Florida Power
CORPORATION

Walter S. Wilgus
Vice President
Nuclear Operations

December 31, 1987
3F1287-27

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Technical Specification Change Request No. 159
Post-Accident Monitoring Instrumentation

Dear Sir:

Florida Power Corporation (FPC) hereby submits Technical Specification Change No. 159 requesting amendment to Appendix A of Operating License No. DPR-72. As part of this request, the proposed replacement pages for Appendix A and the bases are enclosed.

This submittal revises surveillance and operability requirements for post-accident monitoring instrumentation installed to satisfy FPC's commitments to Regulatory Guide 1.97, Revision 3. In determining which post-accident monitoring instruments should be included in Technical Specifications (T.S.), the Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors was utilized. The application of these criteria is consistent with the Babcock and Wilcox Owners Group Technical Specification Committee position.

This change request supercedes the previous technical specification post-accident monitoring update as submitted in Change Request No. 82.

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w/check
$150.00

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An amendment application fee, check number 19282 of one hundred fifty dollars ($150), as required by 10 CFR 170, has been included with this change request.

Sincerely,

W.S. Wilgus
Vice President
Nuclear Operations

JBC/dhd
Enclosure

xc: Dr. J. Nelson Grace
Regional Administrator, Region II

Mr. T.F. Stetka
Senior Resident Inspector

STATE OF FLORIDA

COUNTY OF CITRUS

W.S. Wilgus states that he is the Vice President, Nuclear Operations for Florida Power Corporation; that he 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 his knowledge, information, and belief.

W S Wilgus

W.S. Wilgus, Vice President
Nuclear Operations

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 31st day of December 1987.

Jean Lisa Buffe

Notary Public

Notary Public, State of Florida at Large,
My Commission Expires: June 21, 1991

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER	)	
	)	DOCKET NO. 50-302
FLORIDA POWER CORPORATION	)	

CERTIFICATE OF SERVICE

W.S. Wilgus deposes and says that the following has been served on the Designated State Representative and Chief Executive of Citrus County, Florida, by deposit in the United States mail, addressed as follows:

Chairman,
Board of County Commissioners
of Citrus County
Citrus County Courthouse
Inverness, FL 32650

Administrator
Radiological Health Services
Department of Health and
Rehabilitative Services
1323 Winewood Blvd.
Tallahassee, FL 32301

A copy of Technical Specification Change Request No. 159, Revision 0 requesting Amendment to Appendix A of Operating Licensing No. DPR-72.

FLORIDA POWER CORPORATION

W.S. Wilgus, Vice President
Nuclear Operations

SWORN TO AND SUBSCRIBED BEFORE ME THIS 31ST DAY OF DECEMBER 1987.

Joan Lisa Buffé
Notary Public

Notary Public, State of Florida at Large
My Commission Expires: June 21, 1991

(NOTARY SEAL)

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302/LICENSE NO. DPR-72
REQUEST NO. 159, REVISION 0
POST-ACCIDENT MONITORING INSTRUMENTATION

LICENSE DOCUMENT INVOLVED: Technical Specifications

PORTIONS: 3.3.3.6, Table 3.3-10 and Table 4.3-7

DESCRIPTION OF REQUEST:

This submittal revises surveillance and operability requirements for post-accident monitoring instrumentation installed to satisfy Florida Power Corporation's (FPC) commitments to Regulatory Guide 1.97 (R.G. 1.97), Revision 3.

Instrument names, ranges and minimum number channels operable have been revised to be consistent with FPC's response to R.G. 1.97. Some instruments were added and others were relocated to the FSAR.

REASON FOR REQUEST:

This revision is being requested to reflect changes which will result from the modification currently being installed. R.G. 1.97 instruments which meet the criteria for inclusion in Technical Specifications have been added. Instruments which did not meet the criteria for inclusion in Technical Specifications have been relocated to the FSAR as Supplemental Specifications (SS). The criteria utilized are specified in the Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors (52 Fed. Reg. 3788, 2/6/87).

EVALUATION OF REQUEST:

At the start of an accident, it may be difficult for the operator to determine immediately what accident has occurred or is occurring and therefore to determine the appropriate response. For this reason, reactor trip and certain other safety actions (e.g., emergency core cooling actuation, containment isolation, or depressurization) have been designed to be performed automatically during the initial stages of an accident. Instrumentation is also provided to indicate information about plant variables required to enable the operation of manually initiated safety systems and other appropriate operator actions involving systems important to safety. The post-accident monitoring instrumentation, comprised of instrumentation displays to assess plant and environs conditions during and following an accident, provides this information.

In order to determine which post-accident monitoring instruments should be included in Technical Specifications, the Commission Policy Statement on Technical Specification Improvements for Nuclear Power Reactors was utilized. This policy statement addresses the scope and purpose of Technical Specifications. In doing so, it established a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

CRITERION 1: Installed instrumentation that is used to detect and indicate in the Control Room, a significant abnormal degradation of the reactor coolant pressure boundary;

CRITERION 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

CRITERION 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier;

In addition to those structures, systems, and components captured by the above criteria, it is the Commission's policy that Licensees retain in their technical specifications LCO's, Action Statements, and Surveillance Requirements for systems which operating experience and Probabilistic Risk Assessment have generally shown to be important to public health and safety.

The application of these criteria is shown in Attachment 1. The instruments which did meet the criteria have been located in Technical Specifications. For the instruments which did not meet the criteria, FPC proposes to remove them from the Technical Specification and relocate the instruments, their requirements and bases in the FSAR. These requirements will be termed Supplemental Specifications (SSs).

10CFR50.59 will be utilized as the control mechanism for the SSs as they will be placed in the FSAR. This would allow Florida Power Corporation (FPC) to make changes to these specifications if the change does not involve an unreviewed safety question. Plant Review Committee (PRC) and Nuclear General Review Committee (NGRC) review and approval will be required prior to implementation of changes. A report of changes to SSs, made utilizing the provisions of 10CFR50.59 will be furnished annually to Region II with a copy to the Director of Inspection and Enforcement. Additionally, the FSAR will be updated annually, pursuant to 10CFR50.71 to reflect any changes to these specifications.

Maintenance of the instrumentation in the SS will not be affected by this Technical Specification change. FPC will continue to perform periodic testing to assure that the instrumentation is maintained operable. Relocating specific requirements for the instruments shown in the attached Table 15.3-1 will have no impact on their operability.

For any future changes to the instruments shown in Table 15.3-1, the provisions of 10CFR50.59 will be utilized. Additionally, a multidisciplinary review by the PRC and the NGRC will be performed. These controls are considered adequate for assuring these instruments are maintained operable.

The proposed amendment incorporates operability and surveillance requirements for additional post-accident monitoring instruments consistent with FPC's commitment to R.G. 1.97. These additional instruments are: Containment Sump Level (Narrow Range), Core Exit Thermocouples, Emergency Feedwater Tank Level and Containment High Range Radiation Monitor. These instruments were added as a result of applying the T.S. criteria discussed previously.

This amendment also revises some existing T.S. instrument names and minimum number channels operable consistent with FPC's specific commitments to R.G. 1.97. Ranges were also changed, where necessary, to be consistent with FPC's commitment and the intent of R.G. 1.97. Instruments whose range changed have been designed to meet or exceed the existing design requirements. This instrumentation is capable of surviving the accident environment in which it is located for the length of time its function is required. Instruments have also been designed to continue to function following a seismic event. Therefore, the post-accident instrumentation is designed to withstand the effect of an earthquake. As such, better assurance is provided that these instruments will be available for operator use following an accident.

The NRC has, in their SER dated June 16, 1987, approved FPC's specific commitments for ranges with regard to compliance to R.G. 1.97, revision 3.

Operability and surveillance requirements have been relocated to SSs in the FSAR for the following instruments: Reactor Building Pressure, Reactor Coolant Total Flow, Startup Feedwater Flow, PORV Position Indicator (Primary and Backup Detectors), PORV Block Valve Position Indicator, Safety Valve Position Indicator (Primary and Backup Detectors), and Emergency Feedwater Flow. These instruments have been relocated as a result of T.S. criteria application discussed above.

Additionally, references to instrument recorders have been removed from Technical Specifications. This information is considered to be part of the design basis and therefore, it would be more appropriate to describe them and relocate them to the FSAR. The FSAR will be revised to reflect recorder information specifically provided for in FPC's commitment to R.G. 1.97. 10CFR50.59 will be utilized as the control process for changes to recorder design.

These proposed changes will not affect the design basis of the system or components, as instrumentation will still be available to assess plant and environmental conditions during and following an accident.

SHOLLY EVALUATION OF REQUEST:

Florida Power Corporation (FPC) proposes the change to the surveillance and operability requirements for post-accident monitoring instrumentation does not involve a significant hazards consideration.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated. The proposed amendment ensures operability of instrumentation installed to monitor selected plant variables and systems during and following the postulated accidents previously analyzed and has no affect on the probability of occurrence or consequences of these previously evaluated accidents.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed amendment has no affect on the possibility of creating a new or different kind of accident from any accident previously evaluated. The proposed amendment ensures operability of instrumentation installed to monitor selected plant variables and systems during and following the postulated accidents previously analyzed and is unrelated to the possibility of creating a new or different kind of accident.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety. The proposed amendment constitutes an additional control not presently included in the Technical Specifications. Additionally, all instruments are now qualified in accordance with Regulatory Guide 1.97. Therefore, the overall margin of safety for the plant is increased.

FPC proposes that relocation of operability and surveillance requirements for the nine (9) instruments, which do not meet the criteria for Technical Specification inclusion, does not involve a significant hazards consideration. The requirements and surveillances for those instruments will continue to be maintained and performed.

Based on the above, FPC finds that this change will not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated because the instrument operability and surveillance requirements will not be affected. The proposed change relocates requirements and surveillances for post-accident monitoring instruments which did not meet the criteria for inclusion in Technical Specifications. These instruments will be relocated from the Technical Specifications to the FSAR. These operability requirements and surveillances will continue to be maintained pursuant to 10 CFR 50.59.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated because adequate controls for the relocated requirements exist under the provisions of 10 CFR 50.59.

3. Involve a significant reduction in the margin of safety because the instrument requirements and surveillances are the same or more restrictive than the existing Technical Specification. For any future changes to the post-accident requirements, the provisions of 10 CFR 50.59 and 50.71 will be upheld.

ATTACHMENT 1

APPLICATION OF

TECHNICAL SPECIFICATION CRITERIA

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
Neutron Flux	A,B/1	Monitor reactivity, accomplishment of mitigation	3/YES	Immediately following a reactor trip, the operator monitors the initial drop in the neutron flux level to ensure that the reactor has been shutdown. The operator can continue to monitor the neutron flux level decay down to the source range levels. Additionally, entry into the emergency reactivity control procedure is based on the neutron flux variable. Operator actions: Start boration at $\geq$ ( ) GPM from BAST.
Control Rod Position	B/3	Verify reactor is shutdown	NO	This provides verification that the reactor is shut down by monitoring the position of the control rod. This provides backup verification for neutron flux, therefore, does not meet criteria.
RCS Soluble Boron	B/3	Verify reactor is maintained subcritical	NO	The RCS soluble boron concentration can be determined two ways: 1) by direct measurement with a device such as a boronmeter or 2) by manual sampling and laboratory analysis. Instrumentation is not installed in the plant. Manual sampling and laboratory analysis is considered to be sufficient. The loss of the negative reactivity due to xenon decay is sufficiently slow that the control room operator need not know instantaneously or continuously what the boron concentration is in the RCS. Therefore, does not meet criteria.
RCS Hot Leg Water Temperature	A,B/1	Monitor core cooling, accomplish mitigation	3/YES	Input into subcooling margin determination which is used for RC pump trip criteria. Variable for monitoring the core cooling plant safety function. It is used with RCS pressure to monitor the status of RCS with respect to saturation and subcooling margin limits. It is used with RCS Cold Leg Water Temperature to measure the temperature rise across the core for verification of natural circulation conditions when the Reactor Coolant Pumps are tripped.

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
RCS Hot Leg Water Temperature (Continued)				Operator actions: Reduce OPERATING RCPs to one per loop. Maintain full HPI/LPI flow until required subcooling margin is established.
RCS Cold Leg Water Temperature	B/3	Monitor core cooling, accomplish mitigation	NO	This is a backup instrument to the RCS Hot Leg Water Temperature and Core Exit Temperature instruments. It is not part of the primary success path and therefore does not meet criteria.
RCS Pressure	A,B,C/1	Monitor core cooling, accomplish mitigation, detect RCS breach	1,3/YES	Input into subcooling margin determination which is used for RC pump trip criteria. Operator actions: Verify PORV or Hot Leg vents or PZR vents used to maintain RC pressure.
Core Exit Temperature	A,B,C/1	Detect breach of fuel clad, accomplish mitigation	3/YES	Used as input to subcooling margin monitor during loss of offsite power to mitigate the DBA without RC pump flow. Operator actions: Utilizing a graph of RCS pressure versus Core Exit Thermocouple Temperature, the operator is required by procedures to take immediate action to provide core cooling.
Coolant Inventory (Reactor Vessel Level)	B/1	Detect voids in coolant, accomplish mitigation	NO	The reactor coolant inventory tracking system was not designed to properly function while the reactor coolant pumps (RCPs) are running (ΔP type system). It is to be used for trending purposes only. Since the RCPs will be utilized to facilitate mitigation of an accident, the use of this system would be longer term. Does not meet purpose of Tech Specs. Inventory tracking systems are post-TMI concepts that will provide information for any condition of partial voiding, but are primarily intended for unusual events where extraordinary measures (such as secondary blowdown, or loop or head venting) may be needed. Credit for inventory tracking systems is not needed for Design Basis Accidents.

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
				The method used is to measure hot leg water level (delta P); the measurement can only be taken with the RC pumps stopped. The instrument is not sufficiently precise to be used for purposes other than general trending.
Degrees of Sub-Cooling	A,B/2	Accomplish mitigation (subcooling margin is the RCP trip criteria)	3/YES	Part of operator action to trip RC pumps on loss of subcooled margin. Operator actions: Reduce <u>OPERATING</u> RCPs to one per loop. Maintain full HPI/LPI flow until required subcooling margin is established.
Containment Sump Water Level				
- Narrow Range	B,C/2	Accomplish mitigation, detect RCS breach	1,3/YES	Used for RCS leakage detection system currently in Technical Specifications.
- Wide Range	B,C/1	Accomplish mitigation, detect RCS breach	NO	The main intent is to provide containment inventory tracking for flooding of important components for events where continued injection from alternate sources is used after the BWST contents are exhausted. Critical components and instrumentation have been located above flood level for Design Basis Events where injection is limited to the BWST contents and sump recirculation is used.
Containment Pressure	B,C/1	Accomplish mitigation, detect RCS breach	NO	The typical range for the instrument is slightly above containment design pressure. The principal use is to confirm that containment heat removal systems are causing a containment pressure decrease following an energy release. Thus, it permits backup verification of heat removal system performance. Failure to control pressure usually means that all heat removal systems have failed (beyond Design Basis) or the energy input is excessive (beyond Design Basis) and that systems to limit energy have failed in some fashion (high volume H_2, continued secondary steam input, failure to to trip the reactor, etc.)

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
				While the instrument provides a useful monitor, usually actions based on its information are only contingency actions to correct for unexpected problems and other instrumentation must be additionally consulted to develop decisions for corrective action. Containment pressure is not used for specific operator corrective actions following a DBA.
Containment Isolation Valve Position	B/1	Accomplish containment isolation	NO	Does not meet criteria. CIV position is not part of primary success path. Automatic actuation occurs on ES signal. No manual operator action required.
Radioactivity Concentration or RCS Radiation Level	C/3	Indicate fuel failures	NO	No instrumentation exists to adequately measure this variable on-line. The primary means of monitoring this variable must therefore be by sampling and analysis. Does not meet criteria for Tech Specs.
Analysis of Primary Coolant	C/3	Detail analysis of RCS source term	NO	Long term surveillance of source term in RCS for dose calculations.
Containment Area Radiation - High Range	C,E/1	Detect a RCS breach	1/YES	Used for RCS leakage detection system currently in Technical Specifications.
Effluent Radioactive - Noble Gas effluent from Condenser Air Removal System Exhaust	C/3	Detect RCS breach in the reactor coolant pressure boundary	NO	This is a backup instrument to the RCS Leakage Detection System already in Tech Specs. A breach will be detected by the RCS Leakage Detection System. Therefore, does not meet criteria.

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
Containment Hydrogen Concentration	C/1	Detect RCS breach	NO	No immediate operator action required. Long term occurrence. The original intent of the analyzers (monitors) and recombiners (where installed) is to control hydrogen in the long term after a Large Break LOCA. Assumptions of the LOCA analysis include minor hydrogen generation from the zirc-clad-water reaction, from spray reaction with aluminum and zinc surfaces in the containment, from radiolytic decomposition of the injection fluid and other sources. For these analyses, gradual hydrogen accumulation is conservatively calculated to occur over several days/weeks after the accident is terminated and core cooling stabilization has occurred. No immediate post-accident threat is predicted. These components are not used for mitigation of the core thermal hydraulic transient, and are not directly related to the Design Basis Accident sequence. PRA studies of core melt sequences show that large volumes of hydrogen are developed very rapidly from the zirc-water reaction. The H_2 volume, is too great for control by recombiners. Therefore, these components are not important for post-core melt H_2 control.
Containment Effluent Radioactivity - Noble Gases from Identified Release Points	C/2	Detect Breach of Containment	NO	These are backup instruments to RCS leakage detection instruments for indication of a breach of containment.
Effluent Radioactivity - Noble Gases (from buildings or areas where penetrations and hatches are located)	C/3	Indication of Containment Breach	NO	These are backup instruments to RCS leakage detection instruments for indication of a breach of containment.

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
RHR System Flow (Decay Heat)	D/2	Monitor Operation of Decay Heat Removal System	NO	Monitor only - does not require immediate operator action, therefore, does not meet criteria.
RHR System Outlet Temperature (Decay Heat)	D/2	Monitor Operation of Decay Heat Removal System	NO	Monitor only - does not require immediate operator action, therefore, does not meet criteria.
Accumulator Tank Level and Pressure (Core Flood Tank)	D/2,3	Monitor Tank Pressure and Monitor Tank Level	NO	Pressure: during an accident, this variable is not monitored and no operator actions are based on it. Level: ATOG guidelines do not use tank level as the primary input variable for isolating the tank; LPI flow is used.
Accumulator Isolation Valve Position (Core Flood Tank)	D/2	Indication of Position.	NO	Position indicator only - No operator response required, therefore, does not meet criteria.
Boric Acid Charging Flow	D/2	Monitor Operation of the ECCS	NO	Design of B&W plant does not include a charging system. Normal means for injecting boric acid into the RCS are via the Makeup and Purification System or High Pressure Injection System.
Flow in HPI System	D/2	Monitor Operation of HPI System	NO	Monitor only; does not require operator immediate actions, therefore, does not meet criteria.
Flow in LPI System	D/2	Monitor Operation of LPI System	NO	Monitor only; does not require operator immediate actions, therefore, does not meet criteria.
Borated Water Storage Tank Level	A,D/1	Monitor Operation	3/YES	Switchover from BWST to sump is a manual operator action. Operator actions: BWST level $\leq$ [ ] feet, then establish HPI suction from LPI. BWST level $\leq$ [ ] feet, then establish LPI suction from RB sump.

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
Reactor Coolant Pump Status	D/3	Monitor Operation of RCPs	NO	Monitor only, does not require operator immediate actions, therefore, does not meet criteria.
Primary System Safety Relief and Block Valve Positions	D/2	Monitor Valve Position	NO	Provides indication only. No immediate operator action required. Does not meet criteria. For pressurizer safeties, this only confirms an open leak path for those plants with safety valve P.I.s. Action to close leaking code safeties is not possible. The instrument is not used for Design Basis Accident evaluations in the SBLOCA analyses. Both P.I.s are post-TMI-2 additions considered useful for a prominent location for an isolable loss-of-coolant leak path. Some plants may not have a block valve P.I. The position indication is not specified for Design Basis Events; in essence, Design Basis SBLOCAs have examined a spectrum of RCS leaks to the containment and shown plant safety without isolation. Obviously, it is desirable to isolate leaks as the preferred method of termination, but PORV isolation is not strictly required for core protection.
Pressurizer Level	D/1	Monitor Status of RCS	3/YES	Many operator actions during transients and accidents are dependent on knowledge of the status of pressurizer level in order to determine and control RCS inventory.
Pressurizer Heater Status	D/2	Monitor Pressurizer Heater Status	NO	Monitor only. No operator immediate action required, therefore, does not meet criteria.
Quench Tank Level (Reactor Coolant Drain Tank)	D/3	Monitor Operation	NO	Monitor only. No operator immediate action required, therefore, does not meet criteria.
Quench Tank Temperature (Reactor Coolant Drain Tank)	D/3	Monitor Operation	NO	Monitor only. No operator immediate action required, therefore, does not meet criteria.

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
Quench Tank Pressure (Reactor Coolant Drain Tank)	D/3	Monitor Operation	NO	Monitor only. No operator immediate action required, therefore, does not meet criteria.
Steam Generator Level	A,D/1	Monitor Operation of Secondary System	3/YES	Operations needs to have level information in order to maintain proper levels in the steam generators based on automatic or manual action. Operator actions: Ensure full HPI/LPI flow; OTSG levels at 95%; OTSG TSAT 90-110°F below TSAT for existing RC pressure.
Steam Generator Pressure	A,D/1	Monitor Operation of Secondary System	3/YES	Operator needs to have pressure information in order to maintain proper levels in the steam generators based on automatic or manual action. Operator actions: Ensure PORV block valve is open; open PORV; open PZR vents.
Safety/Relief Valve Positions or Main Steam Flow	D/2	Monitors Valve Position	NO	Position indication only. No immediate operator response required, therefore, does not meet criteria.
Main Feedwater Flow	D/3	Monitor Operation	NO	Backup variable to monitor steam generator level and pressure. No immediate operator response required, therefore, does not meet criteria.
Startup Feedwater Flow	N/A	Monitor Operation	NO	Does not meet criteria. No manual operator action required, nor is there any automatic actuation on an ES signal. Adequate steam generator control exists with EFIC System (in the emergency mode). Startup feedwater flow is used only after recovery from an accident.

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
Auxiliary or Emergency Feed-water Flow	D/2,3	Monitor Operation	NO	Does not meet criteria. Automatic actuation occurs on ES signal. No manual operator action required. Although emergency feedwater (EFW) is important for DBA mitigation, emergency feedwater flow rate indication is not needed. EFW system operability can be verified by monitoring steam generator level and reactor coolant system temperature. EFW flow rate indication is not used to initiate any specific operator corrective actions to mitigate the consequences of a DBA.
Emergency Feedwater Tank Water Level	A/1	Ensure water supply for feedwater	3/YES	Primary source of water for EFW System. Operator will have to manually align to another source, if required, to maintain EFW flow. Operator actions: Maintain EFT tank level > [ ] feet.
Containment Spray Flow	D/2	To Monitor Operation	NO	Does not meet criteria. Monitor only. Automatic actuation occurs on ES Signal. No manual operator action required.
Heat Removal by the Containment Fan Heat Removal System	D/2	To Monitor Operation	NO	Does not meet criteria. Monitor only. Automatic actuation occurs on ES Signal. No manual operator action required.
Containment Atmosphere Temperature	D/2	To Monitor Containment Temperature	NO	Does not meet criteria. Monitor only. No immediate operation action required.
Containment Sump Water Temperature	D/2	To Monitor Operation of Containment Cooling System	NO	Does not meet criteria. Monitor only. Not required to mitigate the consequences of a design basis event. No manual actions are initiated based on this temperature.
Make-up Flow-In	D/3	To Monitor Operation	NO	Does not meet criteria. Monitor only. No immediate operator action required.

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
Letdown Flow-Out	D/3	To Monitor Operation	NO	Does not meet criteria. Monitor only. No immediate operator action required.
Make-up Tank Level	D/2	To Monitor Operation	NO	Does not meet criteria. Monitor only. No immediate operator response required.
Component Cooling Water Temperature to ESF System	D/2	To Monitor Operation	NO	Does not meet criteria. Monitor only. No immediate operator response required.
Component Cooling Water Flow to ESF System	D/2	To Monitor Operation	NO	Does not meet criteria. Monitor only. No immediate operator response required.
High-Level Radio-active Liquid Tank Level	D/3	To Indicate Storage Volume	NO	Does not meet criteria. Monitor only. No immediate operator response required.
Waste Decay Tank Pressure	D/3	To Indicate Storage Capacity	NO	Does not meet criteria. Monitor only. No immediate operator response required.
Emergency Venti-lation Damper Position	D/2	To Indicate Damper Status	NO	Does not meet criteria. Monitor only. No immediate operator response required.
Status of Stand-by Power and Other Energy Sources Important to Safety (electric, hydraulic, pneumatic,) (voltages, currents, pressures)	D/2	To Indicate System Status	NO	Does not meet criteria. Monitor only. No immediate operator response required.
Radiation Exposure Rate (inside build-ings or areas where access is required to service equipment important to safety	E/3	Detection of significant releases; release assessment; long-term surveillance	NO	Not in Crystal River Unit 3 design. Does not meet criteria.

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
Containment or Purge Effluent	C,E/2	Detection of significant releases; release assessment	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Reactor Shield Building Annulus (if in design)	E/2	Detection of significant releases; release assessment	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Auxiliary Building (including any building containing primary system gases, e.g., waste gas decay tank)	C,E/2	Detection of significant releases; release assessment; long-term surveillance	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Condenser Air Removal System Exhaust	C,E/2	Detection of significant releases; release assessment	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Common Plant Vent or Multi-purpose Vent Discharging Any of Above Releases (if containment purge is included)	E/2	Detection of significant releases; release assessment; long-term surveillance	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Vent from Steam Generator Safety Relief Valves or Atmospheric Dump Valves	E/2	Detection of significant releases; release assessment	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
All Other Identified Release Points	E/2	Detection of significant releases; release assessment; long-term surveillance	NO	Does not meet criteria. For release assessment only. No immediate operator response required.

VARIABLE	TYPE/ CATEGORY	FUNCTION	CRITERIA/ RETAINED IN TS	COMMENTS
All Identified Plant Release Points (except steam generator safety relief valves or atmospheric steam dump valves and condenser air removal system exhaust). Sampling with Onsite Analysis Capability	E/3	Detection of significant releases; release assessment; long-term surveillance	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Airborne Radiohalogens and Particulates (portable sampling with onsite analysis capability)	E/3	Release assessment; analysis	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Plant and Environs Radioactivity (portable instrumentation)	E/3	Release assessment; analysis	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Wind Direction	E/3	Release assessment	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Wind Speed	E/3	Release assessment	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Estimation of Atmospheric Stability	E/3	Release assessment	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Primary Coolant and Sump	E/3	Release assessment; verification; analysis	NO	Does not meet criteria. For release assessment only. No immediate operator response required.
Containment Air	E/3	Release assessment; verification; analysis	NO	Does not meet criteria. For release assessment only. No immediate operator response required.