motor
EFP to Steam Gen Isolation (2)	EFV-7,8	Aux	95	309,308/G	(1)	JM 398 Packing + Motor
Condensate Storage Tank Isolation	EFV-4	Aux	95	308/G	(1)	Asbestos Packing + Motor
EFP Discharge Iso Crosstie (2)	EFV-12,13	Aux	95	308/G,H_1	(1)	Asbestos Packing
EFP Discharge Iso to Steam Gen (2)	EFV-11,14	Aux	95	308/G-309/H_1	(1)	Asbestos Packing + Motor
EF to Steam Generator Isolation (2)	EFV-32,33	Aux	95	309/H	(1)	Braided Asbestos Packing + Motor Oper. JC-187
Radwaste Disposal Control Board		Aux	95	304/N	5.659 + 4	
RB Purge Exhaust Valve	AHV-1A	Aux	143	305/N	(1)	
RB Purge Supply Valve	AHV-1D	Int	119	309/J2	3.359 + 2	
DH Pit Air Handling Unit (2)	AHF-15A,B	Aux	95	306/T	5.299 + 4	Neoprene & Fiberglass Insul.
Spent Fuel Coolant Pump Cooling Air Handling Unit (2)	AHF-8A,B	Aux	119	304/J	(1)	
Control Complex Emergency Supply Fan (2)	AHF-18A,B	CC	164	301,304/H	(1)	
Control Complex Return Fan (2)	AHF-19A,B	CC	164	303/G	(1)	
Control Complex Heating Coil (2)	AHHE-4A,B	CC	164	303/G	(1)	Glass-Teflon Insul.
Control Complex Cooling Coil (2)	AHHE-5A,B	CC	164	303/G	(1)	Red Rubber or Neoprene Gasket 1/16" Thck
Emergency Diesel Gen Supply Fan (4)	AHF-22A,B,C,D	DG	119	301/Q	(1)	Neoprene
Emerg DG Filter (2)	AHFL-5A,B	DG	119	301/Q	(1)	Glass Mat Media, Neoprene Base Adhesive
ES Motor Control Center 3A1 (5)	MTMC-3	Aux	95	301/K	5.299 + 5	
ES MCC 3A2	MTMC-4	Aux	119	301/L	1.[illegible]2 + 2	
ES MCC 3B1	MTMC-5	Aux	119	301/M	1.852 + 4	
ES MCC 3B2	MTMC-6	Aux	95	303/N	5.659 + 4	
ES MCC 3AB	MTMC-7	Aux	119	304/O	1.732 + 5	
Power & Control Cable	- - -	- - -	- - -	- - -	(2)	Silicon Rubber, Teflon, Asbestos, Hypalon, Kerite (Brand), Crosslink Polyethylene
Instrument & Control Cable	- - -	- - -	- - -	- - -	(2)	
Control & Thermocouple Cable	- - -	- - -	- - -	- - -	(2)	
Wall Mounted Contactors - MTMC-4	- - -	Aux	119	301/L	1.732 + 2	
Wall Mounted Contactor - MTMC-5	- - -	Aux	119	301/M	1.732 + 2	
Wall Mounted Contactor - MTMC-7	- - -	Aux	119	304/O	1.732 + 5	

FOOTNOTE 1: Negligible <100 R

FOOTNOTE 2: Dose not calculated. Cable has been qualified to 10^8 rads.

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TABLE 5-2

Radiation Sensitive Component Material	Reference Material	Exposure (Rads)
Ethyl - Ppro seals	EPM ethylene propylene monomer	5.0×10^6 (3)
Epoxy impregnated cellulose fiber filter	Epoxy cellulose pulp filled	10^8 (1)
Asbestos	Asbestos fabric filled phenolic	10^{10} (1)
Evalpak 187-1X packing	- - -	
JC-187-I-CR Asbestos packing	Asbestos fabric filled phenolic	10^{10} (1)
Rubber gromm	Natural rubber	4.5×10^7 (2)
Ethylene propylene	EPM ethylene propylene monomer	5.0×10^6 (3)
TFE	Teflon	$< 1.0 \times 10^6$ (1)
SAE P-3 Felt	Felt	5.4×10^6 (4)
Braided asbestos packing JG-187	Asbestos fabricfilled phenolic	10^{10} (1)
Braided asbestos packing JC-187	Asbestos fabric filled phenolic	10^{10} (1)
Garlock #7022 gasket	- - -	
A-108	- - -	
Rubber seal	Natural rubber	4.5×10^7 (2)
Neoprene gaskets	Neoprene	5.0×10^6 (3)
Garlock 7021 gaskets	- - -	
Crane super seal #1	- - -	
Buna N o-ring	HYCAR OR-15 NBR, HYCAR PA-21	1.5×10^7 (2), 1.6×10^7 (2)
Carbon bushings	Carbon	negligible damage
Buna N (HYCAR) Seat & Seals	HYCAR OR-15 NBR, HYCAR PA-21	1.5×10^7 (2), 1.6×10^7 (2)
Vinyl coated mylar label	Mylar A	10^8 (1)
Grade M EPT Nordel diaphragm	EPDM ethylene propylene diene monomer	5.0×10^6 (3)

1755 050

TABLE 5-2 (Cont.)

Radiation Sensitive Component Material	Reference Material	Exposure (Rads)
Elastomer diaphragm	Butyl rubber GR-I 50, Styrene - butadiene rubber (SBR) (GRS-50)	2.1×10^7 (2), 1.2×10^7 (2)
Neoprene	Neoprene	5.0×10^6 (3)
JM-398 packing	- - -	
Viton o-ring	- - -	
Red rubber	Natural rubber	4.5×10^7 (2)
Nitrile diaphragm	Nitrile rubber HYCAR OR-15 NBR	1.5×10^7 (2)
EPR o-ring	EPM ethylene propylene monomer	5.0×10^6 (3)
Plastic shim washer	- - -	
Armco-I JAF gasket	- - -	
Plastic insulation	- - -	
ASB polyacrylate silicon	- - -	
Sponge	- - -	
Synthetic rubber	Butyl rubber GR-I 50, Styrene-butadiene rubber (SBR)(GRS-50)	2.1×10^7 (2), 1.2×10^7 (2)
INVAR	- - -	
Carbon bushings	Carbon	negligible damage
EPDM Nordel (sulphur free)	EPDM	5.0×10^6 (3)
Teflan seat and seal	Teflon	$< 1.0 \times 10^6$ (1)
Fiberglass insulation	Fiberglass	5.0×10^9 (4)
Glass - teflon insulation	Teflon	$< 1.0 \times 10^6$ (1)
Glass mat media	Glass	5.0×10^9 (4)

1755 051

TABLE 5-2 (Cont.)

Radiation Sensitive Component Material	Reference Material	Exposure (Rads)
Neoprenebase adhesive	Neoprene	5.0×10^6 (3)
Silicon rubber	Silicone rubber Silastic 7-170	6.8×10^6 (2)
Teflon	Teflon	$< 1.0 \times 10^6$ (1)
Hypalon	HYPALON	5.0×10^6 (3)
Kerite (Brand)	- - -	
Crosslink polyethylene	Black XLPE crosslinked polyethylene	5.0×10^6 (3)

(1) Exposure which the given material received and upon subsequent testing retained 80% or more of its pre-irradiation mechanical properties of elongation and tensile strength. Engineering Compendium on Radiation Shielding, Vol. II, Springer-Verlag, Berlin/Heidelberg, pp. 311-314, (1975).

(2) The 25% damage dose for the mechanical properties of tensile strength or elongation. Engineering Compendium on Radiation Shielding, Vol. II, Springer-Verlag, Berlin/Heidelberg, p. 316 (1975).

(3) Exposure which the material received and upon subsequent testing retained 75% or more of its pre-irradiation mechanical properties of tensile strength and elongation. "Insulations and Jackets for Control and Power Cable in Thermal Reactor Nuclear Generating Stations," R. B. Blodgett and R. G. Fisher, Okonite Co., Passaic, NJ, 07055, Institute of Electric and Electronic Engineers - June (1969).

(4) Dose at which a physical or mechanical property changes by 25%. Remote Handling of Mobile Nuclear Systems, D. C. Layman and G. Thornton; U.S. Atomic Energy Commission/Div. of Tech. Information, Oak Ridge, Tennessee, pp. 506-515, Jan. (1966).

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ATTACHMENT IV

VENTING DESIGN CRITERIA

A. Summary Description

Various postulated small breaks can lead to accident scenarios in which steam and/or non-condensible gases accumulate in the reactor vessel head, the upper portion of the hot legs and in the pressurizer. Following repressurization of the RCS by HPI, which will tend to collapse steam bubbles, remotely controlled vents on the upper hot legs and pressurizer can be used to vent non-condensible gases to aid in refilling the RCS and promoting natural circulation flow for core cooling.

B. Piping and Valving Considerations

1. Vents shall be provided at the following reactor coolant system high points:

   a. Top of each hot leg (2 vents total, one per hot leg)

   b. Top of pressurizer (1 vent)

2. Vent piping and valving shall be designed and sized such that the failure to close off any one (1) of the vent points listed above could not cause a loss of coolant at a rate in excess of the normal makeup system capability at full design RCS pressure.

3. The effluent flow from all vent points shall be routed directly to the containment atmosphere. The region into which the discharge is diverted shall enhance mixing and dilution so as to minimize the potential for local regions from reaching flammable concentrations of gases. Discharge needs to be routed and directed so that liquid effluent can not discharge on or fall on electrical equipment or mechanical operating equipment.

4. The piping and valving for the venting system shall be routed, oriented and protected so that damage from pipe whip, jet impingement and missiles will not occur.

5. Pipe routing, orientation and elevation shall assure that all remotely operable valves are (a) located well above the maximum level of water in the containment expected for the worst case DBA and (b) protected from the containment spray and relief discharges. Each vent shall be designed to remain functional after all design basis events except large LOCA's, evacuation of the Main Control Room and loss of all AC power.

6. Vent piping and valving shall be designed for 2500 psig and 670°F and any gaskets or seals shall be compatible with all anticipated effluent fluids. This includes water, saturated steam, steam water mixture, superheated steam, fission product gases, helium, nitrogen and hydrogen. Provisions for venting hydrogen shall include the necessity for spark-free valves.

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B. Piping and Valving Considerations (Cont'd)

7. Each venting point shall be individually operable independent of any other point vent. Operator guidelines shall be provided to reduce the possibility of venting from more than one vent point at a time and to minimize the possibility of excessive RC system depressurization. Ref. Section B2.

8. All piping and valving shall be connected to the RCS and supported in such a manner so that any stress due to weight, thermal transients, internal piping conditions and external environment will be within the maximum allowable stresses at the existing vent nozzles. Piping shall be designed to prevent the formation of traps and minimize the possibility of water and/or steam hammer.

9. Existing nozzles in the RC System shall be used for the venting system. No new nozzles shall be added exclusively for venting.

10. All remote operable "two position" (on-off) valves shall be of the fail closed type, with power required to open and power required to maintain open. Valves shall have proven fail "closed" action on loss of power. No packings are permissible.

C. Control and Instrumentation Considerations

1. All valves for any one vent nozzle shall be powered from a supply separate from that which powers the valves for any other vent nozzle so that any single power supply failure cannot cause a failure to vent at more than one vent nozzle. Vital power supplies shall be provided for all of the vent isolation valves. Complete vent shutoff of any one vent nozzle shall be assured on loss of all power to its venting system.

2. Control of vent valves shall be remote manual and operable from the control room only. There is no requirement for operation from any auxiliary location. Direct indication of valve positions shall be provided in the control room.

3. Control of valves for any one vent point shall be independent of the control for valves for any other vent point.

4. Both vent valves at a vent point shall be powered by the same power source, but controlled by two (2) independent switches. An alarm shall indicate both valves are energized.

5. The vent valve operating switches shall be such that the vent valve will not open when power is applied to the switch. It must take an independent action to operate the switch.

D. Operating Guidelines and Modes

1. The venting system may be used to vent the RCS during RCS filling operations if no venting system functional requirements are violated.

D. Operating Guidelines and Modes (Cont'd)

2. Priorities of system design shall be as follows:

a. RC system integrity

b. Capability to vent to containment atmosphere

3. Special precautions shall be taken to prevent any unauthorized venting. Such precautions shall be in addition to normal administrative controls.

E. Testability

1. Provisions shall be made for testing all portions of the venting system at any time during startup of the plant. Testing shall consist of the following:

a. Confirmation of free flow passage from each vent point. This will include exercising of all valves and checking the flow.

Flow indication need only indicate that flow is present, and quantification is not required. Such testing shall normally be done during initial fill.

b. Confirmation of vent shutoff capability shall be established by initially filling the vent lines during pre-service hydro and establishing effluent flow into the containment with the vent valves open. The vent valves in each vent system will then be closed until effluent flow is observed to cease. If it does not cease after a few minutes and continues at some observable rate, this should be quantified by timing flow into a vessel of known volume. A leakage rate in excess of 10cc per hour from a vent system at pre-service hydro pressures shall be considered unacceptable. No external leakage is permissible.

c. Flow indications which have been previously tested shall be monitored to assure that gross inadvertent venting is not occurring during normal reactor operation. Ref. Section E.1.a. above.

F. Thermal Stress and Insulation Considerations

1. When practical, effluent piping from the high point vent nozzles to the vent valves shall be thermally insulated to:

a. Provide piping protection

b. Reduce heat losses

c. Minimize piping stresses

d. Provide personnel protection

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F. Thermal Stress and Insulation Considerations (Cont'd)

2. Provisions shall be made to minimize thermal stresses due to venting, so that intermittent venting can be tolerated.

G. Environmental Qualifications

1. The high point vent system shall be designed to maintain its integrity and function for the lifetime of the plant, assuming periodic replacement of consummables.

2. All components located inside primary containment shall be qualified to the maximum LOCA or main steam line break (MSLB) environmental conditions and to the process conditions stated in Section B.6.

H. Safety Classification

1. All fluid portions of the venting systems from the vent nozzles, up to and including all vent valves, are part of the reactor coolant pressure boundary and as such are seismically qualified and classified per Reg. Guide 1.26 as safety class A or B, depending upon piping size.

2. The valves shall be classed as active, subject to the requirements of Reg. Guide 1.48.

3. As a minimum, all electrical portions of the system for hot leg "two position" (on-off) valves shall be Class IE.

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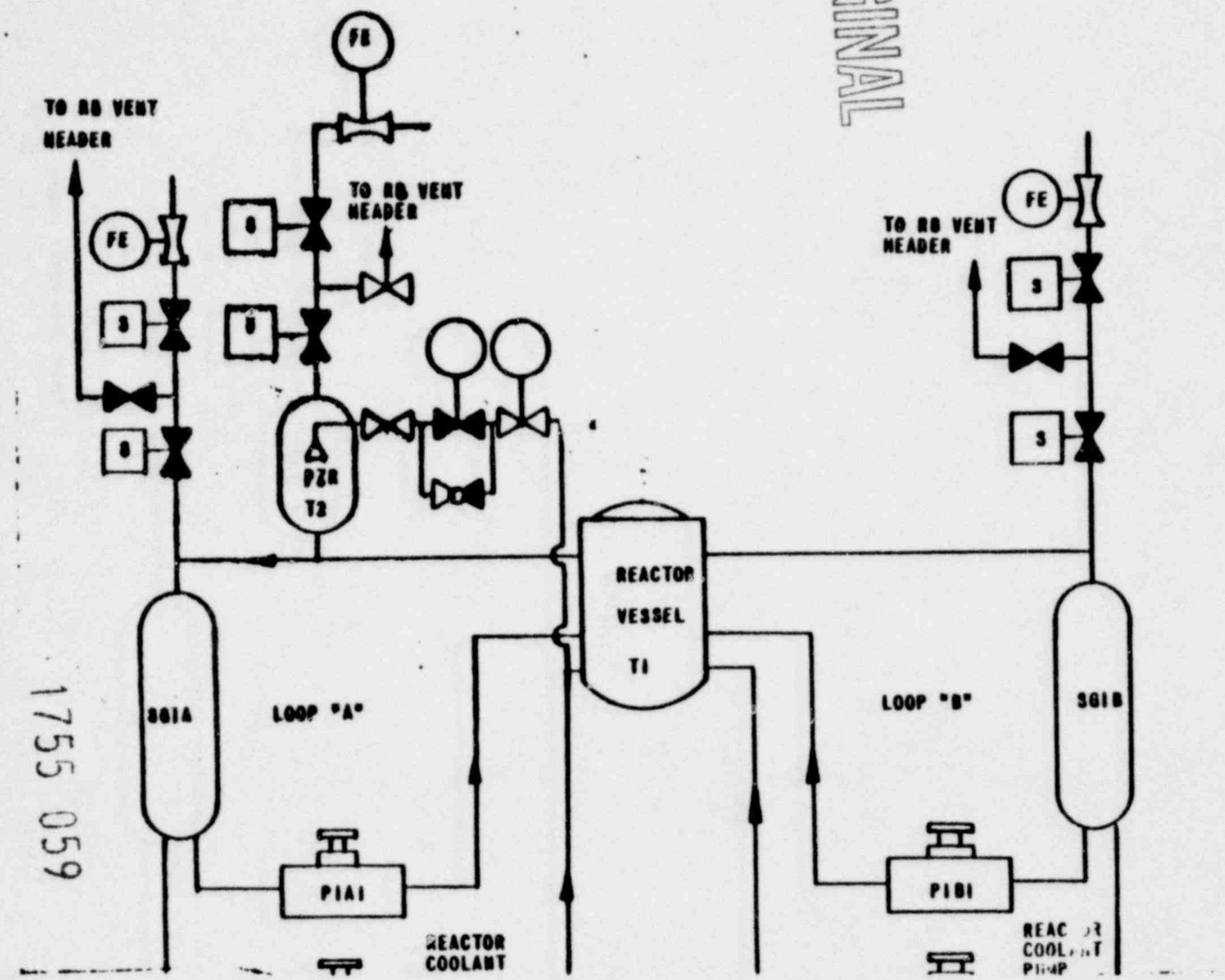
VENTING SYSTEM

SCHEMATIC



NOTES:

1. ALL "ON OFF" SOLENOID VALVES FAIL CLOSED ON LOSS OF POWER (NC VALVES)
2. FLOW ELEMENTS ARE "FLOW DETECTORS" ONLY. PRECISE FLOW MEASURING IS NOT REQUIRED.
3. ALL VENT PIPING IS 1" SS-SCH 160 FROM RC SYSTEM TO LAST "ON-OFF" SOLENOID VALVES.
4. HIGH POINT VENT ASSY'S UTILIZE EXISTING RC SYSTEM NOZZLES.
5. B&W REQUIRES HOT LEG & PRESSURIZER VENTS ONLY. REACTOR VESSEL VENTS ARE OPTIONAL.
6. OPTIONAL VENT LINES ARE SHOWN AS (---).
7. ALL VALVES ARE SHOWN IN DE-ENERGIZED POSITIONS.

HIGH POINT VENTS
PRELIM. CONCEPTUAL SCHEMATIC

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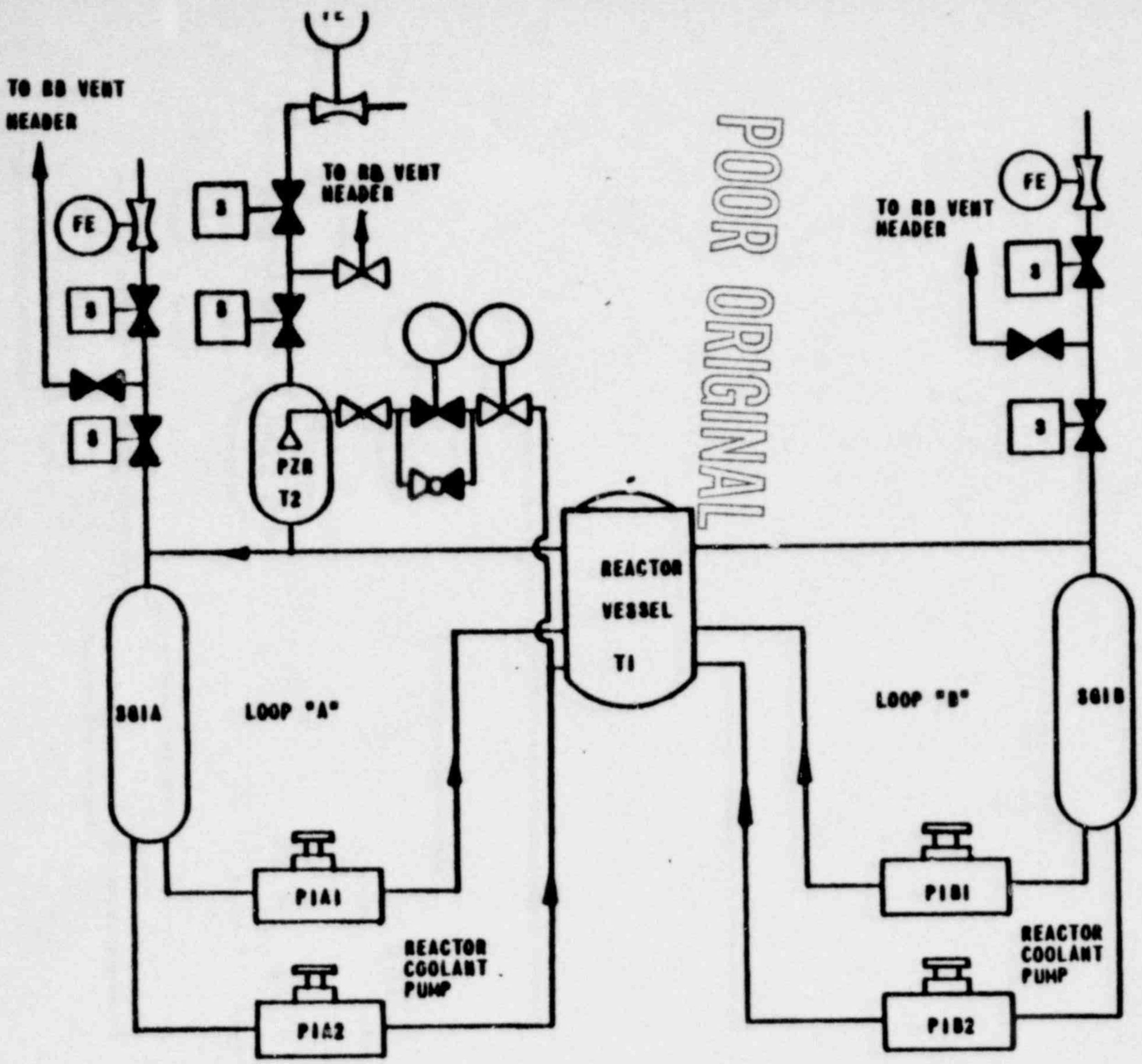
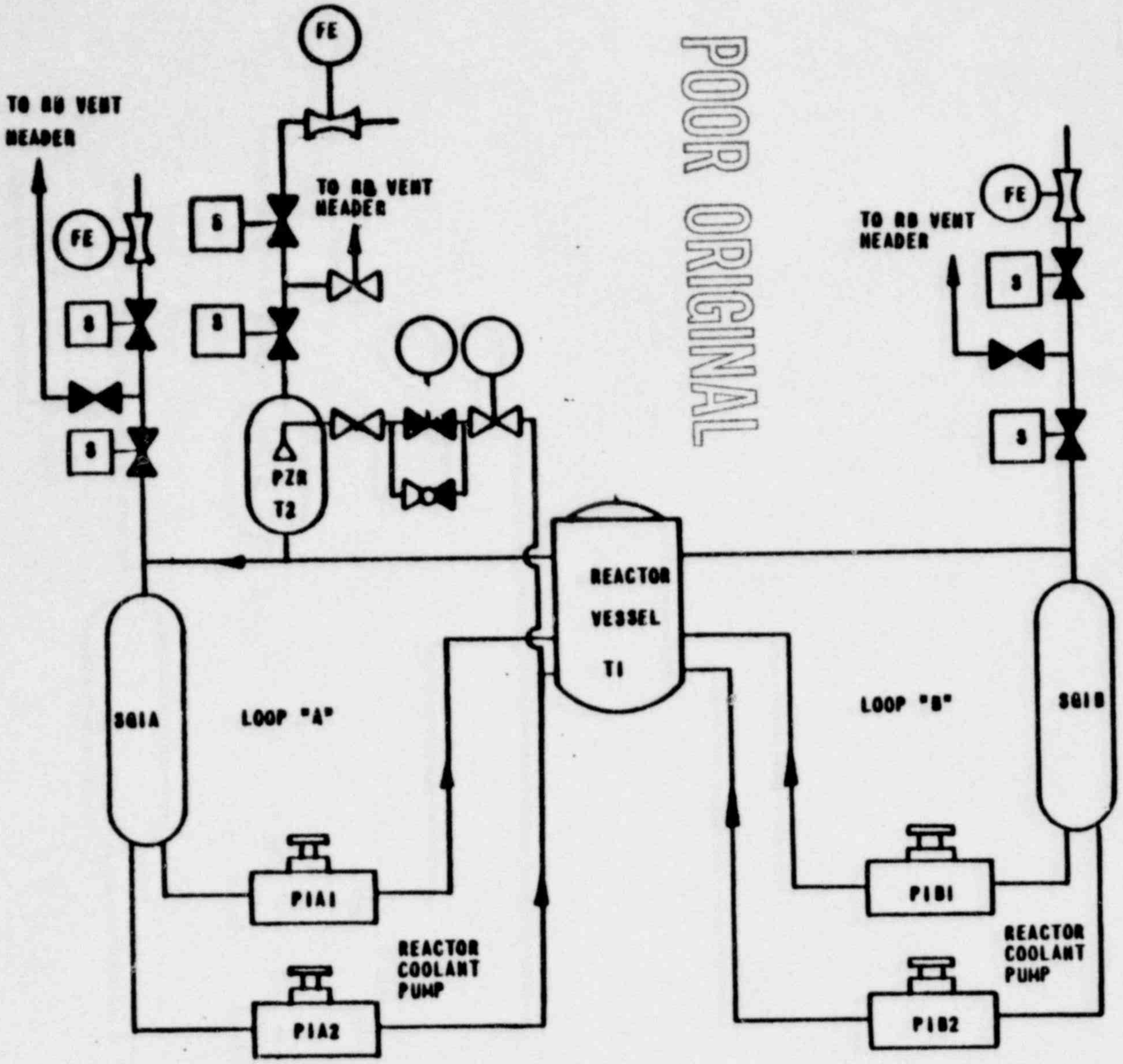


1. ALL "ON OFF" SOLENOID VALVES FAIL CLOSED ON LOSS OF POWER (VALVES)
2. FLOW ELEMENTS ARE "FLOW DETECTORS" ONLY. PRECISE FLOW MEASURING IS NOT REQUIRED.
3. ALL VENT PIPING IS 1" SS-SCH 160 FROM RC SYSTEM TO LAST "ON-OFF" SOLENOID VALVES.
4. HIGH POINT VENT ASSY'S UTILIZE EXISTING RC SYSTEM NOZZLES.
5. B&W REQUIRES HOT LEG & PRESSURIZER VENTS ONLY. REACTOR VESSEL VENTS ARE OPTIONAL.
6. OPTIONAL VENT LINES ARE SHOWN AS (---).
7. ALL VALVES ARE SHOWN IN DE-ENERGIZED POSITIONS.

HIGH POINT VENTS
PRELIM. CONCEPTUAL SCHEMATIC



NOTES:

1. ALL "ON OFF" SOLENOID VALVES FAIL CLOSED ON LOSS OF POWER (NC VALVES)

2. FLOW ELEMENTS ARE "FLOW DETECTORS" ONLY. PRECISE FLOW MEASURING IS NOT REQUIRED.

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4. HIGH POINT VENT ASSY'S UTILIZE EXISTING RC SYSTEM NOZZLES.

5. B&W REQUIRES HOT LEG & PRESSURIZER VENTS ONLY. REACTOR VESSEL VENTS ARE OPTIONAL.

6. OPTIONAL VENT LINES ARE SHOWN AS (---).

7. ALL VALVES ARE SHOWN IN DE-ENERGIZED POSITIONS.

HIGH POINT VENTS
PRELIM. CONCEPTUAL SCHEMATIC

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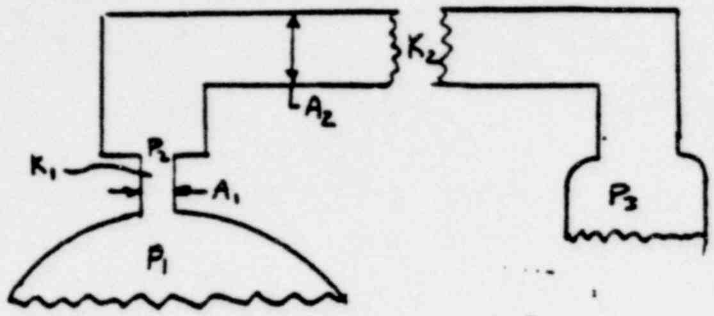
VENT FLOWRATE CURVES

A study was performed to determine the flow rate of water, saturated steam, super heat steam and non-condensible gases through High Point Vent Valves (HPV). The venting system envisioned is shown below:



P_1 = System Pressure at Vent

P_2 = Pressure at End of Vent Pipe

P_3 = Receiver Pressure

K_1 = Flow Resistance k-factor of Vent Pipe

K_2 = Flow Resistance k-factor of All Down Stream Piping, Valves, Turns, etc.

The venting rates will be a function of P_1, P_2, P_3, K_1, K_2 and type of fluid being vented. Since venting rates are a function of so many variables, the following restrictions were put on the calculations:

(1) Minimum vent area is at the system exit (A_1 on figure).

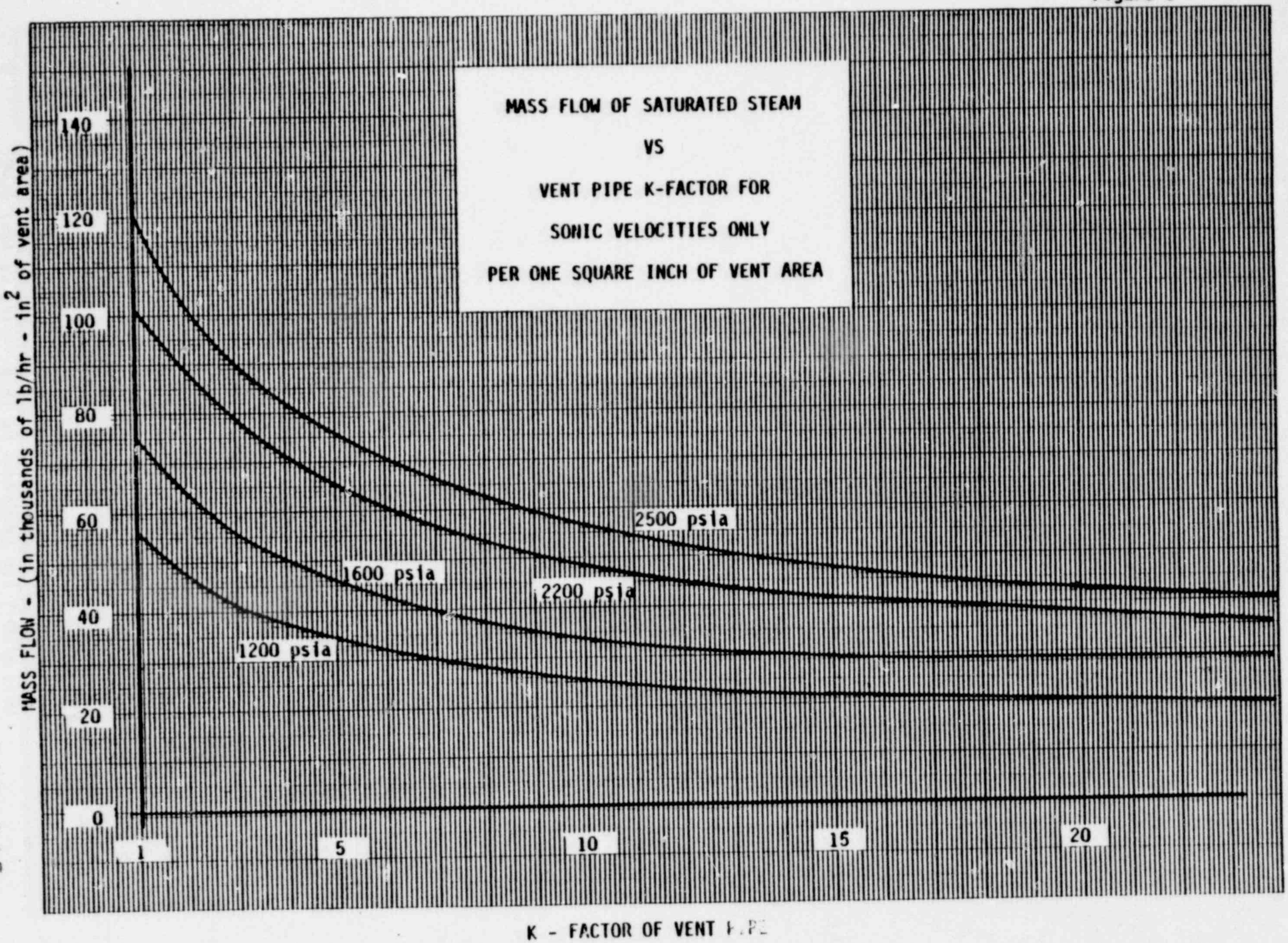
(2) P_1, A_2, K_2, and P_3 are such that P_2 is always less than critical pressure for sonic flow rates.

(3) Non condensible gas temperatures equal temperature of saturated steam.

Minimum flow rates (with sonic velocities) of saturated steam and non condensible gas as a function of K_1 and P_1 are shown on Figures 1 and 2. Superheated steam will have approximately the same Cp/Cv ratio as saturated steam and the flow rates will therefore be equal to the flow rates on Figure 1 times a correction factor. This correction factor is equal to the square root of the density ratio of superheated to saturated steam. Figure 3 shows venting rates of water as a function of the whole system k-factor and pressure per one square inch of vent area. Two phase flow will be assumed

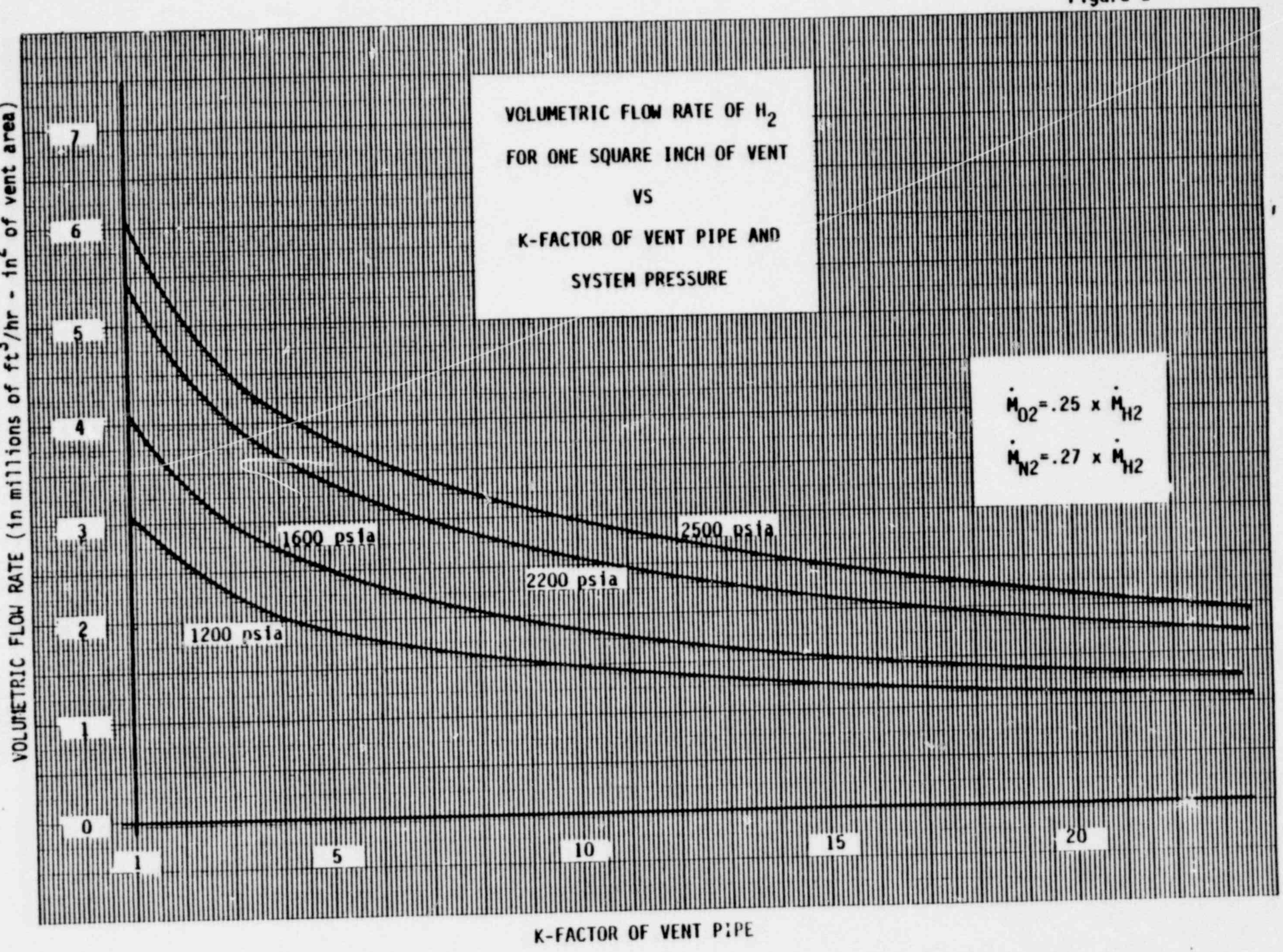
as the linear ratio of saturated steam and water weighed by the percent quality (i.e., zero percent quality would be all water, 50% quality would be half water half steam and 100% quality would be all steam).

Figure 1


MASS FLOW OF SATURATED STEAM
VS
VENT PIPE K-FACTOR FOR
SONIC VELOCITIES ONLY
PER ONE SQUARE INCH OF VENT AREA
2500 psia
2200 psia
1600 psia
1200 psia
140
120
100
80
60
40
20
0
1
5
10
15
20
MASS FLOW - (in thousands of lb/hr - in2 of vent area)
K - FACTOR OF VENT PIPE

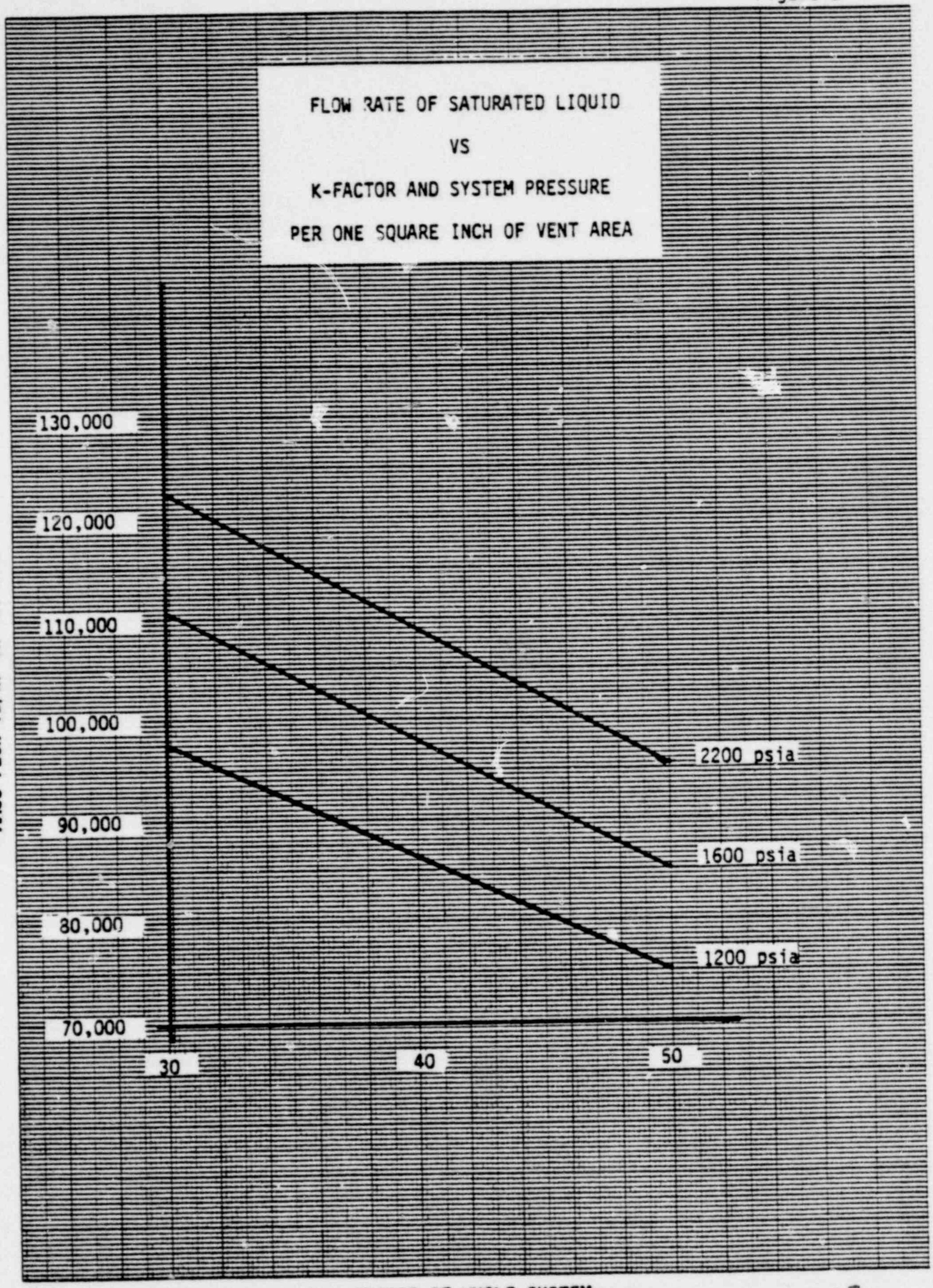
K·E 10 X 10 TO THE CENTIMETER 18 X 25 CM. KEUFFEL & ESSER CO. MADE IN U.S.A. 46 1510

Figure 2



1755 066

Figure 3



K•E 10 X 10 TO THE CENTIMETER 18 X 25 CM. KEUFFEL & ESSER CO. 46 1510

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ANCHOR MOTIONS,

SEISMIC RESPONSE SPECTRA

AND

ALLOWABLE NOZZLE LOADS

1755 068

INTRODUCTION

Contained herein are structural design data to be used in the design and analysis of the proposed high point vent valves. It is being proposed that remotely operated valves be made available: one at the top of the hot leg U bend, and another atop the pressurizer. The data presented in this document are worst case data and should be used for the design and analysis of all proposed valves.

The data consist of anchor motions, seismic response spectra, and allowable nozzle loads. Anchor motions are presented for various loading conditions including deadweight, thermal expansion, steady state hydraulics, operating basis earthquake (OBE), and design basis earthquake (DBE). The latter terminology, DBE, is synonymous to SSE or Safe Shutdown Earthquake. Steady state hydraulic loads are those loads induced by pressure expansion and fluid motion within the system and should be imposed on all components except the pressurizer for all loading conditions requiring the inclusion of operating loads. Thermal and deadweight anchor motions are self-explanatory; however, the thermal motions provided are the maximum experienced during operation at 8%, 15% and 100% power.

A seismic response spectra curve is provided for excitation in each of the X, Y, and Z directions shown in Figures 1 and 2 for both the OBE and SSE. Since a 1" diameter pipe is being considered, the values of critical damping recommended by Regulatory Guide 1.61 are adopted. Excitation at 1% critical damping is provided for the OBE and 2% for the SSE.

Allowable nozzle loads have been provide for a 1" schedule 160 pipe constructed of Ni-Cr-Fe SB167 material. Forces obtained from structural analyses should be compared by the method indicated on page 10 to the allowable loads listed there for the various load conditions. It is important to note that the forces and allowable loads are defined in the coordinate system shown in the accompanying sketch at the bottom of page 10.

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ANCHOR MOTIONS

- The coordinate system is defined in Figures 1 and 2.
- Resultants are obtained by the "square root of the sum of the squares" method.
- Displacements are in inches.
- Rotations are in radians.

	δ_x	δ_y	δ_z	θ_x	θ_y	θ_z
OBE						
X-Earthquake	.122	.001	.001	.00001	.00028	.00375
Y-Earthquake	.002	.008	.008	.00013	.00001	.00001
Z-Earthquake	.003	.026	.148	.00359	.00003	.00001
Resultant	.122	.027	.155	.00359	.00028	.00375
DBE(SSE)						
X-Earthquake	.218	.001	.003	.00002	.00053	.00674
Y-Earthquake	.004	.014	.015	.00023	.00002	.00002
Z-Earthquake	.006	.048	.273	.00654	.00006	.00002
Resultant	.218	.050	.273	.00654	.00053	.00674
DEADWEIGHT	.012	-.047	-.020	-.00040	.00000	.00005
THERMAL EXPANSION	-.049	3.6[illegible]3	-.534	-.00139	-.00002	.00005
STEADY-STATE HYDRAULICS	.000	.096	-.024	-.00023	.00000	.00000

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RESPONSE SPECTRA

Response spectra are provided on the following pages for the conditions listed below:

OBE - Operating Basis Earthquake

X - Direction

Y - Direction

Z - Direction

SSE - Safe Shutdown Earthquake

X - Direction

Y - Direction

Z - Direction

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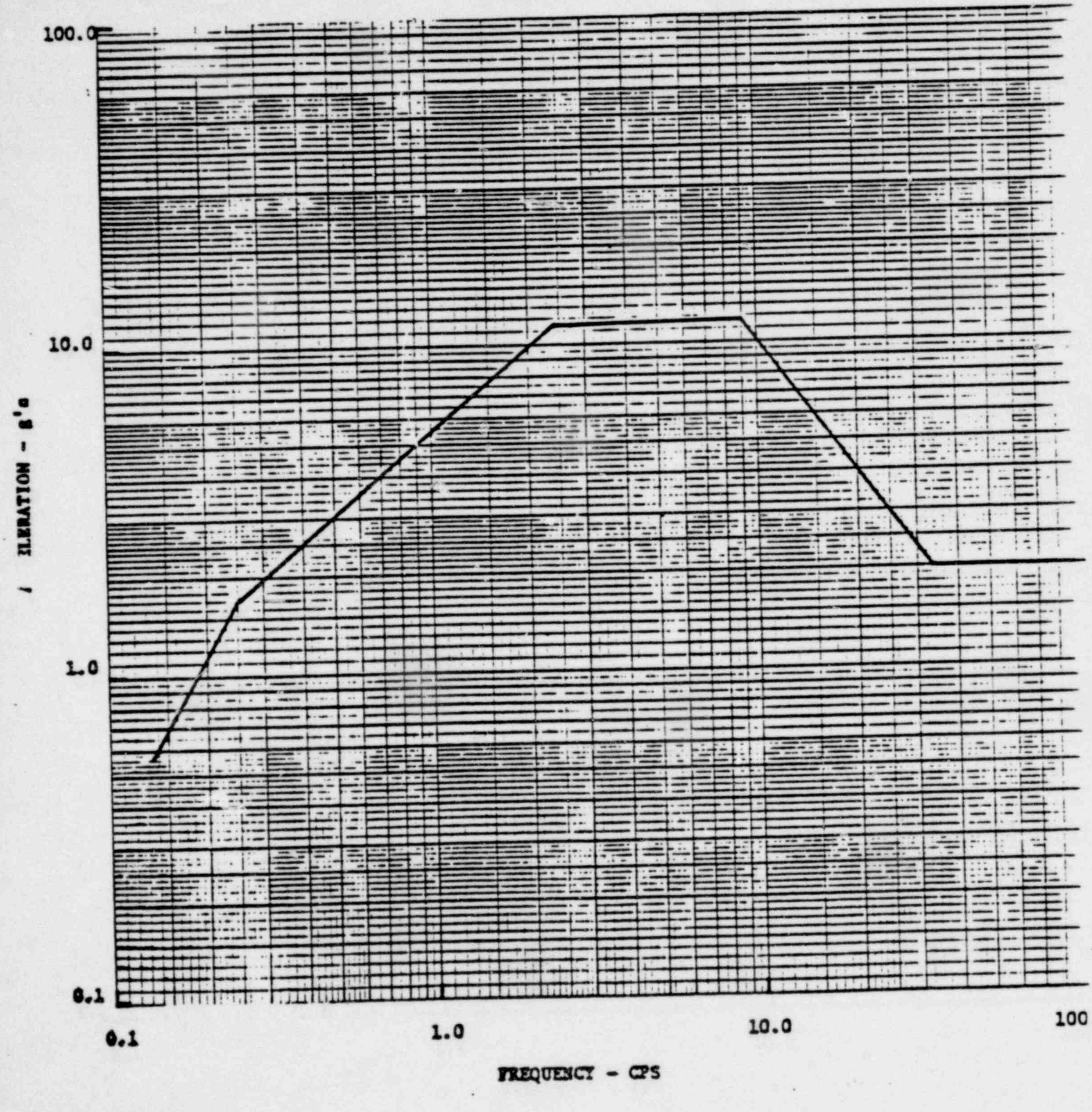
RESPONSE SPECTRA

X-DIRECTION EARTHQUAKE - OBE
1% CRITICAL DAMPING



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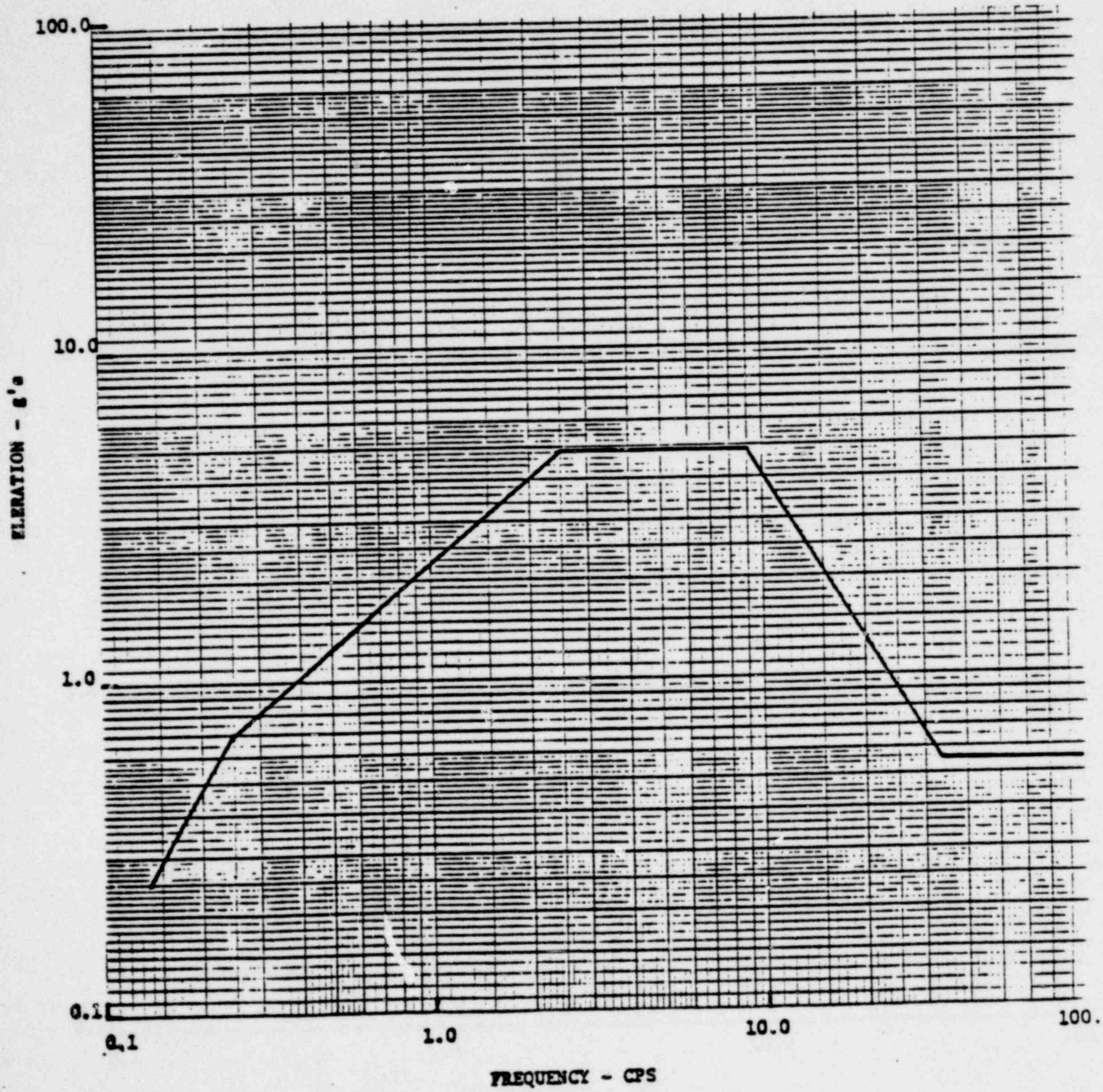
RESPONSE SPECTRA

Y-DIRECTION EARTHQUAKE - OBE
1% CRITICAL DAMPING



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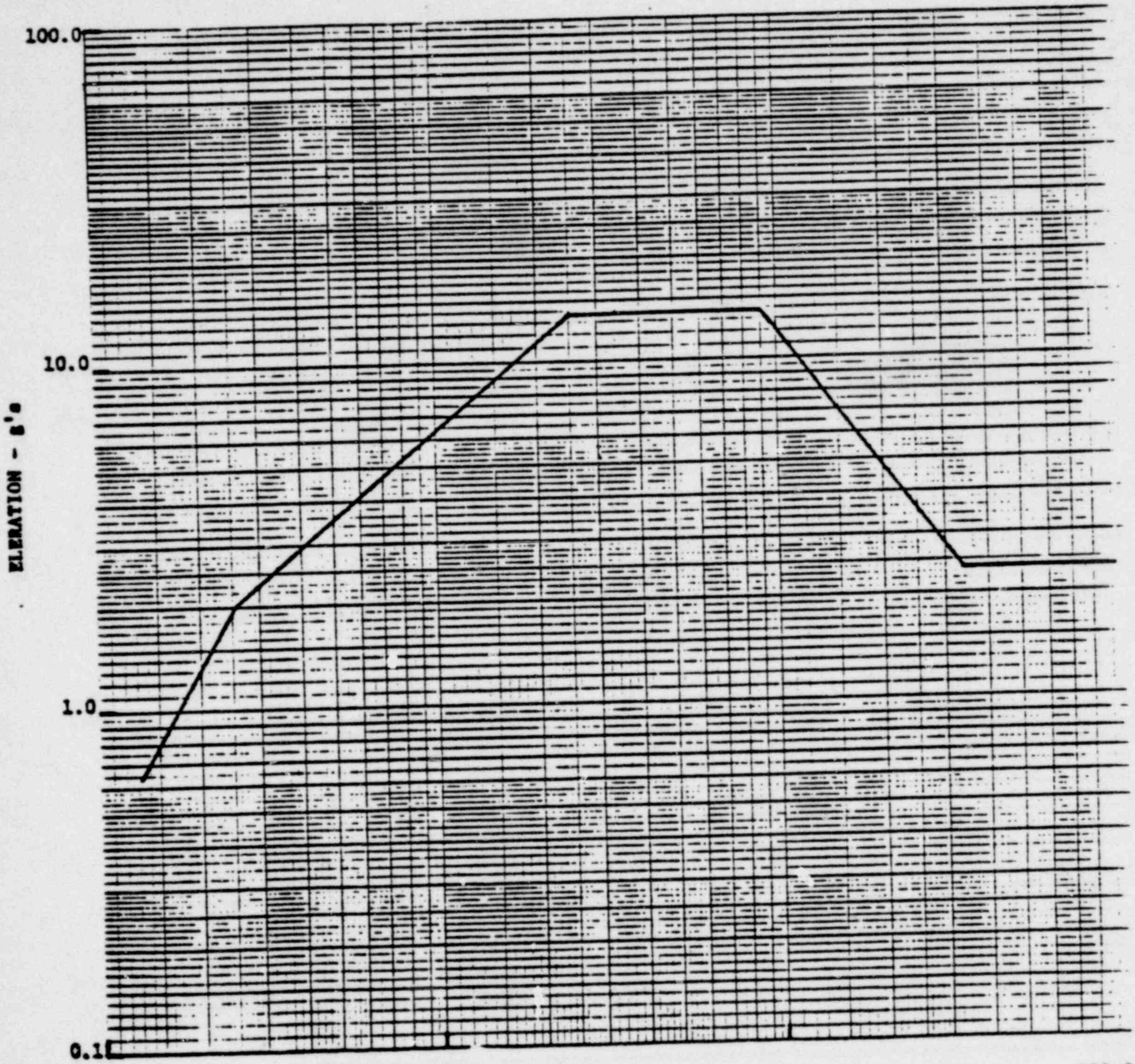
RESPONSE SPECTRA

Z-DIRECTION EARTHQUAKE - OBE
1% CRITICAL DAMPING



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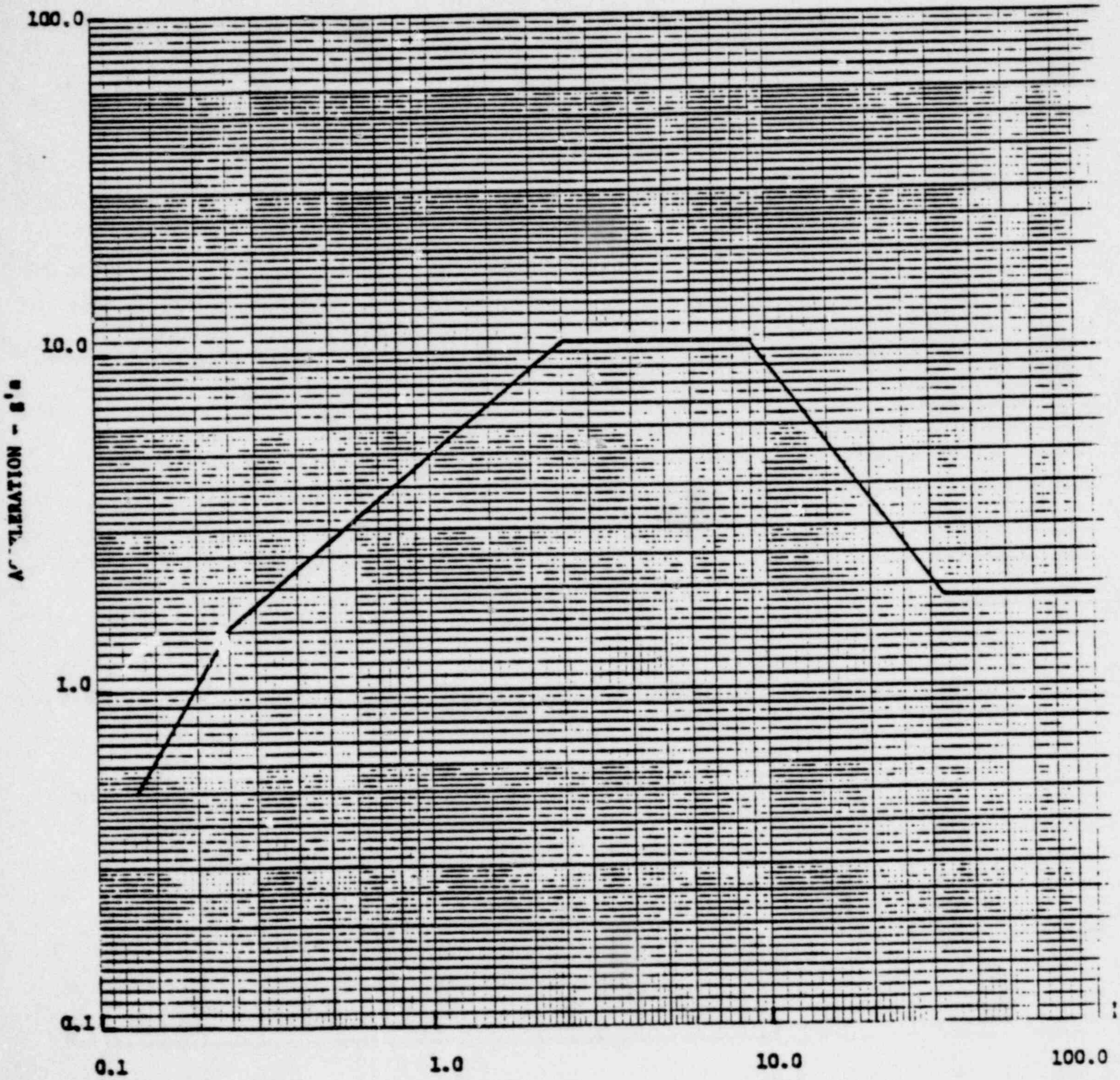
RESPONSE SPECTRA

X-DIRECTION EARTHQUAKE - SSE
2% CRITICAL DAMPING



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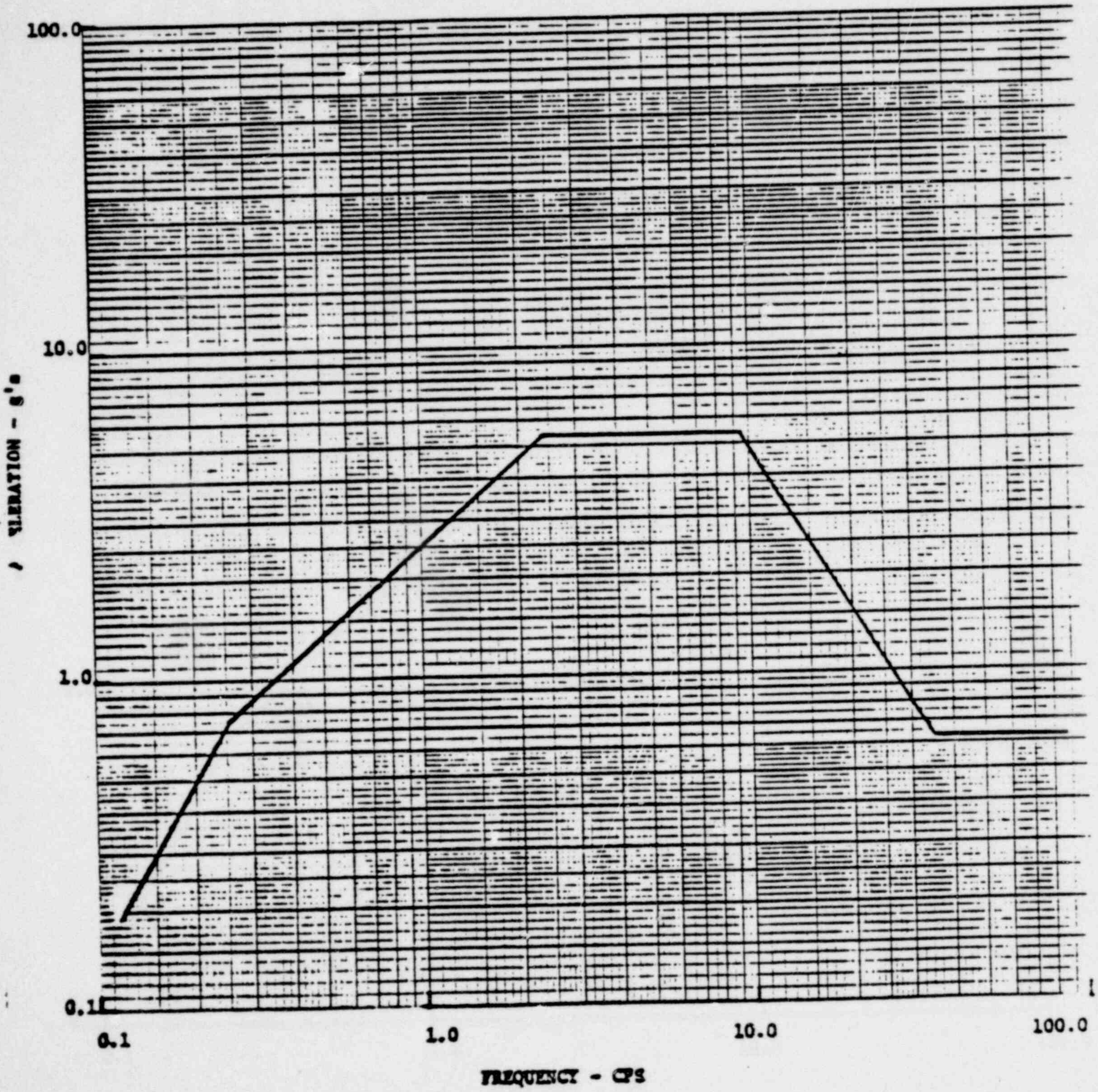
RESPONSE SPECTRA

Y-DIRECTION EARTHQUAKE - SSE
2% CRITICAL DAMPING



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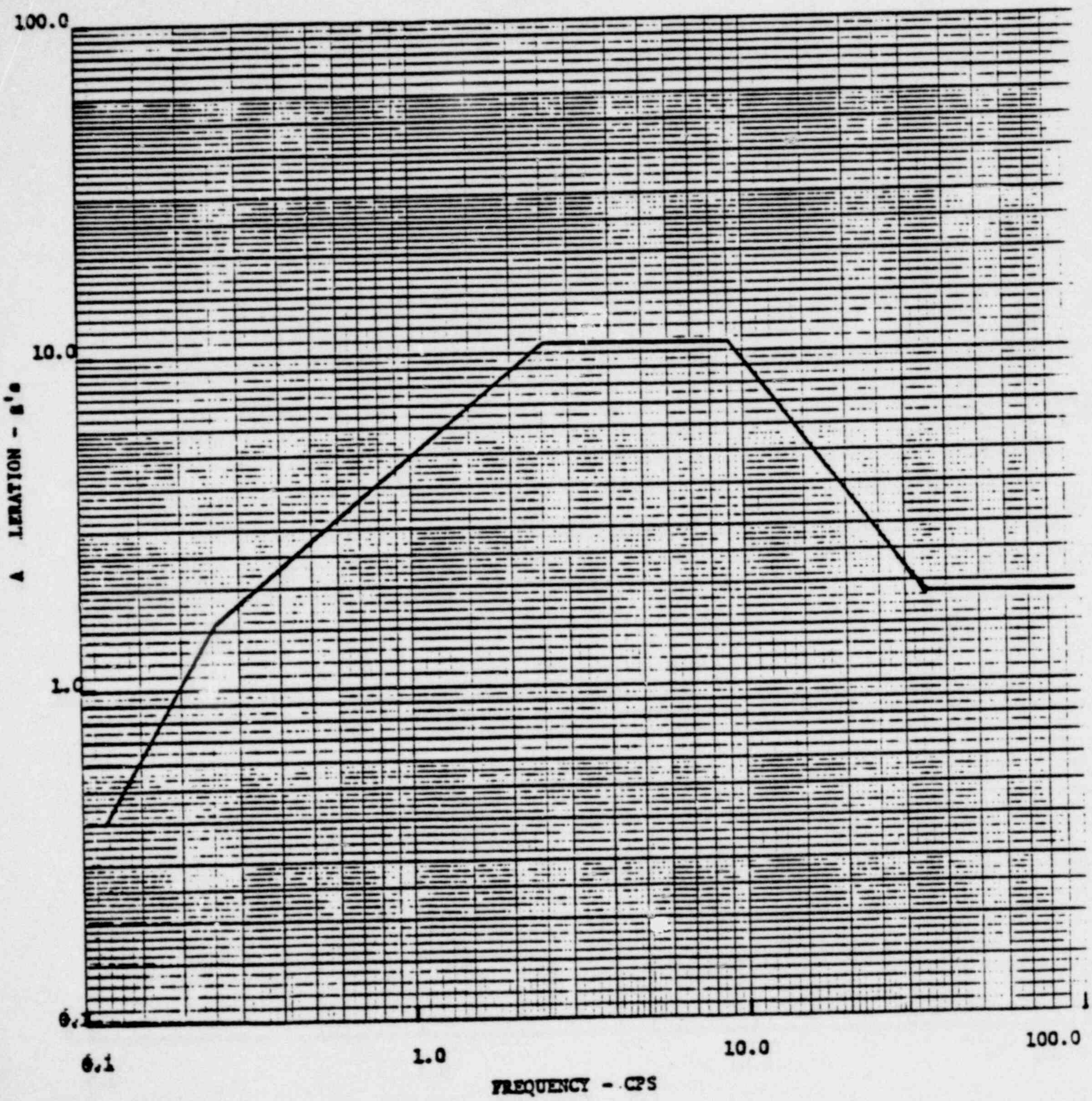
RESPONSE SPECTRA

Z-DIRECTION EARTHQUAKE - SSE
2% CRITICAL DAMPING



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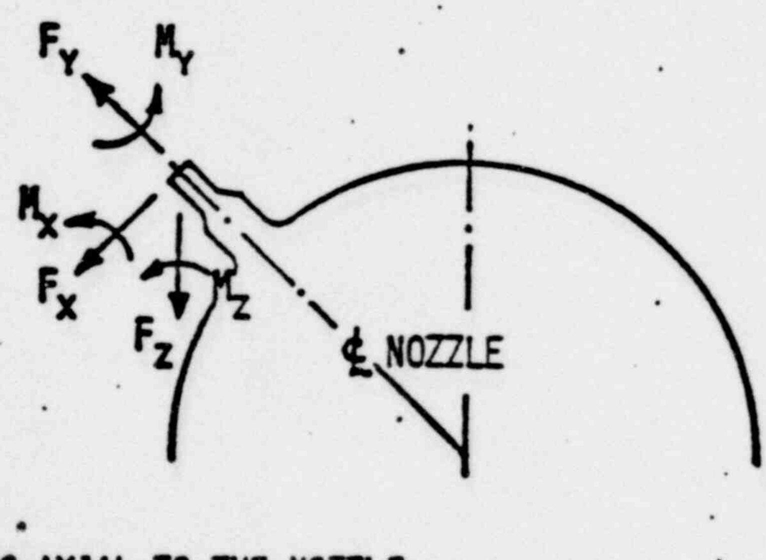
ALLOWABLE NOZZLE LOADS

FOR

1" SCHEDULE 160 PIPE OF Ni-Cr-Fe SB167 MATERIAL

LOAD CASE *	FORCES (KIPS) $\sqrt{F_X^2 + F_Z^2}$	F_Y	MOMENT (FT.-KIPS) $\sqrt{M_X^2 + M_Z^2}$	M_Y
DESIGN	.624	.442	.295	.084
LEVEL A & B	.832	.588	.390	.112



F_Y IS AXIAL TO THE NOZZLE

F_X,F_Z ARE TRANSVERSE TO THE NOZZLE

ALL MOMENT FOLLOW RIGHT HAND RULE

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* SEE FOLLOWING PAGE FOR DESCRIPTION OF LOAD CASE.

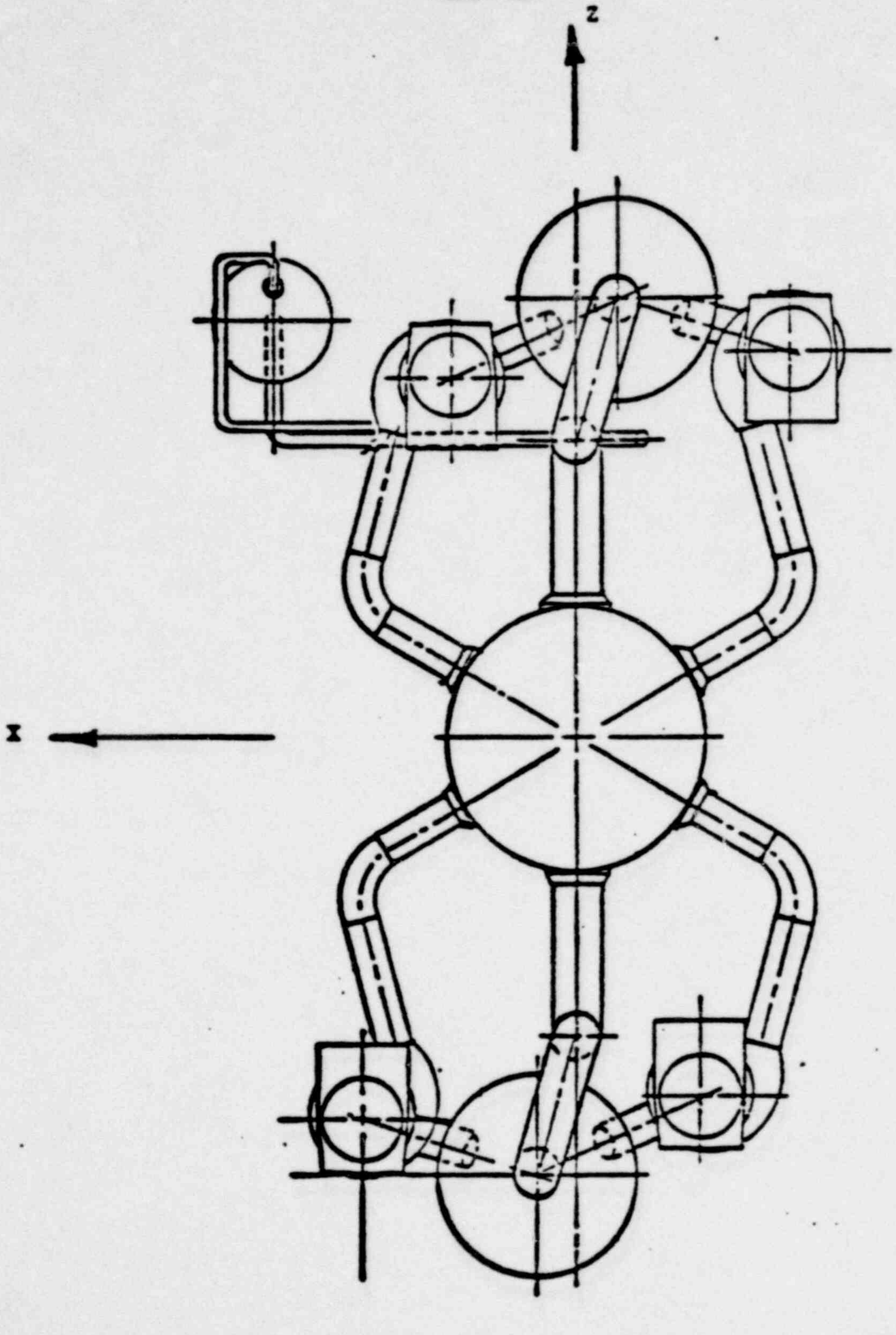
LOAD CASE DESCRIPTIONS

LOADING CONDITIONS	COMBINATION OF LOADS	ASME CODE STRESS LIMITS
Design Pressure Loading	Internal Design Pressure	See ART. NB-3000
Level A (Design Condition)	Deadweight Loads + Internal Pressure + OBE + Steady State Hydraulics	See ART. NB-3000
Level B (Normal and Upset)	Deadweight Loads + Thermal Loads + Internal Pressure + OBE + Thermal Gradients + Steady State Hydraulics	See ART. NB-3000

1755 079

REACTOR COOLANT SYSTEM

PLAN VIEW



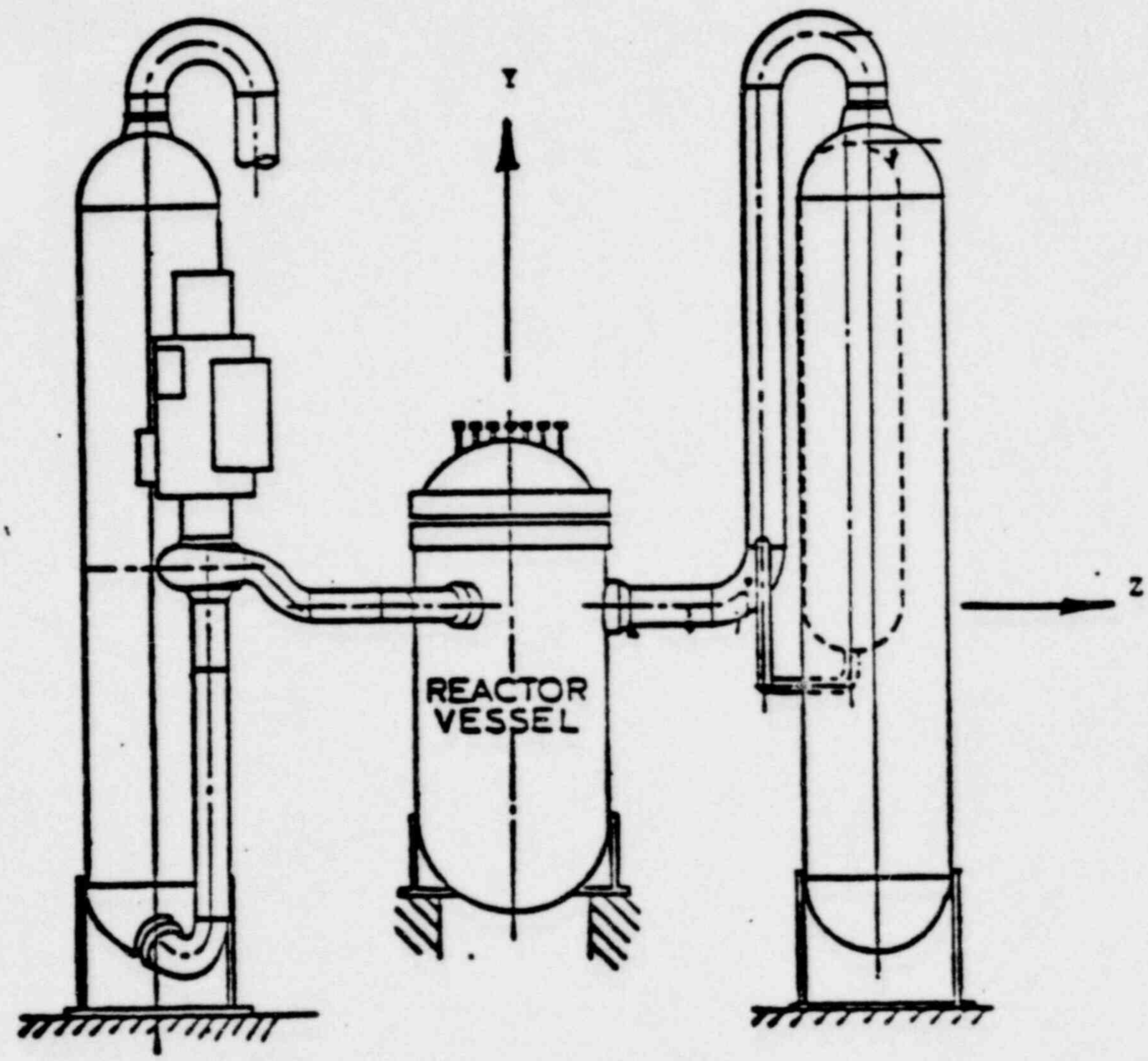
(GLOBAL COORDINATE SYSTEM SHOWN)

1755 080

FIGURE 1

REACTOR COOLANT SYSTEM

ELEVATION VIEW



(GLOBAL COORDINATE SYSTEM SHOWN)

FIGURE 2

1755 081

ATTACHMENT V

Rev. 28 10/25/79

ADMINISTRATIVE INSTRUCTIONS

AI-500

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

CONDUCT OF OPERATIONS

REVIEWED BY: Plant Review Committee

Paul J. McGee

Date 10/25/79

Meeting No. 79-42

APPROVED BY: Nuclear Plant Manager

David C. Poole

Date 11-15-79

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Section (Cont'd) **Page**

ENCLOSURES

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1.0 INTRODUCTION

Plant operations are conducted by the Operations Section with the cooperation and assistance of personnel from other plant sections, corporate headquarters, and non-Company organizations as required. The instructions contained in this section do not supersede or nullify applicable legal requirements or regulations. It is the purpose of these instructions to insure that plant operations are conducted in conformance with applicable legal requirements and regulations, and dictates of good operating practice.

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2.0 SHIFT OPERATIONS

Plant operations are conducted by shift operating personnel assigned to the Operations Section under the direction of the Shift Supervisor.

2.1 SHIFT COMPLEMENT

Each operating shift for Unit 3 will normally consist of a Shift Supervisor, an Assistant Nuclear Shift Supervisor, a Chief Nuclear Operator, a Nuclear Operator, two Assistant Nuclear Operators, a Nuclear Auxiliary Operator, and an Assistant Nuclear Auxiliary Operator. During cold shutdown or refueling conditions, the shift complement may be reduced to a minimum of a Shift Supervisor, a Nuclear Operator, and a Nuclear Auxiliary Operator. The total number and classification of personnel assigned to each shift will be determined by the Operations Superintendent as dictated by plant conditions and anticipated operations. In all cases, the minimum shift composition shall be in compliance with the provisions of Technical Specifications Table 6.2-1 below:

MINIMUM SHIFT CREW COMPOSITION -

LICENSE CATEGORY	APPLICABLE MODES	
	1,2,3,4	5 and 6
SOL	1	1*
OL	2	1
Non-Licensed	3	1

NOTE: Shift crew composition may be less than the minimum requirement for a period of time not to exceed 2 hrs. in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

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*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator LIMITED TO FUEL HANDLING individual supervising core alterations after the initial fuel loading.

a. In addition to the minimum shift requirements of Table 6.2-1, two licensed Reactor Operators shall be in the Control Center during startup and scheduled shutdown of the reactor and during recovery from reactor trips caused by transients or emergencies.

b. At least one licensed Reactor Operator shall be in the reactor building when fuel handling operations are in progress in the reactor building. An operator holding a Senior Reactor Operator's License or a Senior Reactor Operator LIMITED TO FUEL HANDLING and assigned no other concurrent operational duties shall be in direct charge of refueling activities.

c. (deleted)

2.2 ADDITIONAL SHIFT PERSONNEL

Such additional operating personnel, as may be required because of unusual plant conditions or operational needs, will be obtained as directed by the Shift Supervisor. Normally, the Operations Superintendent is

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consulted when arrangements are made for large numbers of operating personnel, but this notification shall not restrict the Shift Supervisor from obtaining such personnel as required for plant operations. It should be noted, however, that operations requiring additional personnel will not be undertaken until the shift is sufficiently manned.

2.3 AUTHORITIES AND RESPONSIBILITIES FOR SAFE OPERATION AND SHUTDOWN

The person charged with the duty of reactor operation has the authority and responsibility for shutting the reactor down to a safe condition by approved procedures when he determines that the safety of the reactor/plant is in jeopardy or when operating parameters exceed any of the reactor protection set points and automatic shutdown does not occur. Upon determination of shutdown, he shall notify the Shift Supervisor in charge who will insure that the plant is shut down in safe condition by approved procedures.

The Shift Supervisor in charge has the responsibility to notify the Operations Superintendent or, in his absence, the person on-call and together they determine the circumstances, analyze the cause, and determine that operations can proceed safely before the reactor is returned to power after a trip or unscheduled or unexplained power reduction. Approval to take the reactor critical following a trip rests with the Operations Superintendent or, in his absence, the person on-call, but the actual startup shall be authorized by the Shift Supervisor in charge and the authorization shall be documented in the reactor startup procedure.

The Shift Supervisor in charge has responsibility to be present at the plant and to provide direction for returning the reactor to criticality or power following a trip or unscheduled or unexplained power reduction.

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All persons charged with the duty of operating the plant have the responsibility to:

a. Believe and respond conservatively to instrument indications unless they are proven to be incorrect by instrument channel check or instrument channel test.

b. Adhere to Technical Specifications.

The Shift Supervisor in charge and the Operations Superintendent have the responsibility to review routine operating data to assure safe operation.

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2.4 SHIFT FUNCTIONS

The personnel assigned to shift operations perform or are prepared to perform three general functions:

a. Continuous normal operation of the plant and its associated equipment, including normal planned power changes, startups, and shutdowns.

b. Maintain the plant in a safe condition during abnormal conditions.

c. Protect plant personnel, the health and safety of the public, and plant equipment during and following an emergency situation.

All major plant operations are conducted from the Control Center in accordance with the Shift Supervisor's instructions. The Shift Supervisor, consistent with the provisions of POQAM (Plant Operating Quality Assurance Manual), effects the Load Dispatcher's orders and the directions of plant staff supervisors. The Shift Supervisor has the responsibility to maintain a broad perspective of operational conditions affecting plant safety as the matter of highest priority at all times when on duty.

Although the Shift Supervisor's normal duty station is the Control Center, he may be anywhere in the plant his attention is required. In his absence, the Assistant Nuclear Shift Supervisor shall assume duty in the Control Center and have responsibility for interpreting the Shift Supervisor's instructions and directing operations.

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During abnormal or emergency conditions where multiple operations are required, the Shift Supervisor shall assume a "command" role. He shall base his decisions on an overview of the condition and direct the activities of the Control Center Operators to insure reactor safety. The Shift Supervisor shall remain in the Control Center at all times during an abnormal or emergency condition until properly relieved. The persons authorized to relieve a Shift Supervisor of his "command" duties are another Shift Supervisor, an Assistant Nuclear Shift Supervisor, or the Operations Superintendent.

Emergency conditions may also require the Shift Supervisor to act as temporary Emergency Coordinator. Responsibilities and authorized relief for this function are defined in EM-100, Emergency Plan. During all conditions, the operating instructions of the Shift Supervisor may not be superceded except by the Nuclear Plant Manager, the Operations Superintendent, or the person on-call.

During all operational conditions, shift operations personnel will be guided by POQAM, supervisory direction, and technical advice and assistance. The Assistant Nuclear Shift Supervisor assists the Shift Supervisor by directing and overseeing the routine operations of the plant in accordance with the Shift Supervisor's general instructions and by performing the preparation and review of changes to POQAM. He also provides assistance by assuring the timely completion of required administrative functions such as Radiation Work Permits (RWP's), Work Requests, and Surveillance Procedures. The Chief Nuclear Operator provides direction and assistance to the operators in the field in the performance of assigned tasks. He is available to provide assistance

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in any area of the plant and makes periodic in-plant rounds to observe operating equipment. A Nuclear Operator and one Assistant Nuclear Operator will normally be in the Control Center and will be operating the plant controls and equipment and maintaining the operating logs. The other Assistant Nuclear Operator and a Nuclear Auxiliary Operator will be in the plant making rounds to observe operating equipment, record operating data, and execute routine operations. In "on-the-job" training functions, Assistant Nuclear Auxiliary Operators will be in the plant assisting the Nuclear Auxiliary Operator or performing work functions as dictated by plant conditions.

At least one CR-3 licensed operator shall be in an area in front of the control board at all times with an unobstructed view of and access to the operational control board. The general area is defined as that area enclosed by the installed red tile stripping in front of the control board.

In the event of an emergency affecting the safety of operations, the operator at the controls may momentarily be absent from the general area in front of the control board in order to verify the receipt of an annunciator alarm or initiate corrective action provided he remains within the confines of the Control Center and maintains an unobstructed view of the operational control panels.

The operator at the controls should not under any circumstances leave the red lined general area for any non-emergency reason without obtaining a qualified relief operator at the controls. Both the Nuclear Operator and the Assistant Nuclear Operator on duty in the Control Center shall at all times be prepared to assume the responsibility of the "operator at the controls" to allow either to leave the red lined general area for non-emergency reasons.

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The controls of the plant shall be manipulated by a licensed operator or Senior Operator except that an individual may manipulate the controls as part of his training to qualify for an operator license under the direction and in the presence of "a licensed operator or Senior Operator".

2.5 CONTROL CENTER RECORDS

Certain records are maintained either in the Control Center or the Shift Supervisor's Office to facilitate communications and operations. These records will consist of but not be limited to the following:

a. Control Center notebook containing short-term instructions, "on-call" notification, Reactor Trip Log, Generator Under Frequency Log, and In-Plant and System Switching and Tagging List.

b. Shift Supervisor's Log

c. Operator's Log

d. Key Control Log

e. Equipment Out-of-Service Log

f. (deleted)

g. Current Radiation Work Permits

h. (deleted)

i. Jumper Log

j. Equipment Clearance Order Log

2.6 ON-CALL

Either the Nuclear Plant Manager or his authorized designee shall be at the plant or on-call at all times. When the Nuclear Plant Manager or his authorized designee is absent from the plant, during backshifts,

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weekends, or holidays, he shall be provided with a voice pager by which the Shift Supervisor can contact him regarding plant problems. At times when use of the voice pager is impossible, he shall advise the Shift Supervisor of the location and telephone number at which he may be contacted. Any Responsible Supervisor may be designated as on-call provided he holds a valid NRC Senior Reactor Operator's License.

2.7 PROCEDURES

Operating Procedures are divided into two basic groups, those which require step-by-step sign-off or initials for proper verification of vital procedural steps and those which are written as operator guides and require no sign-off's or approval.

Procedures dealing with systems which directly affect the reactor, reactor coolant (RC) system, and engineered safeguards (ES) will be of the sign-off and approval type while procedures dealing with secondary and auxiliary systems not directly affecting the reactor, RC system, or ES will be utilized as operator guides.

The procedures used as operator guides may be checked off as the step or section is completed, but procedure verification and/or retention is not required.

Procedures requiring proper verification will be handled in the following way. When an operator has performed an operation or verified the status of the item as required by the specific procedural step, he initials the space provided for that step in the procedure or on a Check-Off List.

1755 094

Sequences or steps in procedures that cannot be performed due to clearances, abnormal conditions, or emergency operating modes shall be marked as "Not Applicable" (N/A) and the specific reason for not performing the step shall be noted and initialed by the Shift Supervisor prior to proceeding.

When all procedural steps have been completed and the required spaces initialed, the operator signs and dates the procedure. The Shift Supervisor or Assistant Shift Supervisor attaches a Procedure Approval and Transmittal Sheet (Form 912019 - RE: AI-400, Section 5.6) to the completed procedure. He then reviews the completed procedure, completes and signs the Procedure Approval and Transmittal Sheet, and forwards the completed procedure to the Operations Superintendent. The Operations Superintendent reviews the completed procedure and forwards it to the Compliance Section for review.

For the purposes of documentation and record retention, the Procedure Approval and Transmittal Sheet attached to the completed surveillance data sheets or the applicable procedural section(s) dealing with the function performed shall be considered a complete procedure.

2.7.1 Adherence to Procedures

Refer to Sections 5.3, 5.4, and 6.0 of AI-400, Plant Operating Quality Assurance Manual Control Document.

1755 095

2.8 C ERATOR RETRAINING

See FSAR, Appendix 12C.

2.9 PLANT MODIFICATION AND SET POINT REVISION NOTIFICATION

To promptly and effectively advise shift personnel of plant modifications and set point revisions, the Operations Superintendent will transmit to the Shift Supervisors a completed copy of Form 912163, Modification Approval Record (MAR), along with a Form 912235, Change Notification. Each Shift Supervisor will insure that each member of his shift has been informed of the plant modification or set point revision and initial the Change Notification. Forms which have been initialed by all Shift Supervisors shall be maintained in the Control Center until the modification is complete, then returned via the Operations Engineer to the Operations Superintendent for filing.

2.10 PLANT OPERATING QUALITY ASSURANCE MANUAL REVISION NOTIFICATION

Temporary and permanent revisions are made to POQAM in accordance with Section 8.0 of AI-400, Plant Operating Quality Assurance Manual Control Document. All revisions made will use Form 91210, Procedure Review Record (PRR). When the revisions are approved, they will be transmitted along with a Form 912235, Change Notification, to the Shift Supervisor who will initial the Change Notification, signifying that each member of his shift has been informed of the revision. It is the responsibility of each Shift Supervisor to insure that his shift is informed. Forms which have been signed by all Shift Supervisors shall be returned via the Operations Engineer to the Operations Superintendent who will file the change form for the latest revision in the POQAM Revision Log.

1755 096

2.11 SUPPLEMENTAL LABORATORY ANALYSIS REQUEST

Occasionally plant modes or conditions dictate that supplementary data be collected to support recovery from off-standard conditions. To insure that the additional data is collected and documented, Form 912225 (Enclosure 2), Supplemental Laboratory Analysis Request Form, shall be sent to the ChemRad Laboratory as conditions dictate. Because initial laboratory sampling may be required on an "as soon as possible" basis, verbal requests will be accepted from the Nuclear Operator, Chief Nuclear Operator, Assistant Shift Supervisor, or Shift Supervisor with the understanding that Form 912225 (Section 1 complete) will be provided to the technician by the time results are available for logging. If requested in Section 1 of Form 912225, results can be verbally transmitted as each is logged on the form. Upon completion of Section 2, the form will be obtained from the ChemRad Office. Section 3 of the form is then completed in accordance with AI-500, Conduct of Operations, Sections 2.5(c) (Control Center Records), 2.7 (Procedures), and 2.20 (Control Center Status Board).

2.12 GENERAL PRACTICES FOR COLLECTING SHIFT RECORDS

Shift records are comprised of the logs, data sheets, recorder charts, and computer printouts which describe or record operating information and action. The information obtained from these records is useful for current operations and for analysis of previous operations. The following general practices are applicable to shift records:

a. All log entries, data sheets, and chart notations must be legible, accurate, complete, and understandable.

1755 097

b. The individuals responsible for maintaining logs and data sheets are responsible for signing and dating the portions of the records which cover their shift assignments.

c. Each in-service recorder chart shall be checked at least once per shift (normally during the first hour of the shift) to verify that the marking is legible and the timing correct.

1755 098

d. Before the chart is placed on or removed from the recorder, it shall be marked with the date, recorder identification symbol, and parameter being recorded. It shall be date stamped daily on the 0000-0800 shift.

e. When significant events or unusual trends in transients occur, the resulting recorder traces are to be identified as to the time and event notation to assist in operations analysis.

f. The Shift Supervisor on each shift will review the data sheets and log records compiled and recorded during that shift. As a part of his tour through the plant, the Assistant Nuclear Shift Supervisor will review log records and data sheets at operating stations outside the Control Center. This review is to detect unusual or abnormal trends or readings that require investigation or remedial action and to check on the completeness and accuracy of the records. The Shift Supervisor or the Assistant Nuclear Shift Supervisor signifies completion of this review by initialing the data sheet or log record and recording the time the review was made.

g. At midnight of each day, the shift records are assembled and checked for completeness by the Shift Supervisor on duty. The shift records are then sent to the Operations Superintendent.

h. The Operations Superintendent is responsible for reviewing shift records. He advises the respective Shift Supervisor of any deficiencies in compiling the information and initiates measures to eliminate the deficiencies.

i. The Assistant Nuclear Shift Supervisor is responsible for insuring that all supplies necessary for keeping shift records (charts, ink, procedures, forms, logbooks, check-off sheets, etc.) are in ample supply.

1755 099

j. Shift operating logs will be taken once per shift (normally during the first hour of the shift) unless otherwise specified.

2.13 SHIFT LOGS

The narrative log notations of plant conditions, operations, and events are a vital portion of the shift records. (Errors in a shift log are corrected by drawing a single line through the incorrect information and writing the correct information adjacent to or in space available with reference to the deleted information. The individual making the correction shall initial and date the deleted information.) All entries shall be made in black ink.

a. The Operator's Log is maintained on a shift basis to record the plant status and events in chronological order. Log entries may include but are not limited to the following:

- Date
- Names of Shift Personnel
- Plant Status
- Water Quantity Used for Makeup
- Waste Disposal System Status
- Starting and Stopping of Equipment
- Change of Auxiliary System Configuration
- Water Quantity Used for Dilution
- Water Quantity and Its Source Used for Boration
- Performance of Surveillance Tests
- Completion of Specified Check-Off Lists
- Maintenance Activities Affecting Operations
- Occurrence of Significant Annunciator Alarms

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- Performance of Special Inspections or Checks (Overspeed Trip, Oil Filters, etc.)
- Reactor Trips
- Instrument or Equipment Malfunctions or Failures
- Unusual Trends or Conditions Observed
- Company Electrical Grid Events Affecting Operations
- Relay Operations and Targets
- Starting and Stopping Gaseous or Liquid Waste Disposal Discharges (List release permit number.)
- Alarm Tests
- The number of the procedures used to perform any of the above operations.

At the end of each shift the Nuclear Operator signs the Operator's Log, signifying that the entries are a complete and accurate record of plant operations. The current logbook and the most recent "back copy" are to be retained in the Control Center.

b. The Shift Supervisor's Logbook is a multiple-sheet record form in a hardbound book. The Shift Supervisor summarizes plant conditions and events during his shift. Events entered may include any of those noted in the Operator's Log, however, the shift report need not repeat routine items which have no safety significance or little operational importance, but will include a detailed explanation of major events.

The Shift Supervisor's Log shall begin with plant status information and should include any changes in status of the availability of systems, unusual occurrences, results of liquid or gaseous releases, sample analysis, and changes of major auxiliary equipment service. At the end of each shift, the Shift

1755 101

Supervisor signs the log signifying that the report is a comprehensive, accurate summary of plant events and activities.

At the end of the 24 hr. period, the copy of the Shift Supervisor's logsheet is detached from the book and forwarded to the Operations Superintendent for review. The original remaining in the book is retained in the Control Center for reference and record. All logbooks except the current one and the most recent "back copy" are transmitted to the Administrative Supervisor for disposition.

2.14 REACTOR TRIP AND PLANT SHUTDOWN

A reactor trip is any reactor protection system (RPS) action, manual or automatic, which causes the de-energizing of the control rod drive mechanisms (CRDM's) and allows the control rod assemblies to drop into the core. A plant shutdown is the opening of generator breakers 1661 and 1662. When a reactor trip or plant shutdown occurs, the Shift Supervisor takes the following action:

a. Insures that the plant is placed in a safe condition by having the necessary operations performed in accordance with approved procedures.

b. Notifies the Operations Superintendent or person on-call.

c. Determines the subsequent action to be taken.

d. Completes Steps 1 thru 8 on Form 912212 (Enclosure 3), Reactor Trip and Shutdown Report, assigns the next consecutive report number, and forwards it to the Operations Superintendent for disposition. A log of Reactor Trip and Shutdown Report dates and types will be maintained in the Control Center notebook.

e. Insures the reactor trip information is entered in the Reactor Operator's Log and Shift Supervisor's Log (reactor trip only).

The reactor will not be taken critical following a reactor trip until a determination of the cause of the trip has been made, corrective action taken, and approval to take the reactor critical has been obtained from the Operations Superintendent or person on-call.

In the interim between trip and approval for recovery, the Shift Supervisor may authorize the withdrawal of Safety Group 1 provided a $\geq$ 1% $\Delta k/k$ shutdown margin is maintained and rod withdrawal is not prohibited by any RPS action statements of Standard Technical Specifications.

2.15 SHORT-TERM INSTRUCTIONS

Short-term instructions are any miscellaneous instructions that may arise and shall be used for routine maintenance and personnel instruction where the plant safety is not affected. All short-term instructions shall automatically expire in 90 days if not previously cancelled. It is the Shift Supervisor's responsibility to review, audit, and maintain the short-term instructions in a current and up-to-date condition. If it is necessary to continue the short-term instruction, it shall be reissued using a new document number.

Short-term instructions shall not be amended. If a change is necessary to an issued short-term instruction, it shall be cancelled and reissued in its correct form.

To complete the Short-Term Instruction Form (Enclosure 6), insert the document number. The document number will be prefixed by the year, then the next progressive short-term instruction number; insert the date.

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The length of effect shall be as conditions dictate up to 90 days. Complete the instruction and action as required. The short-term instruction is immediately effective when signed by a Shift Supervisor or the Operations Superintendent. Any deviation from short-term instructions shall be logged in the Control Center Logbook and reported to the Shift Supervisor on duty. The instructions are logged on the Short-Term Instructions Index (Enclosure 6). Log entry is completed by inserting the document number, subject, and date entered. When it is necessary to cancel a short-term instruction, this may be performed by completing the "Date Removed" and "Removed By" portions of Enclosure 6. Official instruction removal occurs with the initials of a Shift Supervisor or the Operations Superintendent. Cancelled short-term instructions shall be destroyed.

2.16 CONTROL CENTER ACCESS

The Shift Supervisor is responsible for maintaining control of personnel entering the Control Center whether entry is for observing operations, conducting tests, or performing maintenance. The Shift Supervisor is authorized to refuse entry or direct personnel to leave the Control Center.

When an operational transient or accident occurs, the Shift Supervisor, Assistant Nuclear Shift Supervisor, and other Operations personnel in the Control Center shall have an unobstructed view of and immediate access to the operational controls. Additionally, these personnel shall have access to operational auxiliaries such as the computer line printer, the IBM-5100 computer, and the Environmental Monitoring Panel. The Shift Supervisor or Assistant Nuclear Shift Supervisor

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shall limit the Control Center to only those personnel who are essential for the direct operation of the plant and to technical advisors relative to support the particular operating condition.

Based on the specific operational condition and personnel essentiality to that condition, the Nuclear Plant Manager or Operations Superintendent may authorize reprogramming of the security key-card system to restrict access to the Control Center.

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2.17 SHIFT RELIEF

Shift operating personnel, when on duty, are to remain on duty with full responsibilities of their position until properly relieved. A Shift Supervisor, Assistant Nuclear Shift Supervisor, Chief Nuclear Operator, Nuclear Operator, Assistant Nuclear Operator, Nuclear Auxiliary Operator, or Assistant Nuclear Auxiliary Operator is considered to be properly relieved when the individual assigned to relieve him is properly qualified and licensed (if required) to assume the position and has been informed of the status of the plant, operations in progress, and special instructions if any are applicable. The relieving individual has the responsibility of (1) reviewing the Operator's Log, status board, Equipment Out-of-Service Log, and data sheets; (2) discussing operations with on-duty personnel; (3) reading special instructions if any are applicable. These responsibilities shall be carried out as soon after relief as plant conditions will allow. Each Shift Supervisor, Assistant Nuclear Shift Supervisor, Chief Nuclear Operator, Nuclear Operator, and Assistant Nuclear Operator will be responsible for reading the Operator's Log entries back to his last scheduled shift and so indicate by placing his initials in the space provided at the end of the shift immediately preceding his return to operational duties. Shift Supervisors and Assistant Nuclear Shift Supervisors shall also review the Jumper Log, Shift Supervisor's Log, Equipment Out-of-Service Log, and short-term instructions. The oncoming Shift Supervisor shall document his relief by completing Enclosure 9, Shift Relief Checklist, while dicussing operations with the on-duty Shift Supervisor. Immediately after turnover, the on-duty Shift Supervisor shall complete Enclosures 10 and 11 to assure operational and emergency systems' status.

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2.18 INSTRUMENT READINGS AND CONTROL INDICATIONS

Plant operations are conducted by shift operating personnel under the direction of the Shift Supervisor. They have the authority and responsibility to perform the operations necessary to limit plant operations or shut down the plant when such action is warranted by plant conditions, unusual circumstances, or unidentified events. Such actions may be warranted on the basis of instrument readings and/or control indications which are not consistent with expected plant conditions. When analyzing such situations, shift operating personnel must consider the instrument readings and correct indications to be true unless they are proven to be incorrect.

Operating personnel will not manipulate instrument, control, or alarm set points other than those available on the control console or those normally required during routine operations.

2.19 KEY CONTROL

Keys are required for the operation of certain switches, valves, switchgear, and for access to particular rooms and areas. The components or

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areas requiring key control are those with special operational importance or safety significance. A list identifying each key-controlled item and area and the respective key identification is maintained in the Control Center. All keys on this list are identified as "Controlled Keys". These keys are maintained in a locked cabinet under the control of the Shift Supervisor. Permission of the Shift Supervisor or Assistant Nuclear Shift Supervisor is required for the use of a "Controlled Key".

Before authorizing the use of a "Controlled Key", the Shift Supervisor must assure himself that the individual intending to use the equipment or enter an area under key control understands and appreciates the particular operational or safety requirements associated with the equipment or area. At the completion of the operation or task, the Shift Supervisor will assure himself that conditions are proper for placing the lock control in the proper position or locking a door or barrier and returning the key to the key control cabinet. The individual using the equipment or area under key control will be responsible for insuring that unauthorized personnel do not use the equipment or gain access to the area during the period when the control device is unlocked.

"Controlled Keys" are issued to certain "ON DUTY" operating classifications; possession of such keys is necessary to perform normal and emergency duties within their area of responsibility. The "Controlled Keys" remain on the operator's person during his assigned hours of duty and are not permitted to be used by unauthorized individuals.

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Entries are made in the Key Control Log, Form 912220 (Enclosure 4), which show the purpose for which a "Controlled Key" was required, who used it, badge number, and the time if was returned to the key control cabinet. When the Key Control Logsheet has been completely filled out and all keys listed on that sheet have been returned, the logsheet shall be sent to the Operations Superintendent for disposition.

2.20 CONTROL CENTER STATUS BOARD

The Control Center Status Board provides a visual display of information for immediate reference by Control Center personnel. This information includes the condition of engineered safeguards equipment, storage tank levels, and the chemical analysis of the coolant. The status board information is derived from the Equipment Out-of-Service Log and other plant

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records. The Shift Supervisor is responsible for insuring that the status board is updated as new information becomes available.

2.21 CONTROL CENTER REFERENCES

To assist shift operating personnel in the conduct of their duties, reference information related to any aspect of plant operation, safety, and administration is permitted in the Control Center. To meet these needs, the Operations Superintendent will insure that current copies of the reference information listed below are available in the Control Center.

a. Final Safety Analysis Report (FSAR)

b. Technical Specifications

c. System Flow Diagrams and One Line Electrical Diagrams

d. Working Copies of POQAM

Additional reference matter may be kept in the Control Center at the discretion of the Operations Superintendent.

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Typical of this category of reference documents are vendor technical manuals, training course notes, engineering handbooks, and technical texts. All such material shall be subject to approval by the Operations Superintendent.

2.22 NOTIFICATION

Many plant conditions and operating situations are of such a nature that it is necessary or prudent to promptly advise the Operations Superintendent or person on-call of the circumstances. The Shift Supervisor must utilize his judgement and experience in assessing the need for such notification which implicitly includes obtaining advice, assistance, and direction from the Operations Superintendent or person on-call. The following situations require prompt, verbal notification:

a. Reactor Trip
b. Inadvertent (radioactivity bearing) liquid or gaseous releases.
c. Major equipment failure or malfunction (includes all safeguards equipment).
d. Unexplained reactivity changes.
e. Loss of off-site power.
f. Employee injury or radiation overexposure.
g. Accidents occurring on plant property (except minor injury).
h. Events requiring reports within 24 hours to the NRC Region II Office. (Technical Specifications 6.6 and 6.9)
i. Turbine Trip
j. Load Restrictions

The Shift Supervisor shall note in his log when he notifies the Operations Superintendent or person on-call.

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2.23 HOUSEKEEPING

In the interest of safe and efficient operation, plant equipment and areas must be maintained in a clean and orderly manner. The responsibility for keeping the Control Center in this condition is that of the shift operating personnel. All dusting and cleaning of control consoles, instrument panels, and computer consoles, and the orderly storage of books, drawings, and records, will be performed by shift operating personnel.

In all areas outside the Control Center, the responsibility for insuring the Operations Section's work areas are maintained in a clean and orderly condition rests with the Shift Supervisor or, as in the case of fuel handling operations, the Refueling Supervisor.

2.24 EQUIPMENT OUT-OF-SERVICE LOG

The Equipment Out-of-Service Log shall be updated and maintained by Control Center personnel. Any safety-related equipment taken out of service shall be recorded in this log on Enclosure 7. The information provided shall include the nature of the problem, the date taken out of service, the date the equipment must be returned to service (if applicable), clearance number, Technical Specification reference (if applicable), and the date the equipment is returned to service. The Shift Supervisor shall ensure this log is being maintained and up-to-date. This log shall be reviewed at shift turnover as required by Section 2.17. When the Equipment Out-of-Service Log has been completely filled out and all listed equipment returned to service, the logsheet shall be transmitted to the Operations Superintendent for review/disposition.

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Engineered safeguards systems or equipment shall have independent verification of proper valve alignment per Step 6.1.9 of CP-115, In-Plant Equipment Clearance and Switching Orders, and applicable surveillance performed before returning equipment to service.

2.25 UNUSUAL OPERATING EVENTS REPORT

Unusual operating events include, but are not limited to, reactor trips, operating events involving significant unexpected behavior, or significant unusual operating situations such as changing RC pump combinations at power or inadvertent equipment operations. When an unusual operating event occurs, the Shift Supervisor takes the following action:

a. Insures that the plant is placed in a safe condition by having the necessary operations performed in accordance with approved procedures.

   NOTE: If the event is a reactor trip, refer to Section 2.14.

b. Notifies the Operations Superintendent or person on-call.

c. Determines subsequent action to be taken.

d. Insures the unusual operating event information is entered in the Reactor Operators Log and Shift Supervisors Log.

e. Completes Steps 1 thru 9 of Enclosure 8, Unusual Operating Event Summary, and forwards Enclosure 8, including the required information of Step 8, to the Operations Engineer.

The Operations Engineer will collate the Unusual Operating Event Summary and prepare an Unusual Operating Events Report. The report will be divided into four major sections: event synopsis, evaluation and

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recommendations, event details and data, and transient classification. The "Evaluation and Recommendations" section will be further subdivided to discuss expected plant performance, performance deviations, and recommended corrective action. The Unusual Operating Events Report will be routed through the Operations Superintendent and Results Engineer to the plant files (3-0-1-e) and will also be transmitted via AI-500, Conduct of Operations.

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3.0 NON-CONFORMING OPERATIONS

Non-conforming operations are defined and their identification and reporting discussed in CP-111, Procedure for Documenting the Reporting and Review of Non-Conforming Operations.

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4.0 WORK REQUESTS

Work Requests are used by plant personnel to report to the appropriate Maintenance Section equipment or system deficiencies. The Work Request, Form 912104, is described in detail in CP-113, Procedure for Handling Work Requests, Including Discrepancies and Corrective Actions.

4.1

If the Work Request is originated by the Operations Section, the operator originating the Work Request completes Part I and submits four copies to the Shift Supervisor on duty. The Shift Supervisor then determines the following:

a. Is the request of sufficient urgency to call out maintenance personnel? If so, notify the respective supervisor of the need for maintenance.

b. Will the performance of the work by maintenance personnel have an impact on plant safety and/or operations? If so, immediately notify the Responsible Supervisor of any precautions or limitations which need to be imposed.

c. Has work been requested before? If so, how long ago? If time seems too long, notify Operations Engineer, but void new Work Request.

4.1.1

The Shift Supervisor signs the Work Request in the "Responsible Supervisor" blank of Part I. However, if the Responsible Supervisor is indeed the originator, then the "Responsible Supervisor" blank should be marked "N/A" (CP-113, Procedure for Handling Work Requests, Including Discrepan-

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cies and Corrective Actions; Step 4.1.4). He then files copy "D" (gold) in the Control Center Work Request File. The remaining copies are sent to the Planning Coordinator (Maintenance) or Work Supervisor (documentation) for review and distribution per CP-113, Procedure for Handling Work Requests, Including Discrepancies and Corrective Actions. If Operations personnel are to accomplish the work activity, the Shift Supervisor shall complete Part II of the Work Request as required by Section 4.2 of CP-113, Procedure for Handling Work Requests, Including Discrepancies and Corrective Actions.

NOTE: During off-hours, the evaluation of Part II of the Work Request shall be completed by the Shift Supervisor for various departments as priorities dictate. Refer to Section 4.1 above and Section 4.2 of CP-113, Procedure for Handling Work Requests, Including Discrepancies and Corrective Actions.

4.1.2

When the Quality Control Inspector completes his review of the Work Request (Steps 4.6.2 thru 4.6.4 of CP-113, Procedure for Handling Work Requests, Including Discrepancies and Corrective Actions), the completed copy "C" (pink) is returned to the Shift Supervisor. The Shift Supervisor will remove copy "D" (gold) from the Outstanding Status File, destroy it, and route copy "C" (pink) back to the originator for information.

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5.0 RADIATION WORK PERMITS

Radiation Work Permits (RWP's) are used to authorize specific individuals to enter areas to work on equipment where the possibility of radiation exposure or contamination exists.

After the RWP has been initiated by the department responsible for the work to be performed and approved by the ChemRad Section, the three forms are taken to the Control Center where the Shift Supervisor determines if the work can be performed without interfering with plant operations, makes arrangements for any clearances needed, and notes any special conditions or restrictions required by the Operations Section on the RWP.

When the clearances are complete, the Shift Supervisor signs the RWP and work can begin. The Control Center copy of the RWP is posted in the Control Center. The ChemRad Section will notify the Control Center when the RWP is closed out and the Control Center copy shall be removed from the Control Center and sent to the Health Physics Supervisor.

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6.0 EQUIPMENT OUTAGES

When necessary the Operations Section will recommend and schedule the plant operating configuration required for equipment repair. This is the responsibility of the Operating Engineer who is the Staff Engineer reporting to the Operations Superintendent. In addition to his function as a technical advisor, the Operating Engineer will coordinate the functions of Operations, Technical Services, Maintenance and outside contractors during scheduled outage periods.

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ENCLOSURES

Enclosure 1	Form 912235, Change Notification
Enclosure 2	Form 912225, Supplemental Laboratory Analysis Request Form
Enclosure 3	Form 912212, Shutdown Report
Enclosure 4	Form 912220, Key Control Log
Enclosure 5	Form 912226, Short-Term Instruction
Enclosure 6	Form 912227, Short-Term Instructions Index
Enclosure 7	Equipment Out-of-Service Log
Enclosure 8	Unusual Operating Event Summary
Enclosure 9	Shift Relief Checklist
Enclosure 10	Operational Status Checklist
Enclosure 11	Critical Plant Equipment/Parameters Checklist

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CHANGE NOTIFICATION

SAMPLE

SUPERVISORS:

The attached is an approved change. After noting and informing your people, so indicate by dating and initialing below:

MAR NO./TITLE ______________________________

PROCEDURE NO./TITLE ______________________ REV. ______

PERMANENT __________ PAGES AFFECTED ______________

Details: ______________________________

Shift Supervisor	Initials/Date	Asst. Nuclear Shift Supv.	Initials/Date

Operations Engineer ____________

Operations Superintendent ____________

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SUPPLEMENTAL LABORATORY ANALYSIS REQUEST FORM

SECTION 1	REQUEST
SYSTEM / SAMPLE ORIGIN: ______	
REASON FOR REQUEST: ______	
ANALYSIS REQUESTED: ☐ BORON ☐ OTHER, AND SPECIFY; ______	
SPECIAL INSTRUCTIONS: ______	
ORIGINATOR: DATE: TIME:	SHIFT SUPERVISOR: DATE: TIME:

SECTION 2	RESULTS			
SAMPLE	DATE	TIME	INIT.	PARAMETER
OBSERVATIONS (UNITS)	DATE	TIME	INIT.	RESULTS (UNITS)
SAMPLE	DATE	TIME	INIT.	PARAMETER
OBSERVATIONS (UNITS)	DATE	TIME	INIT.	RESULTS (UNITS)
SAMPLE	DATE	TIME	INIT.	PARAMETER
OBSERVATIONS (UNITS)	DATE	TIME	INIT.	RESULTS (UNITS)

REMARKS:

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SECTION 3	ACCEPTANCE *
RESULTS IN OPERATORS LOG: INIT.	
STATUS BOARD UPDATED: INIT.	
* ATTACH TO PROCEDURE APPROVAL AND TRANSMITTAL SHEET AND ROUTE FOR STORAGE UNDER AI-500	

912225

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3

SHUTDOWN REPORT NO. ________

1. TIME AND DATE OF SHUTDOWN ________________________________

2. TYPE OF SHUTDOWN: Forced __________ Scheduled __________

NOTE: Forced shutdowns are those which must be initiated no later than the weekend following discovery of an abnormal system condition.

3. METHOD OF SHUTDOWN: Manual Runback __________ Manual Trip __________

Automatic Trip __________ Other ______________
(Explain in Step 9.)

4. POWER PRIOR TO SHUTDOWN __________%

5. PLANT STATUS AFTER SHUTDOWN: Mode _______ Power _______% _______ MWe

6. APPARENT CAUSE OF SHUTDOWN ________________________________

____

____

7. DURATION OF OUTAGE ________________________________

____

8. CORRECTIVE ACTION ________________________________

____

____

9. REMARKS ________________________________

____

____

____
Shift Supervisor

____
Operations Superintendent

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____
Results Engineer
(for transient cycles)

ENCLOSURE 4

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3

KEY CONTROL LOG

SAMPLE

KEY NO.	DATE/TIME ISSUED	ISSUED TO	BADGE NO.	PURPOSE FOR WHICH KEY IS REQUIRED	DATE/TIME RETURNED

912 220

ENCLOSURE 5

SHORT-TERM INSTRUCTION SAMPLE

Document No. ________ Date ____________

Length of Effect: From ______________ to ______________

Instruction: ____

____

____

____

____

____

Action: ____

____

____

____

____

____

Issued By ____________________

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SHORT-TERM INSTRUCTIONS INDEX SAMPLE

DOCUMENT NO.	SUBJECT	DATE ENTERED	DATE REMOVED	REMOVED BY

912 227

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Equipment Component	Problem/Malfunction	When OOS Date / Time	Req'd Back in Service Date / Time	Clearance / W.R.#	STS or Other Equipment Action #	Ret'd To Service Date / Time

ENCLOSURE 7

UNUSUAL OPERATING EVENT SUMMARY

1. TIME AND DATE OF EVENT: ______________________________

2. TYPE OF EVENT (i.e., reactor trip, significant unexpected behavior during a planned evolution or transient, etc.):

____

3. PLANT STATUS PRIOR TO EVENT: Mode _____ Power _____% _____ MWe

4. PLANT STATUS AFTER EVENT: Mode _____ Power _____% _____ MWe

5. APPARENT CAUSE OF EVENT: (Use attachments as required.)

____

____

____

6. DURATION OF EVENT (i.e., time in degraded condition, approximate time from event to discovery, etc.):

____

____

7. CORRECTIVE ACTION: (Use attachments as required.)

____

____

____

8. REQUIRED INFORMATION CHECKLIST:

   NOTE: This information should be as complete as practical and should include data from a minimum of 1 hr. prior to the event.

   a. Operating Strip Charts: __________

      [OTSG Levels, T(ave), Pressurizer Level, FW Flows, WR/NR, RC Pressure, Header Pressure, Reactor Power, Thot, MWe]

      Denote point of event on each strip chart and assure time, date, and recorder number are on each strip chart.

   b. Computer Alarms Summary Printout: __________

   c. Annunciator Events Printout: __________

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UNUSUAL OPERATING EVENT SUMMARY
(Cont'd)

8. REQUIRED INFORMATION CHECKLIST:

d. Post-Trip Review Summary: (Reactor Trip Only) ____________

e. Operators Log: ____________

(Include at least one shift prior to event.)

f. Shift Supervisors Log: ____________

(Include at least one shift prior to event.)

g. Event Summaries: ____________

NOTE: Each operation classification that observed or took part in the event shall summarize the sequence prior to, during, and after the unusual event to the best of his recollection. Each person should prepare his own reconstruction of the event in detail. If plant conditions permit, this summary should be written immediately after the event, otherwise it shall be written after shift turnover and prior to leaving the plant site.

9. REMARKS: ____

____

____

____

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SHIFT RELIEF CHECKLIST

Information obtained from off-going shift:

a. Plant Status

Mode ______

NI Power Level ______%

MWe ______

MWth ______ (Group 21)

b. Operations in Progress (specify)

____

____

____

c. Special Instructions (explain)

____

____

____

d. Equipment in Degraded Mode

____

____

____

e. "Action Statement" in Effect and Total Time Interval in "Action Statement"

NOTE: Compare with allowed time interval.

____

____

____

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SHIFT RELIEF CHECKLIST
(Continued)

f. Equipment Under Maintenance or Test (specify)

____

____

____

g. Comments

____

____

____

Time/Date/Signature

Off-Going Shift ______/______/______________

On-Going Shift ______/______/______________

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OPERATIONAL STATUS CHECKLIST

	Check (✓)
a. Master Surveillance Plan	
b. Jumper Log	
c. Equipment Out-of-Service Log	
d. Short-Term Instructions	
e. Shift Supervisors Log	
f. Operators Log	
g. Plant Status Board	
h. Annunciator Alarms	

Time/Date/Signature

/ /

NOTE: The "check" on each item above signifies completion of review of that item.

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CRITICAL PLANT EQUIPMENT/PARAMETERS CHECKLIST

A. The Shift Supervisor shall assure by direct observation of controls or indication that the following equipment/parameters are aligned as specified.

NOTE: A check (✓) signifies status as specified.
To be completed at the beginning of each shift.

1. Radiation Monitoring Panel

Check for abnormally high readings, alarms of bypassed channels. ____ ____ ____

2. ES Channel "A"

"Channel Function Enabled" lights lit. ____ ____ ____
"Bypass/Reset" lights lit. ____ ____ ____
No Tripped Channels ____ ____ ____

All ES equipment in normal standby status.

DHP-1A	____	____	____	SWP-1A	____	____	____
RWP-3A	____	____	____	MUP-1B running	____	____	____
DCP-1A	____	____	____	AHF-1A	____	____	____
MUP-1A	____	____	____	AHF-15A	____	____	____
RWP-2A	____	____	____	BSP-1A	____	____	____
AHF-1C	____	____	____				

RB sump pumps in normal standby. ____ ____ ____

All valves on ECCS panel properly aligned with power available. ____ ____ ____

3. ES Channel "B"

"Channel Function Enabled" lights lit. ____ ____ ____
"Bypass/Reset" lights lit. ____ ____ ____
No Tripped Channels ____ ____ ____

All ES equipment in standby status.

DHP-1B	____	____	____	AHF-1C	____	____	____
RWP-3B	____	____	____	AHF-15B	____	____	____
DCP-1B	____	____	____	BSP-1B	____	____	____
MUP-1C	____	____	____				
RWP-2B	____	____	____				
SWP-1B	____	____	____				

All valves on ECCS panel properly aligned with power available. ____ ____ ____

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CRITICAL PLANT EQUIPMENT/PARAMETERS CHECKLIST
(Continued)

4. PSA Panel

EFP-1 in normal standby.
ASV-5 in "Auto".
MSIV's air supply test switches in "Normal".
MS line rupture matrix status lights normal.
MU&P loop proper lineup.

No abnormal unexplained trends on MUT level recorder.

5. ICS Panel

All ICS stations in "Auto".
Pressurizer heaters and spray valve in "Auto".
MUV-31 in "Auto".
No abnormal trends on strip chart recorders.
Recorders are at proper time.

Feedwater valves in "Auto":

FWV-31
FWV-30
FWV-36
FWV-32
FWV-29
FWV-33
FWV-14
FWV-15
FWV-28

All four pump recirc. valves.
No air failures lit.

6. TG Panel

Turbine backup lube oil and EH pumps in normal standby.

Turbine in ICS control.
All turbine drains closed.
H_2 Pressure Normal

7. Electrical Distribution

Voltage on 230 kV line normal.

If 4160V or 6900V unit buses are being fed from auxiliary transformer:

Verify "Auto Transfer" switches in "Auto":

"A"
"B"

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CRITICAL PLANT EQUIPMENT/PARAMETERS CHECKLIST
(Continued)

7. Electrical Distribution (Cont'd)

ES buses being fed from SU transformer: "A" ____ ____ ____
"B" ____ ____ ____

Emergency diesel generator start circuit lights lit:

"A" Diesel Generator - "A" ____ ____ ____
"B" ____ ____ ____

"B" Diesel Generator - "A" ____ ____ ____
"B" ____ ____ ____

Emergency diesel generator high speed light on.

"A" ____ ____ ____
"B" ____ ____ ____

Emergency diesel generator start mode and voltage adjust in "Auto":

"A" ____ ____ ____
"B" ____ ____ ____

Emergency diesel generator breakers' position indication light on and target matched. ____ ____ ____

The following bus voltages are normal:

4160V ES "A" ____ ____ ____
480V ES "A" ____ ____ ____
480V ES "AB" ____ ____ ____
4160V ES "B" ____ ____ ____
480V ES "B" ____ ____ ____

8. HVAC Panel

The following recorders are reading normal, inking properly, and are at proper time:

AH-35-FR ____ ____ ____
AH-717-FR ____ ____ ____
AH-294-FR ____ ____ ____
AH-32-FR ____ ____ ____

No unexplained target mismatch on any ventilation equipment. ____ ____ ____

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CRITICAL PLANT EQUIPMENT/PARAMETERS CHECKLIST
(Continued)

B. If any equipment/parameter is found contrary to that specified in the checklist:

1. Describe immediate action taken to correct the condition. (and)

2. Explain actions taken, instructions given, or recommendations to prevent recurrence.

____

____

____

____

	Time/Date/Shift Supervisor Signature
0000-0800	/ /
0800-1600	/ /
1600-2400	/ /

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ATTACHMENT VI

Florida Power CORPORATION

INTEROFFICE CORRESPONDENCE

Production (OFFICE) C-4 (MAIL CODE)

SUBJECT: Management Responsibility of Nuclear Shift Supervisor

TO: Nuclear Plant Personnel

DATE: December 3, 1979
Alpha 19

The purpose of this directive is to emphasize the primary management responsibility of the Shift Supervisor and clearly establish his command duties.

The primary management responsibility of the Nuclear Shift Supervisor is to provide direct command oversight of plant operations and perform a management review of ongoing operations, maintenance, and support functions important to safety (i.e., to maintain an overview of the situation, to make decisions, and to direct operations). During back shifts and weekends, he is the Senior Management Representative on site and all personnel on site report to him.

The Nuclear Shift Supervisor's command duties require that he be on duty in the Control Room or Shift Supervisor's office and that he maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority.

During accident situations, the Shift Supervisor shall remain in the Control Room to direct the activities of Nuclear Operators until properly relieved by another Shift Supervisor or Assistant Shift Supervisor.

If the Shift Supervisor is temporarily absent from the Control Room during routine operations, the Assistant Shift Supervisor shall assume the Control Room command function with all the responsibilities and authority of the Shift Supervisor.

When functioning as Temporary Emergency Coordinator, the Shift Supervisor has full authority to evaluate and classify the emergency and initiate appropriate actions to mitigate the consequences.

Should his evaluation indicate that extreme measures must be taken, he has the authority to direct any or all personnel to evacuate the plant site, to place any or all generating plants in a safe shutdown condition, and to notify all applicable agencies of the plant status or required outside assistance.

The Nuclear Shift Supervisor is fully supported by corporate and plant management in carrying out the above responsibilities.

G. C. Moore

G. C. Moore
Assistant Vice President
Power Production

xc: NRC Region 1
Office of Inspection
and Enforcement

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ADMINISTRATIVE INSTRUCTIONS

AI-200

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

ORGANIZATION AND RESPONSIBILITY

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REVIEWED BY: Plant Review Committee

Date 8/9/79

Meeting No. 79-32

APPROVED BY: Nuclear Plant Manager

Date 9/21/79

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1.0 INTRODUCTION

The purpose of this section of the Plant Operating Quality Assurance Manual (POQAM) is to:

a. Describe the organization and responsibilities of the sections and individuals responsible for all aspects of the operational Quality Assurance effort.

b. Delineate the qualifications required for positions in CR-3.

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2.0 ORGANIZATION

The organization as related to operational Quality Assurance is as shown on Technical Specifications Figures 6.2-2, Nuclear Plant Organization, and 6.2-1, Offsite Organization Chart.

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3.0 NUCLEAR GENERAL REVIEW COMMITTEE

The Senior Vice President-Engineering and Construction shall appoint a Nuclear General Review Committee having the responsibility for verifying that the operation of the plant is consistent with company policies, rules, approved procedures, and license provisions. This Committee meets the requirements of ANSI N18.7-1976, "Administrative Controls for Nuclear Power Plants", Section 4.

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4.0 PRODUCTION DEPARTMENT

The Director-Power Production is responsible to the Senior Vice President-Engineering and Construction and is given overall authority to staff, operate, and maintain the Nuclear Plant. (Refer to Technical Specifications Figure 6.2-1, Offsite Organization). The Director-Power Production is informed of significant problems or occurrences which relate to safety or Quality Assurance through established administrative procedures.

4.1 SYSTEM PLANT OPERATIONS

System plant operations are under the supervision of the Nuclear Plant Manager and the Manager-Fossil Operations. The Nuclear Plant Manager reports to the Director-Power Production. In addition to the expertise in the Nuclear Plant, there is available at the discretion of the Director-Power Production assistance from other plants with people thoroughly trained in power plant operations, maintenance, and chemistry, as well as the Production staff. Refer to Technical Specifications Figure 6.2-1, Offsite Organization.

4.2 PRODUCTION DEPARTMENT STAFF

The Production Department is divided into various sections as shown on Technical Specification Figure 6.2-1, Offsite Organization. These sections are staffed with personnel experienced in specialized plant operating and maintenance problems, and provide technical support to the Nuclear Plant staff.

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5.0 PLANT ORGANIZATION

The plant organization is shown on Technical Specification Figure 6.2-2, Facility Organization. The Nuclear Plant Manager is directly responsible for the safe operation of the Nuclear Plant. In all matters pertaining to actual operation and maintenance, and to Technical Specifications, the Nuclear Plant Manager shall report to and be directly responsible to the Director-Power Production. The Nuclear Plant Manager is responsible for the activities of the following:

5.1 OPERATIONS SUPERINTENDENT

The Operations Superintendent is responsible to the Nuclear Plant Manager for the operation of the plant. Plant operations are performed by the Operations Section under the general supervision of the Operations Superintendent. Under his supervision the Operations Section shall conduct nuclear plant operations to ensure both short and long range safe, efficient, and timely production of electric power consistent with company and department policies and requirements and with governmental regulations. He shall coordinate overall plant operating plans and schedules with system requirements and coordinate unit analyses to minimize equipment downtime. The responsibilities of the Operations Superintendent include the following:

a. Develop and implement procedures, operating instructions, and emergency procedures required for plant operation.

b. Support the development of and implement uniform policies, procedures, operating instructions, and emergency plans developed by others for plant operation.

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c. Provide lead responsibility for the development and implementation of new unit pre-operational and start-up procedures.

d. Provide the overall plant planning, operational scheduling, and coordination required to ensure minimum equipment downtime during unit outages.

e. Coordinate overall plant operating plans and schedules with the system dispatching group.

f. Perform continuing reviews and appraisals to ensure the safe and efficient condition of plant equipment.

g. Originate and approve work requests for the maintenance of operating equipment.

h. Perform operator maintenance.

i. Prepare and coordinate need statements for initiating betterment projects for the plant.

j. Implement requirements of the quality assurance program as related to plant operations.

k. Implement requirements of plant personnel safety programs.

l. Develop and coordinate recommended requirements for training programs related to plant operations with the plant administrative services staff, and support the implementation of operator training and relicensing program.

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m. Provide operations input to plant manpower, expense, and capital budgets.

n. As required, provide company representation to the local community and represent the needs and interests of the local community to the plant and the Power Production Department.

o. Represent Florida Power Corporation on and participate in the activities of appropriate industry-related committees, professional societies, and codes and standards groups.

p. Assuring that plant operations are conducted in accordance with the requirements of the Operating License and the procedures of POQAM.

q. (deleted)

r. Evaluating Radiation Work Permits (RWP's) as required by RP-101, Radiation Protection Manual.

s. Scheduling and reviewing the results of tests, calibrations and inspections required by SP-300 and 400 series Surveillance Procedures, and determining the status and operability of certain safety-related equipment and systems.

t. Insuring through proper scheduling that each shift is properly manned for the expected operational activities. Shift manning

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schedules must include compliance with Technical Specification requirements (Table 6.2-1) and the dictates of good operation practice.

u. Evaluating shift reports of equipment malfunctions or unusual system behavior to initiate corrective action in the form of maintenance work requests, operational performance tests, special data printout or recording, plant shutdown, or load reduction.

v. Coordinating alarm tests, fire drills, and evacuation exercises as required by Volume I, Administrative Instructions, and Volume III, Emergency Plan Implementing Procedures, of POQAM. Scheduling the training and retraining of the shift personnel. Evaluate the processing of radioactive waste to insure that processing, storing, and disposal are being conducted in accordance with established procedures and specific instructions contained in the appropriate Waste Release Permit.

w. Reviewing and approving the conditions and procedures provided for the performance of operational hydrostatic tests of fluid systems required by a continuing Quality Assurance Program.

x. Participate in Plant Review Committee.

y. Promote safe work practices in all phases of the operation of the Nuclear Plant.

z. Participate in Plant Safety Committee. 1755 149

The Operations Superintendent will be in charge of refueling operations under the technical direction of the Performance Engineering Supervisor.

The Operations Superintendent receives and reviews the reports of unit trips, load restrictions, abnormal occurrences, inadvertent liquid or gaseous activity releases, major equipment failure or malfunction (all safeguards), unexplained reactivity changes, loss of off-site power, and deviations from established procedure. He reviews these reports and initiates corrective action and/or additional reports as required. In addition to prescribing and following up remedial action for the foregoing situations, he reports them to the Technical Services Superintendent.

The Operations Superintendent prescribes and monitors the training of candidates for shift positions. He also prescribes and monitors the refresher training of qualified shift personnel.

The Operations Superintendent reviews and approves the rod worth and boron worth curves developed by the Performance Engineering Supervisor for use by Shift Operations personnel.

5.2 OPERATING ENGINEER

The Operating Engineer reports to the Operations Superintendent and has a dual function. He provides technical guidance for Nuclear Plant operations, including coordination of operating plans and schedules with system requirements. He also plans and coordinates unit outages. This latter function is a major responsibility.

In coordinating unit outages, the Operating Engineer functions in a manner similar to that of a Project Engineer. He coordinates the functions of Operations, ChemRad, Technical Support, Outage Planning, Maintenance, and outside consultants.

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5.3 MAINTENANCE SUPERINTENDENT

The Maintenance Superintendent is responsible to the Nuclear Plant Manager for plant mechanical, instrument, and electrical maintenance. This includes the maintenance of Nuclear Plant equipment, instrumentation, and facilities to ensure both short and long-range safe, efficient, and timely production of electric power consistent with company and department policies and requirements and with governmental regulations. Determine and specify an adequate plant inventory of spare parts, maintenance tools, and operating supplies. Incorporate approved minor plant modifications and additions and support the implementation of plant betterment and other special projects. The responsibilities of the Maintenance Superintendent include the following functions.

a. Develop and implement unique procedures and methods required for plant maintenance.

b. Support the development of and implement uniform policies, procedures, and methods approved for plant maintenance.

c. Support the Power Production Department Maintenance Services Group in the development of a system-wide preventative maintenance program and develop and implement the necessary procedures and schedules involved for the Nuclear Plant.

d. Plan and schedule plant maintenance activities and determine and coordinate the use of outside maintenance services.

e. Initiate, support, and coordinate the development of engineering solutions to maintenance problems with the technical services and other functions.

f. Supervision of the organization, planning and scheduling of all plant instrument, electrical, and mechanical maintenance.

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g. Coordination of the electrical, instrument, and mechanical activities of all departments within (and without) the plant in the total maintenance effort.

h. Organize and conduct the Preventative Maintenance Program on all equipment within the scope of the Maintenance Section's responsibility.

i. Assure that all maintenance activities meet or exceed all applicable codes, specifications, standards, FSAR, and Technical Specifications. The requirements include documentation of activities in the performance of modifications, repairs, or replacement of related components and systems.

j. Responsible for safety in all phases of the maintenance effort. Participate in the Plant Review Committee.

k. Monitor the inventory of spare parts and maintenance supplies to insure minimum investments consistent with reliable maintenance and quality control requirements.

l. Compare and update maintenance budget forecasts.

m. Review related codes, specifications, and standards for latest revisions or addenda applicable to the operating plant.

n. Planning scheduled outages and preparing unit outage reports.

o. Promote safe work practices.

p. Participate in Plant Safety Committee.

q. Organize, develop, and conduct training programs for plant maintenance personnel.

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r. Organize and direct the maintenance effort during activities related to initial and subsequent refuelings.

s. Responsible for applicable Maintenance, Refueling, and Surveillance Procedures.

5.4 TECHNICAL SERVICES SUPERINTENDENT

The Technical Services Superintendent is responsible to the Nuclear Plant Manager for planning, scheduling and supervising the activities unique to nuclear plant operations to ensure safe, efficient, economical and timely production of electric power consistent with company and department policies, and objectives and governmental regulations. In addition he is responsible for plant support services related to plant performance analysis, environment, and engineering to help meet overall plant production, availability, economics, and efficiency objectives. The Technical Services Superintendent is also responsible for planning for plant emergencies relating to public health and safety, and damage control as well as plant improvement studies. The following functions are included in the Technical Services Superintendent responsibilities: health physics, power plant chemistry, compliance engineering, performance testing, environmental data acquisition and analysis, systems analysis, engineering support services, plant improvements, Regulatory Agency liaison, plant accident and other emergency planning and plant records maintenance.

5.5 NUCLEAR QA/QC COMPLIANCE MANAGER

The NQA/QC Compliance Manager shall be responsible to the Nuclear Plant Manager to provide assurance that operation and maintenance of the plant

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is performed in compliance with POQAM, Technical Specifications, and applicable portions of the Code of Federal Regulations. The responsibilities of the NQA/QC Compliance Manager include the following:

a. Maintaining an audit system to verify the existence and location of all documentation required by the above documents.

b. Examining the content and acceptance criteria of each required document.

c. Reviewing each plant-generated Purchase Requisition to verify the safety/non-safety related classification, the Quality and documentation requirements, and to assure that the requirements for design documents have been specified. Those Purchase Requisitions that require a determination of Quality requirements or additional Quality requirements will be forwarded to the Director of Production Engineering in accordance with the provisions of CP-101, Procurement of Material, Equipment, and Services.

d. Directing and approving the preparation, review, and implementation of Quality Control procedures and NQA/QC inspection activities.

e. Reviewing and approving Quality Control procedures and activities necessary for any plant modification.

f. Promoting safe work practices and participating in Plant Safety Committee.

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The NQA/QC Compliance Manager is responsible for accompanying and assisting NRC Directorate of Regulatory Operations Inspectors during routine and special plant inspections as directed by the Technical Services Superintendent.

The NQA/QC Compliance Manager will prepare reports, for use by the Nuclear Plant Manager, summarizing the results of his audits and inspections. Where applicable, he will include suggestions for upgrading the Quality Program or its implementation.

The NQA/QC Compliance Manager will assist the Nuclear General Review Committee with their audits of the plant's operational Quality Assurance program.

The NQA/QC Compliance Manager will work with the plant staff in contract negotiations, cost analysis, and planning of required primary system inservice inspections.

The NQA/QC Compliance Manager will observe performance of the Maintenance, Operations, Performance, ChemRad, and Technical Support Departments' field projects to spot check conformance with established procedures and regulations.

The NQA/QC Compliance Manager will assist the Section Engineers in the areas of planning scheduled outages, post-maintenance testing, and vendor shop inspections.

5.6 CHEMISTRY AND RADIATION PROTECTION ENGINEER

The Chemistry and Radiation Protection (ChemRad) Engineer is responsible to the Technical Services Superintendent for matters pertaining to radiation

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protection of all plant site personnel and for water chemistry of all systems within the confines of the Crystal River Unit 3 boundary. These services are performed by the ChemRad Section, which is comprised of the ChemRad Protection Engineer, ChemRad Plant Engineer, Rad Waste Supervisor, Health Physics Supervisor, Assistant ChemRad Protection Engineer, and ChemRad Technicians. The ChemRad Protection Engineer shall be responsible for and have administrative control of the following:

a. Issuance of Standing Radiation Work Permits to allow routine work to be done in Radiation Controlled Areas (RCA's).

b. Performing radiological assessments and approving RWP's for non-routine work to be done in RCA's.

c. Performing routine and special radiation surveys throughout the plant to assure all direct radiation, contamination, and airborne activity areas are properly posted and/or under control, and in compliance with RWP's.

d. Specifying corrective actic or protective measures to be adopted by plant personnel in order to minimize radiation exposures to personnel.

e. Assisting in the planning of special operating or maintenance work to be done to minimize radiation exposures to personnel doing the work, and to minimize the amount of airborne activity released to the environment.

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f. Maintaining a personnel dosimetry program, such as TLD's, in accordance with 10 CFR 20, on all personnel, and to maintain up-to-date records of all personnel included on the dosimetry program.

g. Training of all personnel working at the plant in Radiological Control Procedures (RP-100 Series of Volume VIII of POQAM).

h. Approving and stipulating the conditions for the release of airborne radioactive materials from Unit 3, and maintaining an up-to-date log on all intermittent and continuous releases in accordance with applicable State and Federal regulations, and Technical Specifications.

i. Approving and stipulating the conditions for the release of all radioactive liquids from Unit 3, and maintaining an up-to-date log of all releases in accordance with applicable State and Federal regulations, and Technical Specifications.

j. Maintaining a record of all radioactive materials shipped from the plant for disposal, and assuring that all shipments are in conformance with applicable NRC, DOT, and State regulations.

k. Sampling and analyzing reactor coolant, auxiliary, and secondary plant water systems on a routine or special basis for chemical analysis, and assuring that all are within the limits of Technical Specifications or other administrative limits. On the basis of results, take corrective action to bring analyses within specified limits.

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l. Providing special analysis, such as boron concentration, as requested by the Operations Section or others.

m. Providing Emergency Plan training to plant personnel, scheduling training drills with outside agencies, and evaluating the results of drills as defined in the Emergency Plan.

n. Maintaining an on-site radiological surveillance program (TLD's and air sampling) to assure dose rates and airborne concentrations are below the 10 CFR 20 criteria for "Unrestricted Areas".

o. Maintaining an adequate supply of radiation protection instrumentation and protective clothing, as well as respiratory and decontamination supplies for use by all plant personnel.

p. Responsible for providing qualified personnel to monitor and advise work personnel doing operations or maintenance work functions, to assure compliance with work permits and/or administrative or regulatory requirements, and to take necessary action when requirements are in non-compliance.

q. Responsible for maintaining equipment in the First Aid Room.

r. Promote safe work practices in all phases of ChemRad area and participate in Plant Safety Committee.

s. Participate in Plant Review Committee.

The ChemRad Protection Engineer is a member of the Emergency Plan Organization. He may fulfill the duties of "Radiation Team Leader" and act as

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temporary "Emergency Coordinator" after receipt of a Reactor Operator or Senior Reactor Operator's License. He also participates as a Plant Review Committee member.

The ChemRad Engineer will evaluate and act upon requests authorizing increases in administrative exposure limits as prescribed by RP-101, Radiation Protection Manual.

5.7 PERFORMANCE ENGINEERING SUPERVISOR

The Performance Engineering Supervisor is responsible to the Technical Services Superintendent for plant thermal and nuclear performance and maintenance software involving the plant computer. He is responsible for the technical direction of refueling operations and for development of core characteristics, rod worth curves, and boron worth curves for use by Operations personnel after review and approval by the Operations Superintendent. His responsibilities also include any other technical services as required in support of operations and in compliance with safety requirements. These services are performed by the Performance Engineering Section which is comprised of the Results Engineer, the Computer and Controls Engineer and the Reactor Engineer. The Performance Engineering Supervisor's responsibilities include the following:

a. Analyze daily logs for evaluation of plant performance and equipment availability.

b. Supervise and direct work of the Results Engineer, the Computer and Controls Engineer and the Reactor Engineer.

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c. Direct activities in emergency support of operations and coordinate activities with other supervisors.

d. Direct the Results Engineer in the evaluation of monthly plant performance and cost analysis reports.

e. Participate in the Plant Review Committee.

f. Calculate monthly budget and update yearly forecast.

g. Promote safe work practices in all phases of the performance engineering effort and participate in Plant Safety Committee.

h. Direct Results Engineer in writing and maintaining Nuclear Plant Reliability Data (NPRD) Reports.

i. Evaluate performance of Performance Engineering personnel to insure optimum operating efficiency.

j. Direct cost analysis studies to determine the effect of plant performance on production costs and make recommendations for corrections to any operating deficiencies.

k. Organize, develop, and direct training programs for Performance Engineering personnel.

l. Provide information to update training manuals and participate in plant training activities.

m. Work closely with other supervisors in conducting fuel loading and unloading operations, especially in the areas of core nuclear instrumentation.

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n. Responsible for such research and development projects as may be assigned with a view toward measuring their effect on plant efficiency.

o. Maintaining complete core history and fuel inventories.

p. Developing operational follow-up techniques to ensure adequate capability to recover from off-design conditions.

5.8 TECHNICAL SUPPORT ENGINEER

The Technical Support Engineer is responsible to the Technical Services Superintendent and shall provide technical services as required in support of operations in compliance with safety requirements. The Technical Support Section is comprised of the Inservice Inspection Engineer, the Nuclear Technical Specifications Coordinator, and Plant Engineers. The specific responsibilities of the Technical Support Engineer include the following:

a. Provide central engineering support for total plant operations in compliance with applicable codes, Regulatory Guides, FSAR, and Technical Specification requirements.

b. Direct activities of the Inservice Inspection Engineer in planning, scheduling, and performance of the plant's Inservice Inspection Program.

c. Direct activities of the Nuclear Technical Specifications Coordinator in maintaining program which controls plant's compliance with Technical Specifications and FSAR.

d. Assuring all Nonconforming Operation Reports (NCOR's), Modification Approval Records (MAR's), and procedure changes comply with Technical Specifications, Regulatory Guides, and FSAR safety considerations.

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e. Participate on Plant Review Committee (PRC).

f. Administrate Plant Fire Protection Program and participate in same as Fire Brigade Chief.

g. Organize, develop, and direct training programs for Technical Support personnel.

h. Control monthly budget and update yearly forecasts.

i. Promote safe work practices in all phases of the Technical Support effort and participate in Plant Safety Committee.

j. Evaluate and order additional test equipment, spare parts, or plant equipment as needed, assuring the Quality Program requirements are maintained.

k. Schedule and conduct the Technical Support Section's outage activities and surveillance requirements as required.

l. Complete special EEI, EPRI, and monthly operating reports to maintain various operating permits.

m. Assure compliance with retraining requirements to maintain Senior Reactor Operator's License.

5.9 ADMINISTRATIVE SUPERVISOR

The Administrative Supervisor is responsible to the Plant Manager for site security, training, clerical and accounting functions, drawing and document control, and building services. These functions are performed by four groups, that is, security guard force, clerical operations, training

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and building services. These four groups are under the general supervision of the Office Manager, the Officer of the Guard, Training Supervisor, and Building Services Supervisor. These individuals are responsible to the Administrative Supervisor for each of their respective areas. The responsibilities of the Administrative Supervisor include the following:

a. <u>Plant Administration</u>

The Office Manager is responsible to the Administrative Supervisor to:

- Formulate, initiate, clear, and administer policy and procedures to assure availability, accuracy, and completeness of Quality Assurance records, plant maintenance records, health physics records, and all other records as required by commitment in the FSAR, ANSI N18.7-1976, or Regulatory Guides.
- Assist in the preparation of reports and permits. (This includes company reports, OSHA reports, NRC reports and permits related to operator licenses, by-products, and special nuclear materials.)
- Coordinate and assist in the preparation of plant operating, construction, and Responsibility Reporting Budgets.
- Coordinate, direct, and administer the clerical services required for recording, transcribing, typing, photocopying, and processing mail.
- Administer the clerical functions and liaison necessary to intra-company relations, for example, the Personnel Department, Payroll Department, Production Department, Controllers Department, Purchasing Department, etc.

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- Plan, direct, and arrange plant tours and other public relations functions of the plant.
- Coordinate training facilities, equipment, and requirements.
- Act as "Emergency Liaison Officer" during emergencies such as fire, bodily injury, radiation and contamination accidents, natural disasters, and a reactor accident.
- Arrange timely renewal of operator licenses.
- Must be thoroughly familiar with and make recommendations regarding company policy and regulatory requirements as they apply to records and personnel.
- This section must establish and resolve problems associated in document and drawing control to provide for revisions and accountability of master files and other files in use at the plant.

b. Plant Security

The Officer of the Guard is responsible to the Administrative Supervisor to:

- Develop, evaluate, implement, direct, review, and control an industrial security plan for the Crystal River site. (This plan must include a written, overall description of the security program designed to protect the Nuclear Plant and written instructions and/or procedures for bomb or other overt threats, civil disturbances, security training, control of incoming packages and materials, response to security alarm systems, etc.).

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- Coordinate with civil authorities regarding site security.
- Audit use of fuel storage area key and direct the proper handling of documents related to fuel receipt and shipment.
- Must determine requirements and provide solutions to NRC industrial security regulations. This includes writing the Security Plan and Security Procedures.
- Solve the problems associated with entry, key control, alarm systems, security lighting, and fence maintenance.
- Establish guardposts and schedule so guards are always available where needed.

c. Building and Grounds Maintenance

The Building Services Supervisor is responsible to the Administrative Supervisor to:

- Plan, develop, coordinate, and direct the administration of building and grounds maintenance and the janitorial services in the sense of housekeeping.
- Must determine requirements and resolve problems associated with manpower and scheduling to provide for custodial services, insect and rodent control, refuse collection and disposal, preventive maintenance, and fire protection and safety.
- Provide radiation contamination cleanup and laundry services, as required by the ChemRad Section.

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d. Plant Fire and Safety

- Assist in the formulation and administering of all plant fire and safety regulations. Also, review working conditions and recommend changes when hazards exist as per Florida Power Corporation Accident Prevention Manual.

- Review industrial and vehicular accidents within and including damage to the perimeter fence, determine their cause, and initiate and forward accident reports.

- Participate in Plant Safety Committee.

- As required, assist the ChemRad Section in the initiation, coordination, and administration of radiation exposure controls, by means of film badges and dosimeters, of all plant staff employees, members of tours, and visitors according to NRC, Federal, corporation or other regulations.

e. Training

The Training Coordinator is responsible to the Administrative Supervisor for the planning, scheduling, coordinating, and development of the training, retraining, and replacement training of licensed operators, Electrical and Mechanical Maintenance Repairmen, Technical Support Technicians, ChemRad Technicians, and Plant Engineers. He is also responsible for general employee training in the Emergency Plan, security, and radiation protection and for training and requalification documentation. It is desirable that he hold a Senior Reactor Operator's License. His responsibilities include the following:

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Training Programs

— Develops and schedules training programs to develop and maintain an organization qualified to be responsible for operation, maintenance, and technical aspects of the plant.

— Maintains all training records and documents.

— Evaluates licensed operators as required by 10 CFR 55 Appendix A, using written tests, oral tests, and operating demonstrations.

Safety

— Responsible for all emergency training and drills. Special emphasis is given to the safe practices applicable to accidents which might conceivably result in radioactive material release in and beyond the site boundary.

— Keep informed of the latest ideas in plant safety equipment.

— Train power plant employees in safe work practices.

— Participate in Plant Safety Committee.

Tours

— Responsible for conducting and/or coordinating all plant tours.

Outages

— Assist as outage supervisor supplying licensed Senior Reactor Operator supervision, if qualified, in the reactor building, spent fuel building, and auxiliary building as required.

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5.0 PLANT REVIEW COMMITTEE

The Plant Review Committee is appointed by the Plant Manager. This committee shall provide an on-site review to the Plant Manager in matters of nuclear safety, radiation exposure, and review and audit of plant operation, maintenance, and technical matters. This committee meets the requirements of ANSI N18.7-1976, "Administrative Controls for Nuclear Power Plants", Section 4.5. Refer to AI-300, Plant Review Committee Charter.

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7.0 REQUIRED STAFF QUALIFICATIONS BY POSITION

This section describes the minimum qualifications of education, skill, and experience required for each Nuclear Plant staff position. These requirements are consistent with ANSI 18.1-1971, paragraphs 4.2 thru 4.6. Please refer to Technical Specifications Figure 6.2-2, Facility Organization. Each supervisor shall be responsible for assuring that the employees reporting to him meet the minimum qualifications for the position he is filling.

7.1 Nuclear Plant Manager

The Nuclear Plant Manager shall have 15 years of responsible power plant experience of which a minimum of three years shall be nuclear power plant experience. The Nuclear Plant Manager shall have acquired the experience and training normally required for examination by the NRC for a Senior Reactor Operator's License whether or not the examination is taken. The Nuclear Plant Manager shall have a recognized Bachelor's Degree or higher degree in an engineering or scientific field generally associated with power production.

7.2 Technical Services Section

7.2.1 Technical Services Superintendent

The Technical Services Superintendent shall have 10 years of responsible power plant experience of which a minimum of three years shall be nuclear power plant experience. The Technical Services Superintendent shall have acquired the experience and training normally required for examination by the NRC for a Senior Reactor Operator's License whether or not the examination is taken. The Technical Services Superintendent shall have a

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recognized Bachelor's degree or higher degree in an engineering or scientific field generally associated with power production.

7.2.2 Performance Engineering

7.2.2.1 Performance Engineering Supervisor

The Performance Engineering Supervisor shall have a Bachelor's Degree in an engineering field normally related to power plant work and a minimum of eight years power plant experience. He should have advanced training in Nuclear Engineering, including studies in core analysis.

7.2.2.2 Results Engineer

The Results Engineer shall have a Bachelor's Degree in an engineering field normally related to power plant work and a minimum of five year's power plant experience.

7.2.2.3 Computer and Controls Engineer

The Computer and Controls Engineer shall have an advanced knowledge of Computer programming, nuclear and power plant instrumentation, principles of nuclear power plant operation, and a minimum of five years power plant experience. He shall have a Bachelor's Degree in Engineering.

7.2.2.4 Reactor Engineer

The Reactor Engineer shall have a Bachelor's Degree with advanced training in Nuclear Engineering, including studies in core analysis. He shall have five years of related experience, including two years of Nuclear Plant experience.

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7.2.3 Chemistry & Radiation Protection

7.2.3.1 ChemRad Protection Engineer

The ChemRad Protection Engineer shall cumulatively have a minimum of five years professional experience in radiation protection. A minimum of three of the five years shall be in a nuclear facility dealing with similar problems as experienced in this station. This position requires a Bachelor's Degree in engineering, chemistry, or physics, or equivalent acceptable experience.

7.2.3.2 Assistant ChemRad Protection Engineer

The Assistant ChemRad Protection Engineer shall have a minimum of five years experience in chemistry, radio-chemistry, and radiation protection. A minimum of two years of this five years experience will be related technical training. These positions require a Bachelor's Degree in engineering, chemistry, or physics, or equivalent acceptable experience.

7.2.3.3 Plant Engineer (ChemRad)

The Plant Engineer shall have a Bachelor's Degree or equivalent in an engineering or scientific field generally associated with power production.

7.2.3.4 Radiation Waste Supervisor

The Radiation Waste Supervisor shall have a minimum of five years experience in radiation protection at a nuclear facility. A minimum of two years of this five years experience should be related technical training. A minimum of four years of this five years experience may be fulfilled by related technical or academic training.

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7.2.3.5 Health Physics Supervisor

The Health Physics Supervisor shall have a minimum of five years experience in radiation protection at a nuclear facility. A minimum of two years of this five years experience should be related technical training. A maximum of four years of this five years experience may be fulfilled by related technical or academic training.

7.2.3.6 Chief ChemRad Technician

The Chief ChemRad Technician shall have three years of working experience in chemistry and radiation protection, one year of which shall be in a nuclear plant, and must meet the top qualifications of ChemRad Technician.

7.2.3.7 ChemRad Technician

The ChemRad Technician shall have two years of working experience involving chemistry, radio-chemistry, and radiation protection; must have one year of related technical training in chemistry and radio-chemistry or radiation protection; and must meet the top qualifications of Assistant ChemRad Technician.

7.2.3.8 Assistant ChemRad Technician

The Assistant ChemRad Technician shall pass the FPC Equivalency Exams in physics, chemistry, and math with a minimum score of 70 in each phase. The Assistant ChemRad Technician will also be screened by a professional consultant to determine his capability for learning basic skills in chemistry, radiochemistry, and radiation protection operations.

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7.2.4 Technical Support

7.2.4.1 Technical Support Engineer

The Technical Support Engineer shall have a minimum of eight years of responsible power plant experience of which one year shall be nuclear power plant experience. A maximum of four years of the remaining seven years of experience can be fulfilled by satisfactory completion of academic training. He shall have a Bachelor's Degree in engineering.

7.2.4.2 Inservice Inspection Engineer

The Inservice Inspection Engineer shall have a Bachelor's Degree in an engineering field normally related to power plant work and a minimum of five years power plant experience. He should be knowledgeable of Quality Control practices and Quality Assurance and Surveillance as required by ASME Section XI.

7.2.4.3 Nuclear Technical Specifications Coordinator

The Nuclear Technical Specifications Coordinator should have a Bachelor's Degree in an engineering field normally related to power plant work and a minimum of five years power plant experience. He should be knowledgeable of Technical Specifications, FSAR, Facility License, and Code of Federal Regulations requirements and reporting functions.

7.2.4.4 Plant Engineer

The Plant Engineer shall have a Bachelor's Degree or higher degree in an engineering or scientific field generally associated with power production, or a minimum of six years of engineering or related work experience of which three years shall be nuclear plant experience.

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7.2.4.5 Engineering Assistant

The Engineering Assistant shall have a minimum of an Associates Degree or equivalent in an engineering or scientific field generally associated with power production.

7.3 Maintenance Section

7.3.1 Maintenance Superintendent

The Maintenance Superintendent shall have a minimum of ten years responsible power plant experience or applicable industrial experience, a minimum of four years of which shall be nuclear power plant experience. He shall have non-destructive testing familiarity, craft knowledge, and an understanding of electrical, pressure vessel, and piping codes. He shall have a Bachelor's Degree in engineering.

7.3.2 Maintenance Staff Engineer

The Maintenance Staff Engineer shall have a minimum of seven years of responsible power plant experience or applicable industrial experience, a minimum of one year of which shall be nuclear power plant experience. He shall have non-destructive testing familiarity, craft knowledge, and an understanding of electrical, pressure vessel, and piping codes. He shall have a Bachelor's Degree in engineering.

7.3.3 Mechanical Supervisor

The Mechanical Supervisor shall have a high school diploma or equivalent and a minimum of four years experience in mechanical maintenance.

7.3.4 Electrical Supervisor

The Electrical Supervisor shall have a high school diploma or equivalent and a minimum of four years experience in electrical maintenance.

7.3.5 Instruments and Controls Supervisor

The Instruments and Controls Supervisor shall have a minimum of five years experience in instrumentation and control, of which a minimum of six months

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shall be in nuclear instrumentation and control. A minimum of two years of this five years experience should be related technical training. A maximum of four years of this five years experience may be fulfilled by related technical or academic training.

7.3.6 Contractor Supervisor

The Contractor Supervisor shall have a college diploma or equivalent and a minimum of four years of experience in the supervision of contractors.

7.3.7 Nuclear Master Mechanic

This position must have four years of working experience in related mechanical systems, one year of which shall be in a Nuclear Plant, and must meet the top qualifications of Nuclear Certified Welder Mechanic.

7.3.8 Nuclear Certified Welder Mechanic

This position must meet the top qualifications of Nuclear Mechanic and must be certified to FP-81 welding procedure.

7.3.9 Nuclear Mechanic

This position must have a minimum of three years of related mechanical experience; have certification of successful completion of a course in mechanical maintenance; and meet the top qualifications of Nuclear Apprentice Mechanic.

7.3.10 Nuclear Apprentice Mechanic

This position must have certification of successful completion of a course in basic shop fundamentals and will be screened by a professional consultant to determine capability for learning basic skills in mechanical maintenance.

7.3.11 Nuclear Chief Electrician

This position must have four years of working experience in related

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electrical systems, one year of which shall be in a Nuclear Plant, and meet the top qualifications of Nuclear Electrician.

7.3.12 Nuclear Electrician

This position must have three years working experience in related electrical systems; have certification of successful completion of a course in electrical theory; and meet the top qualifications of Nuclear Apprentice Electrician.

7.3.13 Nuclear Apprentice Electrician

This position must have certification of successful completion of a course in basic electrical theory and will be screened by a professional consultant to determine capability for learning basic skills in nuclear electrical maintenance operations.

7.3.14 Chief Nuclear Instruments and Controls Technician

This position must have four years of working experience in instrumentation and controls, one year of which shall be in a Nuclear Plant, and must meet the top qualifications of Nuclear Instruments & Controls Technician.

7.3.15 Nuclear Instruments & Controls Technician

This position must have three years of working experience in instrumentation and control systems; have one year of related technical training associated with instrumentation and control systems; and meet the top qualifications of Assistant Nuclear Instruments & Controls Technician.

7.3.16 Assistant Nuclear Instruments & Controls Technician

This position must pass the FPC Equivalency Exams in physics, chemistry, and math with a minimum score of 70 in each phase. The Assistant Nuclear

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Instruments & Controls Technician will also be screened by a professional consultant to determine capability for learning basic skills in nuclear instrumentation maintenance operations.

7.3.17 Planning Engineer

The Planning Engineer shall have a Bachelor's Degree in an engineering field normally related to power plant work and a minimum of five years power plant experience.

7.3.18 Planning Coordinator

The Planning Coordinator shall have a high school diploma plus two years technical school or equivalent. He shall also have five years of power plant operation/maintenance planning of which one of these five years shall have been nuclear power plant experience and three years shall have been maintenance or maintenance planning experience.

7.3.19 Maintenance Engineer

The Maintenance Engineer shall have a Bachelor's Degree in an engineering field normally related to power plant work and a minimum of five years power plant experience.

7.3.20 Materials Coordinator

The Materials Coordinator shall have a high school diploma plus two years technical school or equivalent. He shall also have five years of similar experience in maintenance or materials control.

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7.3.21 Administrative Technician

This position requires various general office and clerical skills consistent with FPC personnel policies.

7.4 Operations Section

7.4.1 Operations Superintendent

The Operations Sueprintendent shall have a minimum of ten years of responsible power plant experience of which a minimum of five years shall be nuclear power plant experience. The Operations Superintendent shall hold a Senior Reactor Operator's License.

7.4.2 Operating Engineer

The Operating Engineer shall have a minimum of eight years of responsible power plant experience of which a minimum of three years shall be nuclear power plant experience. The Operating Engineer shall hold a Senior Reactor Operator's License.

7.4.3 Shift Supervisors

The Shift Supervisor shall have a minimum of a high school diploma or equivalent and four years of responsible power plant experience of which a minimum of one year shall be Nuclear Plant experience. The Shift Supervisor shall hold a Senior Reactors Operator's License.

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7.4.4 Assistant Shift Supervisor

The Assistant Shift Supervisor shall have a minimum of a high school diploma or equivalent and four years of responsible power plant experience of which a minimum of one year shall be Nuclear Plant experience. The Assistant Shift Supervisor shall hold a Senior Reactor Operator's License.

7.4.5 Chief Nuclear Operator

This position must hold an NCR Reactor Operator's License for CR-3; have a minimum of three years power plant experience of which one year must be Nuclear Plant experience; and meet the top qualifications of Nuclear Operator.

7.4.6 Nuclear Operator

This position must hold an NRC Reactor Operator's License for CR-3 and meet the top qualifications of Assistant Nuclear Operator.

7.4.7 Assistant Nuclear Operator

This position must hold an NRC Reactor Operator's License for CR-3; have a minimum of two years power plant experience of which one year must be Nuclear Plant experience; and have the top qualifications of a Nuclear Auxiliary Operator.

7.4.8 Nuclear Auxiliary Operator

This position must be qualified as an Auxiliary Operator or higher at a fossil or nuclear power plant with one year power plant operating experience and must meet the top qualifications of Assistant Nuclear Auxiliary Operator.

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7.4.9 Assistant Nuclear Auxiliary Operator

This position will be screened by a professional consultant to determine capability for learning basic requirements of nuclear operations and progressing to eventual NRC Licensing. The Assistant Nuclear Auxiliary Operator shall have a high shcool diploma or equivalent.

7.5 Administrative Section

7.5.1 Administrative Supervisor

The Administrative Supervisor shall be a high school graduate with a minimum of five years experience in the planning, scheduling, and coordination of clerical functions, building services, and security systems.

7.5.2 Office Manager

The Office Manager shall be a high school graduate with a minimum of five years experience in the planning, scheduling, and coordination of clerical functions, building services, and security systems

7.5.3 Office Clerks

These positions depending upon the classification, require skills developed through formal training, experience, or a combination of the two in general clerical activities such as: filing and retrieval, records processing, exposure to common office machinery and specific skills related to the classification as specified in FPC personnel policies.

7.5.4 Officer of the Guard

Security personnel shall have qualifications as stated in Section 7.0 (Security Guard Force) of the Crystal River Nuclear Plant Security Plan and which are consistent with ANSI N18.7-1976, Section 4 (Administrative Controls).

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7.5.5 Guards

Security personnel shall have qualifications as stated in Section 7.0 (Security Guard Force) of the Crystal River Nuclear Plant Security Plan and which are consistent with ANSI N18.7-1976, Section 4 (Administrative Controls).

7.5.6 Training Supervisor

The Training Supervisor shall have a high school diploma or equivalent and advanced courses in nuclear power, power plant theory, and electronics beyond the level required for a Senior Reactor Operator's License. He shall hold, have held, or be in the process of obtaining a Senior Reactor Operator's License.

7.5.7 Training Records Clerk

This position requires skills developed through formal training, experience, or a combination of the two in general clerical activities such as: filing and retrieval, records processing, and other such functional skills as specified in FPC personnel policies.

7.5.8 Training Specialist

The Training Specialist shall have a minimum of a high school diploma or equivalent and four years of responsible power plant experience of which a minimum of one year shall be nuclear power plant experience.

7.5.9 Building Services Supervisor

The Building Services Supervisor shall have a high school diploma or equivalent and a minimum of four years of experience in decontaminating Nuclear Plant, maintaining office, and other service areas in proper order.

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7.5.10 Administrative Planner

This position requires various general office and clerical skills, including experience in scheduling and dispatching materials and personnel for building service tasks.

7.5.11 Building Servicemen

These positions will require the general ability, maturity, and responsibility to perform janitorial, yardwork, laundry, and other service tasks for maintaining offices and other service areas in proper order. They will be screened by a professional consultant to determine their capability for learning basic skills.

7.6 NUCLEAR QA/QC COMPLIANCE SECTION

7.6.1 Nuclear QA/QC Compliance Manager

The NQA/QC Compliance Manager shall have an extensive knowledge of QA practices and regulatory requirements, including Codes of Federal Regulations, FPC Quality Programs, CR-3 FSAR, Technical Specifications, and applicable Codes and Standards. Although a Bachelor's Engineering Degree is desirable, it should not be a prerequisite as experience should be able to preclude the requirement. As a minimum, the requirements should be a B. A. and five years of QA-related experience or no degree and 10 yrs. QA-related experience.

7.6.2 Nuclear QA/QC Supervisor

The NQA/QC Supervisor shall have an extensive knowledge of QA practices and regulatory requirements, including Codes of Federal Regulations, FPC Quality Programs, CR-3 FSAR, Technical Specifications, and applicable Codes and Standards. Although a Bachelor's Engineering Degree is desir-

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able, it should not be a prerequisite as experience should be able to preclude the requirement. As a minimum, the requirements should be a B. A. and five years of QA-related experience or no degree and 10 yrs. QA-related experience.

7.6.3 Nuclear Compliance Supervisor

The Nuclear Compliance Supervisor shall have an extensive knowledge of QA practices and regulatory requirements, including Codes of Federal Regulations, FPC Quality Programs, CR-3 FSAR, Technical Specifications, and applicable Codes and Standards. Although a Bachelor's Engineering Degree is desirable, it should not be a prerequisite as experience should be able to preclude the requirement. As a minimum, the requirements should be a B. A. and five years of QA-related experience or no degree and 10 yrs. QA-related experience.

7.6.4 Nuclear Compliance Auditors

The Nuclear Compliance Auditors shall have a minimum of 4 yrs. of engineering or related work experience, at least 2 yrs. of which were related to Compliance or QA work on one or more of the following areas: design, construction, testing, maintenance, or system operation. They should have an Associate's Degree in an engineering field, or equivalent.

7.6.5 Nuclear QA/QC Inspector

The NQA/QC Inspector shall have a high school diploma plus 2 yrs. of technical school or college education. He should be certified, or have training, as a non-destructive examiner in ultrasonic, penetrant, magnetic particle, or eddy current methods. He should be capable of meeting the experience requirements of ANSI N45.2-6 (1973) for a Level

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II person involved in inspections and tests. The NOA/QC Inspector must have 4 yrs. of experience in QA testing or inspection, 2 yrs. of which have been in a Nuclear Plant.

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ATTACHMENT VIII

LONG RANGE PLAN FOR UPGRADING THE ON-SITE TECHNICAL SUPPORT CENTER

Florida Power Corporation will construct a new facility to act as a Technical Support Center (TSC) during emergency situations. This new facility will be located inside the security perimeter, below the berm, northeast of the plant.

The size of the TSC will be sufficient to support a minimum of 25 people and provide room for all required records, drawings, communications, and instrumentation.

The instrumentation within the TSC shall include the capability to monitor current and historical data for the primary and secondary plant parameters which are regarded as necessary for plant accident analysis. In addition, plant radiological parameters and site meteorological data is planned to be displayed in the TSC.

The CR-3 plant computer Central Processor Unit is not presently capable of providing independent access from remote locations, but the outputs can be paralleled and monitored in the TSC. FPC is planning to replace the existing computer Central Processing Unit with a unit capable of providing independent data access from remote terminals. The present schedule for completion of this modification is Fall of 1981.

Reliable communications will be available in the TSC and will include dedicated lines between the TSC and the Control Room, between the TSC and the NRC. In addition, commercial telephone lines for on-site and off-site communications will be available as well as intra-plant lines which include private, party and paging functions.

The power supply for the TSC will be designed to be available at all times, including during a loss of offsite power. Where needed, to avoid loss of stored data due to momentary loss of power or switching transients, the power to the applicable instruments and equipment will be continuous (uninterruptable).

Sufficient records and drawings to aid in accident analysis will be permanently stored and available for use in the TSC.

The structural design of the TSC will take into account the effects of natural phenomena that may occur at the site, and will be designed to meet or exceed existing codes.

Protection from radiological hazards, including direct radiation and airborne contaminants as per General Design Criterion 19 and SRP 6.4, will be provided for in the design of the TSC. Permanent monitoring systems will be provided to indicate radiation dose rates and airborne radioactivity concentrations inside the TSC. At the present time, FPC does not have a system of off-site radiological monitoring systems which can provide data, but instead by utilizing known source release and known meteorological information, the release data can be properly evaluated by overlays and equations. If required, a survey team could be dispatched from the TSC to sample and verify the release data.

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