FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302/LICENSE NO. DPR-72
REQUEST NO. 82, SUPPLEMENT NO. 3
NUREG 0737 REQUIRED TECHNICAL SPECIFICATIONS

LICENSE DOCUMENT INVOLVED: Technical Specifications Change Request No. 82, Attachment C.

PORTION:

- I. Safety Grade Anticipatory Reactor Trips
- II. Containment Isolation Valves
- III. Reactor Coolant High Point Vents
- VIII. Reactor Building Purge Supply and Exhaust Valves

DESCRIPTION OF REQUEST:

Revise the referenced portions of Attachment C to Technical Specification Change Request No. 82 to be consistent with recommendations presented by the staff in their October 1, 1984 letter to FPC specific changes are described in the enclosures.

REASON FOR REQUEST:

These changes are being made in response to NUREG-0737 and NRC recommendations to include these additional limitations in the Technical Specifications.

EVALUATION OF REQUEST:

All of the changes proposed herein involve additional limitations. Thus, this request will increase plant safety.

I. SAFETY GRADE ANTICIPATORY REACTOR TRIPS

Proposed Change

Change Technical Specificaton 3.3.1.1 and the Reactor Protection System Setpoints to include two new reactor trips. These new reactor trips are:

a. Anticipatory Reactor
   Trip-Main Turbine

b. Anticipatory Reactor
   Trip-Main Feedwater Pumps

These reactor trips are installed to trip the reactor in the event that the main turbine or both main feedwater pumps trip. This new specification requires that four channels be used to monitor the main turbine and four channels monitor the Main Feedwater Pumps. In the event that Reactor Power is greater than 20% RATED THERMAL POWER and two channels indicate a loss of the main turbine or both Main Feedwater Pumps, a reactor trip will result.

The main turbine is considered to be not operating when the turbine control oil pressure monitor indicates less than or equal to 45 psig. A main feedwater pump is considered to be not operating when the pump control oil pressure monitor indicates less than or equal to 55 psig.

Reasons for the Proposed Change

This change is being made in response to NUREG-0737, Section II.K.2.10 and Generic Letter 82-16, dated September 20, 1983. These new reactor trips are installed and operable.

Safety Analysis

Generic Letter 82-16 established guidelines for including the Main Turbine and Main Feedwater Pump Anticipatory Reactor Trip in the Technical Specifications. Where the guidelines do not fit the characteristics of Crystal River Unit 3, we have deviated from the recommendations. The main turbine anticipatory trip will not activate upon turbine stop valve closure, thus trip B, as described in Generic Letter 82-16, was omitted from this submittal.

Because these trip functions are bypassed during MODE 2, we have deleted MODE 2 applicability. This change is consistent with Generic Letter 82-16 recommendations.

Because Generic Letter 82-16 was intended for a Westinghouse PWR, the recommended ACTION statements are not applicable to Crystal River 3. The current ACTION statement 3 requirement is consistent with the required action for similar trip functional units.

TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTION UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8.	Pump Status Based on Reactor Coolant Pump Power Monitors (1)	More than one pump drawing $\leq$1152 or $\geq$14,400 kw	More than one pump drawing $\leq$1152 or $\geq$14,400 kw.
9.	Reactor Containment Vessel Pressure High	$\leq$ 4 psig	$\leq$ 4 psig
10.	Anticipatory Reactor Trip - Main Turbine (2)	Main Turbine Control Oil Pressure $>$ 45 psig	Main Turbine Control Oil Pressure $>$ 45 psig
11.	Anticipatory Reactor Trip - Both Main Feedwater Pumps (2)	Pump Control Oil Pressure $>$ 55 psig	Pump Control Oil Pressure $>$ 55 psig

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(1) Trip may be manually bypassed when RCS pressure $\leq$ 1720 psig by actuating Shutdown Bypass provided that:

a. The Nuclear Overpower Trip Setpoint is $\leq$5% of RATED THERMAL POWER.
b. The Shutdown Bypass RCS Pressure - High Trip Setpoint of $\leq$1720 psig is imposed.
c. The Shutdown Bypass is removed when RCS Pressure $>$1800 psig.

(2) Trip bypassed below 20% of RATED THERMAL POWER.

TABLE 3.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Manual Reactor Trip	1	1	1	1, 2 and *	8
2.	Nuclear Overpower	4	2	3	1, 2	2#
3.	RCS Outlet Temperature - High	4	2	3	1, 2	3#
4.	Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE	4	2(a)	3	1, 2	2#
5.	RCS Pressure - Low	4	2(a)	3	1, 2	3#
6.	RCS Pressure - High	4	2	3	1, 2	3#
7.	Variable Low RCS Pressure	4	2(a)	3	1, 2	3#
8.	Reactor Containment Pressure - High	4	2	3	1, 2	3#
9.	Intermediate Range, Neutron Flux and Rate	2	0	2	1, 2 and *	4
10.	Source Range, Neutron Flux and Rate					
	A. Startup	2	0	2	2## and *	5
	B. Shutdown	2	0	1	3, 4 and 5	6
11.	Control Rod Drive Trip Breakers	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#
12.	Reactor Trip Module	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#
13.	Shutdown Bypass RCS Pressure - High	4	2	3	2**, 3**, 4**, 5**	6#
14.	Reactor Coolant Pump Power Monitors	2 per pump	1 from 2 or more pumps (a)	2 per pump	1, 2	25
15.	Anticipatory Reactor Trip - Main Turbine	4	2(c)	3	1	3#
16.	Anticipatory Reactor Trip - Both Main Feedwater Pumps	4 per pump	2 per pump (c)	3 per pump	1	3#

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal.

** When Shutdown Bypass is actuated.

\# The provisions of Specification 3.0.4 are not applicable.

\#\# High voltage to detector may be de-energized above 10^{-10} amps on both Intermediate Range channels.

(a) Trip may be manually bypassed when RCS pressure less than or equal to 1720 psig by actuating Shutdown Bypass provided that:

(1) The Nuclear Overpower Trip Setpoint is less than or equal to 5% of RATED THERMAL POWER,
(2) The Shutdown Bypass RCS Pressure--High Trip Setpoint of less than or equal to 1720 psig is imposed, and
(3) The Shutdown Bypass is removed when RCS pressure greater than 1800 psig.

(c) Trip automatically bypassed below 20 percent of RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the control rod drive trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within one hour.

b. The Minimum channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per specification 4.3.1.1.

TABLE 3.3-2

REACTOR PROTECTION SYSTEM INSTRUMENTATION RESPONSE TIMES

	Functional Unit	Response Times
1.	Manual Reactor Trip	Not Applicable
2.	Nuclear Overpower *	≤ 0.266 seconds
3.	RCS Outlet Temperature - High	Not Applicable
4.	Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE *	≤ 1.79 seconds
5.	RCS Pressure - Low	≤ 0.44 seconds
6.	RCS Pressure - High	≤ 0.44 seconds
7.	Variable Low RCS Pressure	Not Applicable
8.	Pump Status Based on RCPPMs*	≤ 0.56 seconds
9.	Reactor Containment Pressure - High	Not Applicable
10.	Anticipatory Reactor Trip - Main Turbine	Not Applicable
11.	Anticipatory Reactor Trip - Both Main Feedwater Pumps	Not Applicable

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.	Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2.	Nuclear Overpower	S	D(2) and Q(7)	M	1, 2
3.	RCS Outlet Temperature--High	S	R	M	1, 2
4.	Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE	S(4)	M(3) and Q(7, 8)	M	1, 2
5.	RCS Pressure--Low	S	R	M	1, 2
6.	RCS Pressure--High	S	R	M	1, 2
7.	Variable Low RCS Pressure	S	R	M	1, 2
8.	Reactor Containment Pressure--High	S	R	M	1, 2
9.	Intermediate Range, Neutron Flux and Rate	S	R(7)	S/U(1)(5)	1, 2 and *
10.	Source Range, Neutron Flux and Rate	S	R(7)	S/U(1)(5)	2, 3, 4 and 5
11.	Control Rod Drive Trip Breaker	N.A.	N.A.	M and S/U(1)	1, 2 and *
12.	Reactor Trip Module	N.A.	N.A.	M	1, 2, and *
13.	Shutdown Bypass RCS Pressure--High	S	R	M	2**, 3**, 4**, 5**
14.	Reactor Coolant Pump Power Monitors	S	R	M	1, 2
15.	Anticipatory Reactor Trip - Main Turbine	S	R	M	1
16.	Anticipatory Reactor Trip - Both Main Feedwater Pumps	S	R	M	1

Reactor Containment Vessel Pressure - High

The Reactor Containment Vessel Pressure-High Trip Setpoint less than or equal to 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RCS Pressure - Low trip.

Reactor Coolant Pump Power Monitors

In conjunction with the power/imbalance/flow trips, the Reactor Coolant Pump Power Monitors trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to more than one reactor coolant pump not operating.

A reactor coolant pump is considered to be not operating when the power required by the pump is greater than or equal to 262% (14,400 kw) or is less than or equal to 20.9% (1152 kw) of the operating power (5500 kw). In order to avoid spurious trips during normal operation, the trip setpoints have been selected to maximize the operating band while assuring that a reactor trip will occur upon loss of power to the pump. The 20.9% trip setpoint and response time are based on the maximum time within which an RCPPM-RPS trip must occur to provide DNBR protection for the four pump coastdown. Florida Power has agreed to take credit for the pump overpower trip in order to assure that certain potential faults (such as a seismically induced fault high signal) will not prevent this instrumentation from providing the protective action (i.e., a trip signal). Thus, the maximum setting, approximately 262% (14,400 kw), was selected.

Anticipatory Reactor Trips

The "Main Turbine" and both "Main Feedwater Pump" Anticipatory Reactor Trips are intended to reduce the consequences of undercooling transients that result in a pressure increase in the reactor coolant system. The trips "anticipate" a certain class of pressure increasing transients (i.e., loss of heat sink on the secondary side).

The Main Turbine is considered to be not operating when the turbine control oil pressure monitor indicates less than or equal to 45 psig. A Main Feedwater pump is considered to be not operating when the pump control oil pressure monitor indicates less than or equal to 55 psig.

II. CONTAINMENT ISOLATION VALVES

Proposed Change

Change T.S. 3.6.3.1 to include eight additional liquid sampling (CA) valves and sixteen additional gaseous sampling (WS) valves.

The liquid and gaseous sampling valves were a part of the Post Accident Sampling System and as such are normally locked closed and will not, with one exception, receive an ES actuation signal. The one exception is CAV-431, which is required to close within 60 seconds of actuation.

Reasons for the Proposed Change

This change is being made due to the addition of twenty-four (24) new containment isolation valves accommodating the new Post-Accident Sampling System. In the event of an accident, these systems will provide information about radiological conditions within the Reactor Building.

Safety Analysis

Adding all of the new containment isolation valves to Specification 3.6.3.1 is consistent with past practices. The addition of new containment penetrations should not increase the likelihood of an accident or increase the consequences of an accident. Including these valves in Specification 3.6.3.1 will help to assure that containment integrity and reliability is maintained.

For the new containment isolation valves asterisked (*), ACTION statement b. or c. will place the penetration in its post containment isolation position. The entry into other OPERATIONAL MODES should not be prohibited because the associated penetration will be in the isolation position and would not contribute to an accident, if containment isolation were required. This is consistent with Amendment 63, which granted similar relief for several other valves.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
A.	CONTAINMENT ISOLATION		
1.	BSV-27 check #	closed dur. nor. operation and open dur. RB spray	NA
	BSV-3 #	"	60
	BSV-26 check #	"	NA
	BSV-4 #	"	60
2.	CAV-126 (A)*	iso. CA sys. fr. RC letdn.	60
	CAV-1 (A)*	iso. CA sys. fr. pzr.	60
	CAV-3 (A)*	"	60
	CAV-2 (B)*	iso. CA sys. fr. RB	60
	CAV-4 # (A)*	isolate liquid sampling system	60
	CAV-6 # (B)*	"	60
	CAV-5 # (A)*	"	60
	CAV-7 # (B)*	"	60
	CAV-429 *	iso. CA fr. RC	NA
	CAV-430 *	"	NA
	CAV-433 *	iso. CA fr. RB sump	NA
	CAV-434 *	"	NA
	CAV-431 *	iso. CA fr. RB	60
	CAV-432 *	"	NA
	CAV-435 *	"	NA
	CAV-436 *	"	NA
3.	CFV-20 check	iso. N_2 supply fr. CFT-1A	NA
	CFV-28 (A/B)*	"	60
	CFV-17 check	iso. N_2 supply fr. CFT-1B	NA
	CFV-27 (A/B)*	"	60
	CFV-18 check	iso. MU system fr. CFT-1B	NA
	CFV-26 (A/B)*	"	60
	CFV-19 check	iso. MU system fr. CFT-1A	NA
	CFV-25 (A/B)*	"	60
	CFV-42 (B)*	iso. liquid sampling fr. CF system	60
	CFV-15 (A)*	iso. WD sys. fr. CF tanks	60
	CFV-16 (A)*	"	60
	CFV-29 (B)*	"	60
	CFV-11 (A)*	iso. CF tanks fr. liquid sampling system	60
	CFV-12 (A)*	"	60

TABLE 3.6-1 (continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
9.	MUV-40 (A)*	iso. MU system from RC	60
	MUV-41 (A)*	"	60
	MUV-49 (B)	"	60
	MUV-261	iso. MU system from control bleed-off	60
	MUV-260	"	60
	MUV-259	"	60
	MUV-258	"	60
	MUV-253	"	60
	MUV-163 check #	open during HPI and closed dur. nor. operation	NA
	MUV-25	"	60
	MUV-164 check #	"	NA
	MUV-26 #	"	60
	MUV-160 check #	"	NA
	MUV-23 #		60
	MUV-161 check #	"	NA
	MUV-24 #	"	60
	MUV-27 #	open dur. nor. operation and closed during RB Isolation	60
10.	SWV-39 #	iso. NSCCC from AHF-1C	60
	SWV-45 #	"	60
	SWV-35 #	iso. NSCCC from AHF-1A	60
	SWV-41 #	"	60
	SWV-37 #	iso. NSCCC from AHF-1B	60
	SWV-43 #	"	60
	SWV-48 # *	iso. NSCCC from MUHE-1A & 1B and WDT-5	60
	SWV-47 # *	"	60
	SWV-49 # *	"	60
	SWV-50 # *	"	60
	SWV-80 #	iso. NSCCC from RCP-1A	60
	SWV-84 #	"	60
	SWV-82 #	iso. NSCCC from RCP-1C	60
	SWV-86 #	"	60
	SWV-81 #	iso. NSCCC from RCP-1D	60
	SWV-85 #	"	60
	SWV-79 #	iso. NSCCC from RCP-1B	60
	SWV-83 #	"	60
	SWV-109 #	iso. NSCCC from DRRD-1	60
	SWV-110 #	"	60

TABLE 3.6-1 (continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
11.	WDV-4 (B)	iso. WDT-4 from RB sump	60
	WDV-3 (A)	"	60
	WDV-60 (A)*	iso. WDT-4 from WDT-5	60
	WDV-61 (B)*	"	60
	WDV-94 (A)	iso. WDT-4 from WDP-8	60
	WDV-62 (B)	"	60
	WDV-406 (A)*	iso. waste gas disposal from vents in RC system	60
	WDV-405 (B)*	"	60
12.	WSV-3	iso. containment monitoring system from RB	60
	WSV-4	"	60
	WSV-5	"	60
	WSV-6	"	60
	WSV-26*	iso. gaseous sampling sys. fr. RB	NA
	WSV-27*	"	NA
	WSV-28*	"	NA
	WSV-29*	"	NA
	WSV-30*	"	NA
	WSV-31*	"	NA
	WSV-32*	"	NA
	WSV-33*	"	NA
	WSV-34*	"	NA
	WSV-35*	"	NA
	WSV-38*	"	NA
	WSV-39*	"	NA
	WSV-40*	"	NA
	WSV-41*	"	NA
	WSV-42*	"	NA
	WSV-43*	"	NA

B. CONTAINMENT PURGE AND EXHAUST

1.	AHV-1C (A)##*	iso. pur. sup. system fr. RB	60
	AHV-1D (B)##*	"	60
	AHV-1B (A)##*	iso. pur. exhaust system fr. RB	60
	AHV-1A (B)##*	"	60

C. MANUAL

1.	IAV-28	iso. IA from RB	NA
	IAV-29	"	NA
2.	LRV-50	iso. leak rate test system from RB	NA
	LRV-36	"	NA

III. REACTOR COOLANT HIGH POINT VENTS

Proposed Change

Add Technical Specification 3.4.11 and the Bases for this specification to Appendix A. This change specifies operability and surveillance requirements for the recently installed High Point Vent System.

This system includes three sets of two solenoid controlled valves and one manual block valve each on the pressurizer and each high point of reactor coolant loops A and B. During operation, one solenoid valve will function as a block valve and the other solenoid valve will function as the vent valve. The manually operated block valve is inaccessable during normal operations.

Reasons for the Proposed Change

This specification is being added in response to NUREG-0737, Item II.B.1. This system provides the capability to vent noncondensible gases from the Reactor Coolant System which may inhibit core cooling during natural circulation, following an inadequate core cooling event.

Safety Analysis

Generic Letter 83-37, dated November 1, 1983, established guidelines for developing a specification for the High Point Vent System. The Technical Specification proposed herein does not include operability requirements for a reactor vessel head vent as proposed by the draft. Crystal River Unit 3 has not installed this vent path, which is being addressed separately.

The pressurizer steam space vent valve has been treated differently than the reactor coolant loop vents. This approach was taken due to the similarity between the pressurizer vent and the power operated relief valve (PORV). The PORV is capable of performing the pressurizer vent functions. As stated in Action a., with the alternate vent path available (PORV) operation may continue if the pressurizer vent is maintained closed.

Action Statement a. and b. specify that, when the pressurizer vent and alternate vent or a reactor coolant loop vent is inoperable, the path should be returned to operable status within 30 days or a Special Report must be submitted. This provision will ensure that actions to restore the path will be taken and that the staff will be informed if the path cannot be restored within 30 days. FPC considers a special report submittal to be more appropriate than a plant shutdown, if one pressurizer or loop vent is inoperable.

As recommended by Generic Letter 83-37 and the staff letter dated October 1, 1984, this specification includes three tests to: 1) verify valve position, 2) cycle each vent valve, and 3) perform a flow test.

Additionally, Florida Power Corporation proposes that this specification be exempt from the requirements of Specification 3.0.4. (Technical Specification Change Request No. 120, dated earlier addressed this Proposal.) Specification 3.4.11, as proposed by Generic Letter 83-37, would allow STARTUP and POWER OPERATION to continue provided the vent path is maintained closed and deenergized. HOT STANDBY and HOT SHUTDOWN should also be allowed to continue, as there are no additional safety concerns associated with operation in these modes during vent path inoperability.

As an aside, within the Generic Letter proposed specification, there are at least two action statement requirements that could lead to misunderstandings. The requirement to maintain inoperable vent paths closed and deactivated could prevent restoration to operable status. In most cases, equipment is functionally tested following repair to prove that it is conclusively restored to OPERABLE status. If the equipment is required to be isolated and deactivated, this may not be possible. Thus, one action statement requirement prevents fulfillment of another requirement.

The Generic Letter proposed action a. allowing continued STARTUP and POWER OPERATION could also lead to misunderstandings. Specifically, action a. states actions to be taken to allow continued STARTUP and POWER OPERATION. However, no actions are specified for the other applicable modes.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one reactor coolant system vent path consisting of one vent valve and one block valve capable of being powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Pressurizer Steam Space

b. Reactor Coolant Loop A High Point

c. Reactor Coolant Loop B High Point

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With the pressurizer steam space vent path inoperable, maintain the inoperable vent path closed with power removed from the valve actuator of the vent valve and block valve in the vent path and provided an alternate vent path is available; with no alternate vent path available, restore the pressurizer steam space vent path to OPERABLE status within 30 days, or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days describing the reasons for inoperability and a schedule for corrective action.

b. With one of the two reactor coolant loop vent paths inoperable, maintain the inoperable vent path closed with power removed from the valve actuator of the vent valve and block valve in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days describing the reasons for inoperability and a schedule for corrective action.

c. With two reactor coolant loop vent paths inoperable; maintain the inoperable vent paths closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of two of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.11. Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each vent valve and block valve through at least one complete cycle of full travel from the Control Room.

3. Verifying flow through the Reactor Coolant Vent System vent paths.

BASES

3/4.4.11 Reactor Coolant System Vents

The operability and surveillance requirements for the Reactor Coolant System (RCS) Vents ensure that gases which could inhibit core cooling during natural circulation may be vented from the RCS. This system was installed as a result of NUREG-0737, Item II.B.1.

VIII. REACTOR BUILDING PURGE SUPPLY AND EXHAUST VALVES

Proposed Changes

Change Technical Specification 3.6.3.1 to require the reactor building purge supply and exhaust valves (AHV-1A, 1B, 1C and 1D) be maintained closed during MODES 1, 2, 3, and 4. Also add a surveillance requirement requiring that these valves be verified closed every 31 days and leak tested prior to entry into MODE 4, following use of the purge system.

Delete item 4.a., "Reactor Building Purge Exhaust Duct Isolation on High Radioactivity", from Specification 3.3.2.1.

Reasons for Proposed Change

NUREG-0737, Item II.E.4.2.6 requires that reactor building purge valves that do not satisfy the operability criteria of Branch Technical Position CSB 6-4 must be sealed closed during MODES 1, 2, 3, and 4 and verified closed at least every 31 days. This Technical Specification change ensures that these valves are closed and verified closed as required by the letter from the NRC, dated April 6, 1983, concerning the purge valve isolation dependability.

Because Florida Power is required to maintain the reactor building purge valves closed during MODES 1, 2, 3, and 4, the requirement to maintain purge isolation on high radioactivity operable is not necessary. This function should be deleted from Technical Specification 3.3.2.1.

Safety Analysis

This change will increase plant safety. Closing the reactor building purge valves will improve the containment isolation dependability. This change will assure that the purge valves will be closed during those accidents that require containment isolation. Isolation of purge valves, because they are a direct access from the reactor building to the atmosphere, is necessary for all reactor building accidents to reduce off-site dose consequences.

Florida Power Corporation has not observed purge valve seal degradation occurring while the valves are isolated. A recent leak test after approximately 12 months of operation indicated no seal deterioration. We, therefore, have not proposed a periodic leak test of these valves, as requested in the October 1, 1984 staff letter. Instead we have proposed a requirement to perform a leak test following use of the purge system. Experience has shown that this type of test frequency can increase purge system reliability.

Deletion of the purge isolation function on high radioactivity from Specification 3.3.2.1 will not affect plant safety. As currently written, this specification requires operability during MODES 1, 2, 3, and 4, which is also when the purge valves must be maintained closed. Finally, Specification 3.9.9. requires that the purge system isolation on high radiation be verified prior to and periodically during core alterations. The requirements of Specifications 3.6.3.1 and 3.9.9 are sufficient to assure plant safety is not compromised.

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. REACTOR BLDG. SPRAY					
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	3	2	2	1, 2, 3	12
b. Automatic Actuation Logic	2	1	2	1, 2, 3	10
4. OTHER SAFETY SYSTEMS					
a. Deleted					

TABLE 3.3-3 (Continued)

TABLE NOTATION

*Trip function may be bypassed in this MODE with RCS pressure below 1700 psig. Bypass shall be automatically removed when RCS pressure exceeds 1700 psig.

**Trip function may be bypassed in this MODE with RCS pressure below 900 psig. Bypass shall be automatically removed when RCS pressure exceeds 900 psig.

***Trip function may be bypassed in this MODE with steam generator pressure below 725 psig. Bypass shall be automatically removed when steam generator pressure exceeds 765 psig.

#The provisions of Specification 3.0.4 are not applicable.

##Trip function may be bypassed in this MODE prior to stopping the operating main feedwater pump. Bypass shall be manually removed after starting the first main feedwater pump.

ACTION STATEMENTS

ACTION 9 - With the number of OPERABLE Channels one less than the Total Number of Channels operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 10 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.2.1.1.

ACTION 12 - With the number of OPERABLE Channels one less than the Total Number of Channels operation may proceed provided the inoperable channel is placed in the bypassed condition and the minimum channels OPERABLE required is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for Surveillance testing per Specification 4.3.2.1.

ACTION 13 - With the number of OPERABLE Channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

TABLE 3.3-4 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. REACTOR BLDG. SPRAY		
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	$\leq$ 30 psig See 1.a.2, 3, 4	$\leq$ 30 psig See 1.a.2, 3, 4
b. Automatic Actuation Logic	Not Applicable	Not Applicable
4. OTHER SAFETY SYSTEMS		
a. Deleted		
b. Steam Line Rupture Matrix		
1. Low SG Pressure	$\geq$ 600 psig	$\geq$ 600 psig
2. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Emergency Feedwater		
1. Main Feedwater Pump Turbines A and B Control Oil Low	$\geq$ 55 psig	$\geq$ 55 psig
2. OTSG A and B Level Low-Low	$\geq$ 18 inches	$\geq$ 18 inches

TABLE 3.3-5 (Cont'd)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS*
7. Deleted	
8. Main Feedwater Pump Turbines A and B Control Oil Low	
a. Emergency Feedwater Actuation	Not Applicable
9. OTSG A and B Level Low-Low	
a. Emergency Feedwater Actuation	Not Applicable

*Diesel Generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

TABLE 4.3-2 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. REACTOR BLDG. SPRAY				
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	S	R	M(4)	1, 2, 3
b. Automatic Actuation Logic	N/A	N/A	M(1)(3)(5)	1, 2, 3
4. OTHER SAFETY SYSTEMS				
a. Deleted				
b. Steam Line Rupture Matrix				
1. Low SG Pressure	N/A	R	N/A	1, 2, 3
2. Automatic Actuation Logic	N/A	N/A	M (3)	1, 2, 3
c. Emergency Feedwater				
1. Main Feedwater Pump Turbines A and B Control Oil Low	S	R	N/A	1, 2, 3
2. OTSG-A and B Level Low-Low	S	R	N/A	1, 2, 3, 4

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE, during shutdown, at least once per 18 months by:

a. Verifying that on a containment isolation test signal, each automatic isolation valve actuates to its isolation position. The provisions of Specification 4.0.4 are not applicable.

b. Verifying that on a containment radiation-high test signal, each purge and exhaust automatic valve actuates to its isolation position.

4.6.3.1.3 The containment purge supply and exhaust isolation valves shall be determined closed at least once every 31 days when in MODES 1, 2, 3 and 4 with the breakers locked out and the operating air isolated.

4.6.3.1.4 The containment purge supply and exhaust isolation valve seals shall be demonstrated OPERABLE by performance of a leak test prior to entering MODE 4, following use of the purge system.

TABLE 3.6-1 (continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
9.	MUV-40 (A)*	iso. MU system from RC	60
	MUV-41 (A)*	"	60
	MUV-49 (B)	"	60
	MUV-261	iso. MU system from control bleed-off	60
	MUV-260	"	60
	MUV-259	"	60
	MUV-258	"	60
	MUV-253	"	60
	MUV-163 check #	open during HPI and closed dur. nor. operation	NA
	MUV-25	"	60
	MUV-164 check #	"	NA
	MUV-26 #	"	60
	MUV-160 check #	"	NA
	MUV-23 #	"	60
	MUV-161 check #	"	NA
	MUV-24 #	"	60
	MUV-27 #	open dur. nor. operation and closed during RB Isolation	60
10.	SWV-39 #	iso. NSCCC from AHF-1C	60
	SWV-45 #	"	60
	SWV-35 #	iso. NSCCC from AHF-1A	60
	SWV-41 #	"	60
	SWV-37 #	iso. NSCCC from AHF-1B	60
	SWV-43 #	"	60
	SWV-48 # *	iso. NSCCC from MUHE-1A & 1B and WDT-5	60
	SWV-47 # *	"	60
	SWV-49 # *	"	60
	SWV-50 # *	"	60
	SWV-80 #	iso. NSCCC from RCP-1A	60
	SWV-84 #	"	60
	SWV-82 #	iso. NSCCC from RCP-1C	60
	SWV-86 #	"	60
	SWV-81 #	iso. NSCCC from RCP-1D	60
	SWV-85 #	"	60
	SWV-79 #	iso. NSCCC from RCP-1B	60
	SWV-83 #	"	60
	SWV-109 #	iso. NSCCC from DRRD-1	60
	SWV-110 #	"	60

TABLE 3.6-1 (continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
11.	WDV-4 (B)	iso. WDT-4 from RB sump	60
	WDV-3 (A)	"	60
	WDV-60 (A)*	iso. WDT-4 from WDT-5	60
	WDV-61 (B)*	"	60
	WDV-94 (A)	iso. WDT-4 from WDP-8	60
	WDV-62 (B)	"	60
	WDV-406 (A)*	iso. waste gas disposal from vents in RC system	60
	WDV-405 (B)*	"	60
12.	WSV-3	iso. containment monitoring system from RB	60
	WSV-4	"	60
	WSV-5	"	60
	WSV-6	"	60

B. CONTAINMENT PURGE AND EXHAUST

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
1.	AHV-1C ##*	iso. pur. sup. system fr. RB	60
	AHV-1D ##*	"	60
	AHV-1B ##*	iso. pur. exhaust system fr. RB	60
	AHV-1A ##*	"	60

C. MANUAL

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
1.	IAV-28	iso. IA from RB	NA
	IAV-29	"	NA
2.	LRV-50	iso. leak rate test system from RB	NA
	LRV-36	"	NA

TABLE 3.6-1 (continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME
Blind Flange 348	iso. fuel transfer tube from Transfer Canal	NA
Blind Flange 436		NA
Equipment Hatch	iso. RB	NA
Personnel Hatch		MA

\# Not subject to Type C Leakage Test

\## The containment purge supply and exhaust valves must be closed during MODES 1, 2, 3 and 4.

* The provisions of Specification 3.0.4 are not applicable.

(A) Isolates on Diverse Isolation Actuation Signal A

(B) Isolates on Diverse Isolation Actuation Signal B

(A/B) Isolates on Diverse Isolation Actuation Signal A or B

SPECIAL REPORTS (Continued)

p. Inoperable Reactor Coolant Vent Paths, Specification 3.4.11

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility operation covering time intervals at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source and fission detector leak tests and results.

i. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Facility Operating License:

a. Records and drawing changes reflecting facility design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of facility radiation and contamination surveys.

d. Records of radiation exposure for all individuals entering radiation control areas.