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SAFETY PARAMETER DISPLAY SYSTEM

SAFETY ANALYSIS

FOR THE

CRYSTAL RIVER NUCLEAR GENERATION STATION

UNIT 3

FOR

FLORIDA POWER CORPORATION

BY

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EXECUTIVE SUMMARY

The purpose of this document is to provide a written safety analysis for the Crystal River Unit 3 (CR-3) SPDS. This analysis describes the basis on which the selected parameters are sufficient to assess the safety status of the plant with respect to five required functions of the SPDS for a wide range of events. This analysis is in response to a requirement for such a document in Section 4.2 of NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability" (Generic Letter 82-33), dated December 17, 1982.

The CR-3 SPDS consists of two color video monitors (CRTs) with associated control panels. This system, located in the control room, allows the control room personnel to select from a set of pre-programmed displays and also to select certain information for display depending on the status of the plant. The displays, their format, and the parameters monitored are based on two major inputs: (1) NRC requirements for SPDS functions and (2) compatibility to ATOG concepts and requirements.

This document summarizes the NRC requirements for SPDS functions and how the CR-3 SPDS can be used to assess those functions. "Alert" signals to control room personnel allow them to assess the reactivity control, radioactivity control and containment conditions functions. Pressure-temperature information on P-T displays allows control room personnel to monitor for abnormal symptoms regarding subcooling and heat transfer for the reactor core cooling and heat transfer from the primary system function. This same pressure-temperature information allows the control room personnel to monitor against key pressure-temperature limits for the reactor coolant system integrity function.

The events analyzed for the ATOG program are the basis for the CR-3 SPDS. They include a wide range of events of low to moderate frequency of occurrence. They are representative events which provide all the necessary symptoms for which the SPDS and the ATOG are designed to monitor and control. The key parameters chosen for the CR-3 SPDS to meet the above requirements are summarized in Table 2.1. of this report. Based on the information provided in this report, it is demonstrated that the CR-3 SPDS provides control room personnel with sufficient information to enable them to determine the safety status of the plant for a wide range of abnormal and emergency conditions. In addition, it is demonstrated that the CR-3 SPDS provides sufficient information to be used in conjunction with ATOG-type procedures to detect abnormal symptoms and to allow corrective actions necessary to restore the control function or mitigate the consequences of transients and accidents in a rapid and reliable manner.

1.0 INTRODUCTION

1.1. Purpose

The purpose of this report is to provide a written safety analysis for the Crystal River Unit 3 (CR-3) Safety Parameter Display System (SPDS) for describing the basis on which the selected parameters are sufficient to assess the safety status of the plant for a wide range of events which include symptoms of severe accidents.

This report will summarize the NRC requirements for an SPDS and will demonstrate that the CR-3 SPDS meets those NRC requirements. In addition to these NRC requirements, another basis for the CR-3 SPDS is the B&W Owners Group Abnormal Transient Operating Guidelines (ATOG) program. This report will summarize that program, particularly the events analyzed, and demonstrate the compatibility of the CR-3 SPDS with ATOG. Finally, this report will identify the parameters selected for the CR-3 SPDS and the method of their presentation to control room personnel.

This report is in response to the requirement for such a document as contained in Section 4.2 of NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability" (Generic Letter No. 82-33), dated December 17, 1982.

1.2. Background

As a result of the TMI-2 accident, the NRC issued an action plan for items to be addressed in order to correct or improve the regulation and operation of nuclear facilities. That plan was provided in NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident", dated May 1980. In the area of operational safety, it was concluded in every major study of the accident that insufficient attention had been given to ensuring compatibility between control room personnel and the systems they are required to operate. The variety and quantity of information displayed in the control room can often be overwhelming, especially during transient operations or an accident. It has been determined that a concise display of those parameters necessary to assess the safety status of the plant would significantly aid control room personnel in determining plant status and diagnosing accidents. Item 2 of Task I.D of NUREG-0660 established the requirements for a plant safety parameter display console to be installed in the control room.

In November 1980, the NRC issued NUREG-0737, "Clarification of TMI Action Plan Requirements." Concerning the requirements for a plant safety parameter display console, the NRC stated that issuance of NUREG-0696, "Functional Criteria for Emergency Response Facilities" would provide those requirements for an SPDS. NUREG-0696 was issued in February 1981. In December 1982, NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability (Generic Letter No. 82-33)" was issued. NUREG-0737, Supplement 1, describes the NRC requirements for SPDS. The CR-3 SPDS will be designed to those requirements. Section 5 of NUREG-0696 provides guidance for the design of the SPDS including its purpose, location, size, display considerations and design criteria, such as conformance to Regulatory Guide 1.97 for the parameters chosen. These guidelines serve as a basis for the design requirements for the CR-3 SPDS.

The purpose of the CR-3 SPDS is to assist the control room personnel during abnormal and emergency conditions in the evaluation of the safety status of the plant and assessing whether abnormal conditions warrant corrective actions, to avoid a degraded core. Located in the control room, the SPDS will be continuously available during all modes of plant operation including cold or refueling shutdown, heatup and cooldown operations, normal power operations, as well as abnormal or emergency conditions. The CR-3 SPDS video monitors are not seismically qualified and thus, the SPDS may not be available after certain seismic events. Plant parameters which indicate very abnormal conditions such as inadequate core cooling (ICC) shall also be displayed. The CR-3 SPDS is designed to use a minimum number of displays and selected parameters, yet it will be able to concisely present to the operator information concerning the safety status of the following functions as required by NUREG-0737 and NUREG-0696:

(1) Reactivity control

(2) Reactor core cooling and heat removal from the primary system

(3) Reactor coolant system integrity

(4) Radioactivity control

(5) Containment conditions.

Finally, the CR-3 SPDS is designed to be compatible with the control room personnel's training and experience in normal and abnormal or emergency operation. Florida Power Corporation (FPC) is a member of the B&W Owners Group Operator Support Subcommittee which has worked to develop a program to respond to Item I.C.1 of NUREG-0737, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents". That program resulted in the development of the Abnormal Transient Operating Guidelines (ATOG) which employs the concept of symptom-oriented rather than event-oriented procedures for handling abnormal or emergency conditions. The design of the displays for the CR-3 SPDS has incorporated the basic concepts of ATOG in order to present sufficient information to control room personnel for monitoring tho status of the plant during normal, transient and abnormal conditions, diagnosing symptoms and taking corrective actions to achieve stable plant conditions.

The CR-3 SPDS is designed to serve as an aid to the control room personnel and can be used by them to quickly focus on certain key parameters. However, the SPDS should not be used as the exclusive source of information to determine plant status or to monitor control actions. Other instrumentation available in the control room should also be used by control room personnel to determine plant status.

The CR-3 SPDS will be located in the control room. The CR-3 SPDS will provide information to control room personnel via selectable displays and will automatically display alert signals. The selectable displays include the Low-Range Pressure-Temperature (P-T) display, ATOG P-T display, Inadequate Core Cooling (ICC) display, two "normal" power operation displays and a four page alphanumeric display of several key parameters and safety systems actuation status. Control room personnel can select any available display by a single pushbutton on the SPDS control panel. If control room personnel have selected either the Low-Range P-T display or the ATOG P-T displays, they can add any of several P-T limit curves to the base display by a single pushbutton for each limit curve. This feature allows the control room personnel to be able to monitor the plant status versus those P-T limits applicable to the current plant conditions or expected plant evolutions such as plant cooldown to cold shutdown conditions. Each control panel also contains toggle switches which allows control room personnel to select either Loop A or Loop B parameters for display on either CRT and to select spare parameter inputs. There is a feature to the CR-3 SPDS which allows the control room personnel to monitor the history of the RCS pressure-temperature relationship. Using a single pushbutton, control room personnel can demand or erase the history trace of the RCS pressure-temperature relationship. Finally, control room personnel can select the incore thermocouple temperature (average of the five highest thermocouples) to be displayed instead of T-hot on the two P-T displays.

The two "normal" power operation displays, "Normal" and "Flux/Imbalance" can be used by the control room personnel during normal operations to monitor plant conditions against the Reactor Protection System (RPS) P-T and Flux/Flow/Imbalance limits. The Low-Range P-T and ATOG P-T displays can be monitored by control room personnel during heatup, cooldown and normal operations and also during transients and accidents. The ATOG P-T display will be automatically displayed for each RCS loop upon receipt of a reactor trip signal. This feature prevents control room personnel from having to manually switch to these displays in the first several seconds following the reactor trip, freeing them to perform their immediate post-trip actions. The ICC display can be monitored by control room personnel during conditions of RCS saturation or superheat. The six "alert" signals will automatically flash on any selected display if conditions warrant. The four page Alphanumeric display can be used by control room personnel to obtain additional supporting information to the above displays and alerts.

A detailed description of the CR-3 SPDS hardware, displays, alerts, and input signals can be found in the document "CR-3 Safety Parameter Display System Functional Description", Dwg. #1147047. Algorithms for the display curve limits can be found in the document, "Crystal River Unit 3 Safety Parameter Display System (SPDS) Displays", Doc. ID #51-1121942, dated June 1984.

The SPDS displays have been Human Factors engineered to the extent that they present the necessary information to control room personnel in a manner which would allow them to carry out the required actions for abnormal transient operations described in ATOG. This includes displays which have a simple, uncluttered format that can be called up by a single push button command. The displays are continuously updated and allow control room personnel to track trends for key variables. Additional information is presented to control room personnel in order for them to assess the status of the plant with respect to the five critical safety functions. This information is presented in the form of Alerts in order that they not interfere to a significant extent with basic pressure-temperature displays required by ATOG. Further explanation of these features is found in Section 2.0.

2.0 SPDS FUNCTIONS/PARAMETERS MONITORED

Section 1.2 describes the purpose of the SPDS. In order to meet that purpose, the SPDS must be designed to provide information to control room personnel for all modes of normal operation from refueling to power operation. It must also be designed to provide key information to control room personnel during abnormal and emergency conditions so that they can assess the safety status of the plant. In particular, it must provide sufficient information to allow control room personnel to assess the safety status of the plant with respect to five major functions.

Each of the five safety status functions identified is Section 1.2 is discussed in detail below. Each section will include a brief description of what the safety function is followed by a discussion of how information concerning the safety function is provided to control room personnel by the CR-3 SPDS. In addition to the specific displays and alerts which are described in these discussions, it is important to note that a four-page Alphanumeric display is available to provide support information to the other displays and alerts.

The complete list of parameters required to monitor the five SPDS safety functions is provided in Table 2.1. Also included in this table are the signal input string, range, the function it monitors, the means of presentation (display or alert signal) plus a summary discussion. In addition to these parameters, there are many additional parameters available on the Alphanumeric display to aid control room personnel in controlling the plant.

2.1. Reactivity Control

The first function is reactivity control. Following a reactor trip, or when conditions warrant a reactor trip, control room personnel need to know that the control rods have all inserted and that the reactor is subcritical. If not, they must begin taking required actions to achieve reactor subcriticality.

On the CR-3 SPDS, the reactivity control function is monitored by a "Reactivity" alert logic. The parameters monitored for this alert are control rod group in-limit position, source range nuclear instrumentation flux level and reactor trip signal. These parameters are sufficient for monitoring the reactivity control function since they provide the necessary information to control room personnel that control rods have inserted upon reactor trip and that the reactor is subcritical at all times other than when the reactor is already critical or where a planned reactor startup is underway.

2.2. Reactor Core Cooling and Heat Removal from the Primary System

The second function is reactor core cooling and heat removal from the primary system. This function will be discussed as two separate functions. During all modes of operation and following reactor trip, control room personnel must have information to determine the thermal-hydraulic state of the reactor coolant. If the reactor coolant is in a liquid state and subcooled, control room personnel are assured that reactor coolant is available and capable of removing heat from the reactor core and transferring it to the steam generators. If subcooling is lost, these capabilities are in doubt and control room personnel can begin taking actions to restore subcooling.

On the CR-3 SPDS, the reactor core cooling function is monitored in four ways. During normal operation, this function is monitored by the display of the RCS pressure-temperature status points with respect to the RPS pressure-temperature limits which inherently insures that the RCS is in a subcooled state since the RCS would have to pass through these trip parameters before subcooled margin is lost. During post-trip operations, this function is monitored by the display of the RCS P-T status with respect to "expected" post-trip pressure and temperature limits as well as the saturation line and the subcooled margin limit which accounts for possible errors for the pressure and temperature instrument strings. This information is available to control room personnel on the ATOG Display. During inadequate core cooling operations, this function is monitored by the display of the RCS P-T status with respect to saturation and two fuel cladding temperature limits. This information is available to control room personnel on the ICC Display. During cooldown operations, this function is monitored on the Low-Range P-T Display in like manner to the post-trip operations discussed above.

In each of these situations, the parameters to be monitored are RCS pressure, RCS hot leg temperature (T-hot), RCS cold leg temperature (T-cold), and average incore thermocouple temperature. When displayed against saturation and subcooled margin, these are the only parameters that are necesasry to determine the state of the RCS for monitoring the reactor core cooling function in these four situations.

In addition to reactor core cooling, heat removal from the primary system must be monitored. Control room personnel must have information which allows them to monitor the heat transfer coupling between the reactor coolant system and the steam generators. Control room personnel can then begin taking corrective actions to restore that coupling if inadequate heat transfer is occurring. If excessive heat transfer is occurring, control room personnel can begin taking actions to restore the balance between the heat source (reactor core) and heat sink (steam generators).

On the CR-3 SPDS, the heat removal from the primary system function is monitored by the ATOG P-T Display and the "EFW Alert". The RCS pressure-temperature status is displayed as a point and the steam generator saturation temperature based on its pressure is displayed as a vertical line. Five parameters (RCS pressure, T-hot, T-cold, average incore thermocouple temperature and steam generator pressure), provide the symptoms of inadequate or excessive heat transfer and allow verification of natural circulation by control room personnel. The CR-3 SPDS also provides bar charts for steam generator startup and operate range level indication to aid in a more rapid recognition of the problem and the "EFW Alert" which warns control room personnel that a signal from the Emergency Feedwater Instrumentation and Control (EFIC) system is present to initiate EFW.

2.3. Reactor Coolant System Integrity

The third function is reactor coolant system integrity. During all modes of normal operation and during transients and accidents, control room personnel must be able to monitor the status of the reactor coolant system pressure-temperature relationship against several important pressure and pressure-temperature limits to ensure the integrity of the reactor coolant system against overpressurization, pressurized thermal shock or reactor vessel NDT. If limits are being approached or exceeded, control room personnel should begin taking actions required to re-establish an acceptable state. Also, there are a number of pressure, temperature, and pressure-temperature limits important to normal operation. Control room personnel must have the necessary information to determine the status of the plant with respect to these limits and control the plant within these limits.

On the CR-3 SPDS, the reactor coolant system integrity function is monitored using the following parameters: RCS pressure, T-hot and T-cold. These parameters are displayed as P-T status points against the following applicable RCS P-T or pressure limits: RCS design pressure, DHRS design pressure, thermal shock limit, heatup NDT and cooldown NDT limits. Operation of the plant within these limits will satisfy the reactor coolant system integrity function.

2.4. Radioactivity Control

The fourth function is radioactivity control. During all modes of normal operation and especially during accident conditions, control room personnel must be warned of the presence of high radioactivity in the plant and monitor release paths to determine the extent of radioactivity releases from the plant. They can then take necessary actions to prevent or minimize such releases.

On the CR-3 SPDS, the radioactivity control function is monitored by the "Radioactivity Alert". The parameters to be monitored by this alert are the fourteen (14) radiation monitors identified in Table 2.1. These monitors were chosen because they monitor the major radiation release paths from the plant or because they provide specific information to aid control room personnel in determining the event or identifying the location of the radiation leak. More specific information on these radiation monitors is found in the "Discussion" column of Table 2.1.

2.5. Containment Conditions

The final function is containment conditions. The last fission product barrier to the environment is the reactor building. Control room personnel must know the status of the reactor building in order to take necessary actions to ensure or restore its integrity during accident conditions.

On the CR-3 SPDS, the containment conditions function is monitored by three sets of alert logic. They are "Reactor Building Pressure Alert", "Radiation Alert", and "Engineered Safeguards Actuation Alert". Reactor building pressure and radiation are the two parameters which give the first indication of a major accident, such as a LOCA or steam line break, inside containment. The "Reactor Building Pressure Alert" warns of increasing pressure in the RB in advance of RB isolation. The "Radiation Alert" contains inputs from three radiation monitors located in the RB dome, RB purge duct and RB vent duct. If the "Radiation Alert" is present, control room personnel can request the applicable page of the Alphanumeric display to determine if one of the RB monitors is causing the alert. If the "Engineered Safeguards Actuation Alert" is present, control room personnel can page the Alphanumeric display to determine if the RB isolation signal is causing the alert.

TABLE 2.1

PARAMETERS REQUIRED TO MONITOR THE FIVE SPDS SAFETY FUNCTIONS

PARAMETER	SIGNAL INPUT	RANGE	FUNCTION[1,*]	DISPLAY OR ALERT[2,*]	DISCUSSION
1. T-Cold A&B WR	RC5A-TT2,-TT4 RC5B-TT2,-TT4	50 to 650°F	1.2, 1.3	2.1, 2.2, 2.4, 2.9, 2.11	Plotted and displayed with RCS pressure on P-T displays to monitor plant status against NDT, thermal shock, fuel compression and RCP-NPSH limits, and to provide symptoms of excessive or inadequate heat transfer when used with SG saturation temperature.
2. T-Hot A&B WR	RC4A-TT1,-TT4 RC4B-TT1,-TT4	120 to 920°F	1.2, 1.3	2.1, 2.2, 2.4, 2.11	See discussion of T-Cold A&B. In addition, it is displayed against subcooling margin and saturation limits for symptoms of inadequate core cooling. It is also used with Incore Thermocouple Temperature as an indication of natural circulation.
3. RCS Pressure A&B WR	RC3A-PT3, RC-158PT RC3B-PT3, RC-159PT	0 to 2500 psig	1.2, 1.3	2.1, 2.2, 2.3, 2.4, 2.9, 2.11	See discussion of T-Cold A & B. Also plotted with Incore Thermocouple Temperature to monitor natural ciruclation or superheated conditions during ICC.
4. RCS Pressure A&B Low Range	RC-131-PT, RC-131-PT1, RC-147PT, RC-148PT1	0 to 500 psig 0 to 600 psig	1.2, 1.3	2.1, 2.4, 2.9	Used as RCS Pressure signal during low RCS Pressure operation.

TABLE 2.1 (CONT'D)

PARAMETER	SIGNAL INPUT	RANGE	FUNCTION[1,0]	DISPLAY OR ALERT[2,0]	DISCUSSION
5. In core Thermocouple Temperature (calculated average of 5 highest readings from 15 T/Cs)	IM-08H, -06F -09M, -13F, -09G, -12O, -07E, -05F, -07M, -05D, -03F -05O	0-2000°F	1.2	2.1, 2.2, 2.3	Average value plotted with RCS Pressure on ICC curve during ICC conditions and can be plotted with RCS Pressure on other P-T curves. (See discussion of RCS Pressure, WR).
6. OTSG Op. Range	SP1A-LAM 1/2 SP1B-LAM 1/2	0 to 100%	1.2, 1.3	2.1, 2.2, 2.3 2.4	Displayed on P-T displays primarily for immediate detection of excessive feedwater and loss of feedwater events.
7. OTSG SU Range A&B	SP1A LT4/5 SP1B-LT4/5	0 to 250 in.	1.2, 1.3	2.1, 2.2, 2.3	See discussion on OTSG Op. Range A&B. SU Range may be useful in SG tube rupture identification.
8. OTSG Pressure A&B	SP6A-PT 1/2 SP6B-PT 1/2	0 to 1200 psig	1.2	2.2, 2.4	Used as input to SG saturation temperature on Post-Trip ATOG display. Used to display symptoms of excessive heat transfer and inadequate heat transfer when used with T-Cold.

TABLE 2.1 (CONT'D)

PARAMETER	SIGNAL INPUT	RANGE	FUNCTION(1.0)	DISPLAY OR ALERT(2.0)	DISCUSSION
9. Source Range NI A&B	NI-01, NI-02	10^{-1} to 10^{6} cps	1.1	2.4, 2.5	Used in conjunction with reactor trip signal and CRG positions to alert the control room operator to high or unexpected neutron count rate.
10. All Rods IN	(Later)	No/Yes	1.1	2.5	See discussion of source range NI A&B.
11. Reactor Tripped	Rx Patch Pnl	No/yes	1.1	2.2, 2.4, 2.5, 2.10	See discussion of suorce range NI A&B. In addition, this signal also initiates the automatic display the Post-Trip ATOG display for loops A&B on the two CRTs and the RPS actuation alert.
12. Radiation Monitors			1.4	2.4, 2.6	All the radiation monitors are used to monitor potential major release paths. In addition, the individual monitors are used to:
a. RB Dome	RMG-19	5 to $5x10^{5}$ R/hr			a. Help discriminate between major LOCA and Steam Line Break.
b. RB Purge Duct	RMA-1	10 to 10^{6} CPM			b. Provides early indication of vent system radioactivity and possible unplanned release.
c. Aux. Fuel Handling Duct	RMA-2	10 to 10^{6} CPM			c. Same as b.
d. RB Vent Duct	RMA-6	10 to 10^{6} CPM			d. Provides early indication of RCS leakage.

TABLE 2.1 (CONT'D)

PARAMETER	SIGNAL INPUT	RANGE	FUNCTION[1,9]	DISPLAY OR ALERT[2,9]	DISCUSSION
e. Condenser Vacuum Pump Discharge	RMA-12	10 to 10^6 CPM			e. Provides indication of steam generator tube leakage.
f. Main Steam Line	RMG-25	10^{-1} to 10^7 mR/hr			f. Will identify affected steam generator when a tube leak is present.
g. Main Steam Line	RMG-26	10^{-1} to 10^7 mR/hr			g. Same as f.
h. Main Steam Line	RMG-27	10^{-1} to 10^7 mR/hr			h. Same as f.
i. Main Steam Line	RMG-28	10^{-1} to 10^7 mR/hr			i. Same as f.
j. Primary Coolant	RML-1	10 to 10^6 CPM			j. Provides early indication of fuel cladding failures.
k. Plant Discharge	RML-2	10 to 10^6 CPM			k. Provides early indication of abnormalities in the liquid effluent discharge flow path.
l. Turbine Building	RML-7	10 to 10^6 CPM			l. Same as k.
m. Control Room Vent	RMA-5	10 to 10^6 CPM			m. Provides indication of degrading control room environment prior to actuations occurring.
n. Control Room	RMG-1	10^{-1} to 10^4 mR/hr			n. Same as m.
13. Reactor Building	BS90-PT, BS91-PT	0 to 280 psia	1.5	2.4, 2.7	Used in Reactor Building Pressure alert to warn the operator of high pressure conditions as a result of LOCA or steam leak.

TABLE 2.1 (CONT'D)

PARAMETER	SIGNAL INPUT	RANGE	FUNCTION(1,0)	DISPLAY OR ALERT(2,0)	DISCUSSION
14. ES Channels Tripped	18 channels	No/Yes	1.2, 1.5	2.4, 2.8	The 18 ES channels are monitored such that 2 out of 3 tripped in any of the . groups will cause alert signal to warn the control room operator of ES actuation.
15. EFIC Actuation	(later)	No/Yes	1.2	2.4, 2.9	Used in the EFW Actuation Alert to warn the operator that the EFIC signal to initiate EFW is present.

1.0 SPDS Function

- 1.1 Reactivity Control
- 1.2 Reactor Core Cooling and Heat Removal from Primary System
- 1.3 Reactor Coolant System Integrity
- 1.4 Radiation Control
- 1.5 Containment Conditions

2.0 Display or Alert

- 2.1 Low Range P-T Display
- 2.2 ATOG P-T Display
- 2.3 Inadequate Core Cooling (ICC) Display
- 2.4 Alphanumeric Display
- 2.5 Reactivity Alert
- 2.6 Radiation Alert
- 2.7 Reactor Building Pressure Alert
- 2.8 Engineered Safeguards Actuation Alert
- 2.9 Emergency Feedwater Actuation Alert
- 2.10 Reactor Trip (RPS Actuation) Alert
- 2.11 Normal Display

3.0 APPLICABLE EVENTS

3.1. Introduction

In part, the CR-3 SPDS is based on ATOG. Control room personnel must be provided with certain key parameters which they can monitor in conjunction with the ATOG-based emergency operating procedures. The key parameters are those necessary for the identification of the three basic heat transfer symptoms (lack of subcooling, inadequate heat tranfer or overheating, and excessive heat transfer or overcooling) and the special case of steam generator (SG) tube rupture. Once control room personnel identify one or more of these symptoms, they can begin taking action to regain the control functions which are not being controlled. The control functions associated with these symptoms are RC inventory, RC pressure, SG inventory, and SG pressure control.

The key parameters which are necessary for the identification of these four symptoms are also those necessary to assess the safety status of the plant with respect to two of the five required SPDS functions: reactor core cooling and heat transfer from the primary system and reactor coolant system integrity which were discussed in Section 2.0. Additional parameters are provided in order to assess the safety status of the plant with respect to the remaining three required SPDS functions: reactivity control, radioactivity control and containment conditions. Expected control room personnel actions to verify that all control rods have inserted and that measured neutron flux levels are decreasing are the basis for the two parameters necessary for the "Reactivity Alert". Expected control room personnel actions to monitor Reactor Building pressure and radiation monitors are the basis for the parameters necessary for the "Reactor Building Pressure" and "Radiation" alerts. Expected control room personnel actions to determine an event and to monitor key radiation release paths during accident assessment are the bases for the selection of the fourteen (14) Radiation Monitors used in the "Radiation Alert".

In order to better understand the bases for the parameter selection for the CR-3 SPDS, it is important to understand the events that were analyzed for developing the ATOG guidelines.

The ATOG guidelines have been written to cover a very large number of abnormal transient scenarios. This is an inherent benefit of a symptom-oriented approach to operation. However, specific accident events need to be analyzed in-depth to demonstrate that they can be handled using a symptom-oriented approach. These transients can include classic single initiating events as well as additional single or multiple failures.

For the ATOG development program, six initiating events were identified and analyzed in detail. The six initiating events selected were:

1. Excessive feedwater
2. Loss of main feedwater (LOFW)
3. Steam generator tube rupture (SGTR)
4. Loss of offsite power (LOOP)
5. Small steam leak
6. Small break loss of coolant accident (SBLOCA).

These six events were chosen as representative transients, each of which would provide one or more of the basic symptoms on which ATOG is based. Additionally, four specific criteria were used in selecting these six events:

1. Moderate frequency events in which operator action is expected (LOFW, Excessive FW, LOOP).
2. Low probability events that can be confusing and difficult to recognize and mitigate (SGTR, SBLOCA).
3. The events cover a very large spectrum of conditions in the RCS (overheating, overcooling, loss of inventory, loss of subcooling).
4. Time exists for the operator to recognize and do something about the accident or event; therefore, guidelines are appropriate. This would necessarily exclude large, rapid design basis events as such a large break LOCA.

For each of the six initiating events chosen, a detailed event tree was developed to identify a main success path and major failure paths. Analyses using the TRAP code were performed on the main success path and most single failure paths for the ATOG base plant, ANO-1. The ANO-1 plant design, equipment, pertinent plant setpoints, etc. were then compared to CR-3. Transient Information Documents (TIDs) were then developed for use to modify the ANO-1 ATOG guidelines to fit CR-3. Those documents applicable to CR-3 are References 10 through 16. Additional analyses have been performed for the Steam Generator Tube Rupture event. Those analyses are documented in Reference 17. The LOCA events analyses and recommendations were derived largely from those efforts which produced the "Small Break Operating Guidelines".

A brief summary of each of the six initiating events is provided below. Detailed discussions of these events are found in the Part II - Vol. 2 of Reference 5 and Reference 17. Those discussions include actual operating events which have occurred.

3.2. Excessive Feedwater Event

The excessive feedwater event was analyzed for two cases. For the first case, the excessive feedwater addition was terminated by the ICS after reactor trip. The event was analyzed to >200 seconds at which time the RCS had stabilized in a hot shutdown condition. For the second case, the excessive feedwater addition to one SG was terminated by control room personnel approximately 5.5 minutes after event initiation. Prior to termination of feedwater, the overfed SG has filled, the pressurizer has emptied due to RCS contraction, subcooling margin has been lost and ESAS has started HPI and EFW. In addition to terminating main feedwater, control room personnel must make three significant control actions during such an event. The first is to trip all operating RCPs on loss of subcooling margin. The second is to throttle EFW to obtain a gradual increase in SG level to prevent worsening the overcooling. The third is to throttle HPI once subcooling margin is restored. It should be noted that once the Emergency Feedwater Initiation and Control (EFIC) system is implemented at CR-3, control room personnel will only have to verify that main feedwater has been terminated and that emergency feedwater flow is being controlled properly to achieve the desired SG level.

The pressure-temperature relationship response of the primary and secondary systems displayed on the CR-3 ATOG P-T Display will provide control room personnel the symptom of excessive heat transfer. In addition, the two SG level bar charts will indicate the excessive feedwater addition. This information will key control room personnel to take the required actions to terminate main feedwater and to take the three additional control actions.

3.3. Loss of Main Feedwater Event

The second event analyzed was the loss of main feedwater (LOFW) event. This event was initiated by a trip of both main feedwater pumps from 100% FP with anticipatory reactor trip occurring. In its early stages, this transient will look almost identical to a "normal" reactor trip. Control room personnel should be able to identify LOFW by indication of main feedwater pumps tripped or zero main feedwater flow rate at the feedwater control station, or rapidly falling SG levels on the SPDS display or existing control room instrumentation. The only significant control room personnel action is to verify that EFW starts and is providing flow to the SGs. If not, they should take manual control of EFW. This event was also run with a number of failures out to approximately 10 minutes after reactor trip.

The first failure is failure of EFW to start. Based on the pressure-temperature relationship response displayed on the SPDS display, control room personnel should recognize the symptom of inadequate primary to secondary heat transfer and take corrective actions to restore heat transfer. The second failure is EFW overfeed. Control room personnel should recognize the symptom of excessive primary to secondary heat transfer and take corrective actions to throttle EFW. As was previously noted, once EFIC has been implemented at CR-3, control room personnel should only have to verify that EFW has initiated and is being controlled to achieve the desired SG level. The third failure is steam leakage from a stuck open SG steam safety valve or an open turbine bypass valve. Control room personnel should recognize the symptom of excessive primary to secondary heat transfer, and by following plant procedures, they should identify the steam leak and take corrective actions.

The pressure-temperature relationship response of the primary and secondary systems displayed on the CR-3 ATOG P-T Display and the two SG level bar charts will provide control room personnel the symptom of inadequate heat transfer. This information should key control room personnel to take the required actions.

3.4. Steam Generator Tube Rupture (SGTR) Event

A steam generator tube rupture (SGTR) is a loss of coolant accident (LOCA) through the secondary plant. It is an event which can contaminate the secondary plant and can lead to radiation releases if steam from the affected SG is released to the atmosphere. For these reasons, it is important that control room personnel identify the event and the affected SG quickly.

The analysis performed for ATOG began with a SGTR occurring at 100% FP. The initial primary to secondary leak rate was for a double-ended rupture of a single tube (approximately 400 gpm). In the first several minutes, control room personnel will see RCS pressure and pressurizer level decreasing. They will also receive steam line and condenser air ejector off gas radiation alarms. Because the leak rate exceeds the MU system capability, control room personnel must quickly take action to increase makeup flow or manually actuate HPI and terminate letdown flow in order to stabilize the RCS. Once the RCS pressure and pressurizer level are stable, the reactor can be shutdown in a controlled manner. The controlled shutdown of the reactor should prevent lifting the SG steam safety valves. Once the reactor is shutdown, control room personnel can cooldown and depressurize the RCS while maintaining minimum subcooled margin in order to minimize the primary to secondary leak rate.

However, if control room personnel take no action to increase makeup and because the leak rate exceeds MU system capability, the reactor will trip on low pressure. Following the reactor trip, subcooling margin is lost, requiring control room personnel to trip all operating RCPs. ESAS will actuate on low RCS pressure starting HPI and EFW. At approximately 11 minutes into the transient, the pressurizer will empty and the RCS will become saturated. In the next 4 to 5 minutes, control room personnel will throttle EFW to prevent overcooling. After EFIC implementation, they will verify proper EFW control. They will also throttle HPI to stabilize RCS pressure and pressurizer level following the re-establishment of subcooling margin. Once the plant has been stabilized with core decay heat being removed by the SGs via natural circulation, control room personnel can restart RCPs and initiate plant cooldown and depressurization. During the cooldown, control room personnel should maintain a minimum subcooling margin to keep RCS pressure and the primary to secondary leak rate low.

Secondary system radiation alarms in conjunction with the pressure-temperature relationship response of the primary and secondary systems plus the SG level bar charts on the CR-3 ATOG P-T Display will provide information to control room personnel to identify the SGTR event and the affected SG. They can then begin to take required actions to mitigate the consequences of this event.

3.5. Loss of Offsite Power (LOOP)

The analysis for this event begins at 100% FP with the plant separation from the grid. During the reactor and turbine runback, the reactor trips on high pressure. The diesel generators start on detection of undervoltage on the engineered safeguard buses and begin loading within approximately 10 seconds. Secondary steam pressure is controlled by the steam safeties or atmospheric dump valves. Since the RCPs have tripped, natural circulation will be established and EFW will start feeding to establish level in the SGs. That level is reached approximately 8 minutes after grid separation occurs, and the RCS should be stabilized at normal hot shutdown conditions. After EFIC implementation, the EFW flow will be automatically controlled to establish natural circulation level setpoint.

Additional analyses were performed for a total loss of all power except station batteries. For a short period of time, the plant can be controlled at hot shutdown using EFW to the SGs to provide a heat sink. If voids begin to form in the RCS and natural circulation is interrupted, control room personnel should fill the SGs to the required level to establish the core boiling/SG condensing mode for decay heat removal. EFIC will automatically control SG levels to the LOCA setpoint once control room personnel selects that mode.

Every effort must be made to get the diesel generators started. Since the pressurizer heaters and makeup pumps have lost power, control room personnel have no control of primary pressure or inventory. The plant cannot be cooled down because water cannot be added to make up for RCS contraction during cooldown.

The pressure-temperature relationship response of the primary and secondary systems displayed on the CR-3 ATOG P-T Display will provide control room personnel information on the adequacy of natural circulation and the symptom of inadequate heat transfer if natural circulation is interrupted.

3.6. Small Steam Leak

The event analyzed was a failure of the turbine bypass system. This failure results in a steam leak equivalent to approximately 15% of full power steam flow and occurs with an assumed maximum decay heat. In this event, the reactor trips on high flux in about 9 seconds following full opening of the turbine bypass system valves. The turbine trips and ICS runs back main feedwater. The turbine bypass valves on one SG remain open. Approximately three minutes into the transient, the pressurizer drains due to excessive RCS contraction. ESAS initiates due to low RCS pressure and starts HPI and EFW. As RCS pressure and pressurizer level are restored, control room personnel begin throttling HPI. Approximately 5 minutes into the transient, control room personnel identify the affected SG based on symptoms of excessive heat transfer and decreasing steam pressure in the affected SG and allow it to boil dry. At approximately 8 minutes, control room personnel stop HPI, realign to normal makeup/letdown mode and use the atmospheric dump valves on the unaffected SG to stabilize the RCS.

Larger leaks, less decay heat or subsequent failures may cause a transient severe enough to cause loss of subcooling margin and possibly RCS saturation. Control room personnel must follow their procedures for loss of subcooling margin. The steam leak may also be inside the reactor building (RB). If so, it is likely that an ESAS trip on high RB pressure will occur.

Steam leaks and other similar events will exhibit the symptom of excessive heat transfer on the CR-3 ATOG P-T Display. Control room personnel can begin taking the required actions to mitigate the consequences of this event.

3.7. Loss of Coolant Accident

The loss of coolant accident (LOCA) is a complex and difficult accident to handle for the following reasons:

1. A wide range of leak sizes is possible from small breaks in instrument lines to large breaks in the RCS piping,
2. A wide range of break locations is possible and the RCS response can depend on break location,
3. Abnormal system condition such as RCS saturation, interrupted natural circulation, etc. may be a natural consequence of the event,

4. Steam generator heat removal may be degraded,
5. Hot standby conditions may not be a safe, stable condition,
6. The reactor building environment will degrade due to increased pressure, temperature, humidity, and radiation.

LOCAs can be categorized by size. Large break LOCAs result from a major failure in the primary system pressure boundary which depressurizes the RCS rapidly and almost completely. Since this is a design basis event for which extensive analysis is performed in the FSAR, it was not analyzed for ATOG. Small break LOCAs are less severe than large breaks. The RCS depressurization is much slower due to the lower mass flowrate out the break. The Emergency Core Cooling System (ECCS) should maintain adequate core cooling throughout the transient. For small breaks, control room personnel plays a vital role in minimizing the consequences of the accident. Small leaks are events where the loss of reactor coolant is within the capacity of the normal makeup system.

The primary objectives during a LOCA are to maintain core cooling, to cooldown and depressurize the RCS, and to establish a stable, long-term cooling mode. Core cooling is maintained by the proper operation of the ECCS which control room personnel verify. In addition, they should make sure that the reactor building cooling is functioning properly and that long-term cooling is established once the RCS is cooled down and depressurized.

In order to determine actions required by control room personnel to monitor, verify or perform manually, and to identify the expected RCS and system transient response, five small break LOCAs were analyzed as part of the development of the Small Break Operating Guidelines and were translated into the ATOG guidelines. The five events analyzed were the following:

1. Small breaks that are large enough to depressurize the RCS below secondary system pressure with feedwater available.

2. Small breaks which stabilize at approximately the secondary system pressure with feedwater available.
3. Small breaks which may repressurize the RCS in a saturated condition with feedwater available.
4. Small breaks without primary to secondary heat transfer.
5. Small breaks within the pressurizer steam space.

These events were analyzed in sufficient detail to generate symptoms for identifying the type of break as well as distinguishing a LOCA from other events, especially overcooling events.

Reactor Building Pressure and Radiation alerts on the CR-3 SPDS in conjunction with the pressure-temperature relationship response of the primary and secondary systems on the CR-3 ATOG P-T Display will provide the information necessary for control room personnel to identify a LOCA event. They can then begin to take the actions required by their procedures to control the plant and mitigate the consequences of the event.

Additional analyses have been performed in order to generate the subcooling margin limits to be displayed on the two SPDS P-T displays and to generate the inadequate core cooling limits to be displayed on the SPDS ICC display. Those limits and alogrithms are provided in Reference 8. Additional limits such as RCP NPSH, heatup and cooldown NDT, DHRS, and fuel compression were already available and no further analyses were performed. These limits and algorithms are also provided in Reference 8 and are available on the CR-3 SPDS P-T displays.

4.0 CONCLUSIONS

The Crystal River Unit 3 SPDS has been designed to provide control room personnel with sufficient key information to enable them to determine the safety status of the plant with respect to the five required functions discussed in Section 2.0. The CR-3 SPDS has also been designed to provide sufficient information in a display format based on and compatible with the symptom-oriented guidelines developed in ATOG as discussed in Section 3.0. With this key information and concise display format, control room personnel at CR-3 can monitor plant status, detect key symptoms of abnormal plant response and take corrective actions necessary to restore control function or mitigate the consequences of transients and accidents in a rapid and reliable manner.

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