3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

APPLICABILITY: MODES 1, 2*, 3, 4 and 5

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of 11,600 ppm boric acid solution or its equivalent, until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.0% delta k/k:

a. Within one hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable control rod(s).

b. When in MODES 1 or 2#, at least once per 12 hours, by verifying that regulating rod groups withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2## within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading by consideration of the factors of e. below, with the regulating rod groups at the maximum insertion limit of Specification 3.1.3.6.

#With K_{eff} greater than or equal to 1.0.

##With K_{eff} less than or equal to 1.0

*See Special Test Exception 3.10.4.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3, 4 or 5 at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration.
2. Control rod position,
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm 1\%$ delta k/k at least once per 31 Effective Full Power Days. (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.1.e above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the concentrated boric acid storage system via a boric acid pump and makeup or decay heat removal (DHR) pump to the Reactor Coolant System, and

b. A flow path from the borated water storage tank via makeup or DHR pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the flow path from the concentrated boric acid storage system inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% delta k/k at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the flow path from the borated water storage tank inoperable, restore the flow path to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the pipe temperature of the heat traced portion of the flow path from the concentrated boric acid storage system is greater than 105°F.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

REACTIVITY CONTROL SYSTEMS

MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4.1 At least two makeup pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With only one makeup pump OPERABLE, restore at least two makeup pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1.0% delta k/k at 200°F within the next 6 hours; restore at least two makeup pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.

MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4.2 At least one makeup pump shall be OPERABLE.

APPLICABILITY: MODES 4*

ACTION:

With no makeup pump OPERABLE, restore at least one makeup pump to OPERABLE status within one hour or be borated to a SHUTDOWN MARGIN equivalent to 1.0% delta k/k at 200°F and be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*With RCS pressure greater than or equal to 150 psig.

BORIC ACID PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.7 At least one boric acid pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

With no boric acid pump OPERABLE, restore at least one boric acid pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% delta k/k at 200°F within the next 6 hours; restore at least one boric acid pump to OPERABLE status with the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.7 No additional Surveillance requirements other than those required by Specification 4.0.5.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.9 Each of the following borated water sources shall be OPERABLE

a. A concentrated boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 6,730 gallons,

2. Between 11,600 and 14,000 ppm of boron, and

3. A minimum solution temperature of 105°F.

b. The borated water storage tank (BWST) with:

1. A minimum contained borated water volume of 415,200 and 449,000 gallons,

2. Between 2,270 and 2,450 ppm of boron, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With the concentrated boric acid storage system inoperable, restore the storage system to OPERABLE within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% delta k/k at 200°F within the next 6 hours; restore the concentrated boric acid storage system to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 30 hours.

b. With the borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.9 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source.

2. Verifying the contained borated water volume of each water source, and

3. Verifying the concentrated boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the BWST temperature when outside air temperature is less than 40°F.

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. During Modes 1 and 2 the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration and RCS T_{avg}. The most restrictive condition for Modes 1, 2, and 3 occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident a minimum SHUTDOWN MARGIN of 0.60% delta k/k is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

3/4.1.1.2 BORON DILUTION

A minimum flow rate of at last 2700 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual through the Reactor Coolant System in the core during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2700 GPM will circulate an equivalent Reactor Coolant System volume of 12,000 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration reduction will be within the capability for operator recognition and control

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

III. HEATUP AND COOLDOWN CURVES AND SURVEILLANCE CAPSULE SCHEDULE REVISION

Proposed Change

Replace 3.4.9.1.b of the Limiting Conditions of Operation with:

b. For the temperature ranges specified below, the cooldown rates should be as specified (in any one hour period):

i.	$T > 270^{\circ}F$	$100^{\circ}F/Hr$,
ii.	$270^{\circ}F > T > 170^{\circ}F$	$50^{\circ}F/Hr$,
iii.	$270^{\circ}F > T > 70^{\circ}F$	$10^{\circ}F/Hr$,

and

Replace Table 3.4-4 and Figures 3.4-1, 3.4-2, and 3.4-3 with the appropriate revised Table and Figures.

Reasons for the Proposed Change

The proposed changes are submitted as a result of the irradiated reactor vessel surveillance as required by Technical Specification 4.4.9.1.2.

The ability of the reactor pressure vessel to resist fracture is the main factor in ensuring the safety of the primary system. In reactor pressure vessel steels, the most serious mechanical property change that could result in failure is the increase in transition temperature from ductile to brittle fracture (RT_{NDT}) accompanied by a reduction in the upper shelf impact toughness. The proposed changes in the cooldown limit and the pressure temperature curves take into account the effects of irradiation on the RT_{NDT} and other effects on reactor vessel materials as determined by BAW-1679, Rev. 1. BAW-1679, Rev. 1 is an analysis by Babcock and Wilcox of the irradiated speciman removed from Crystal River Unit 3 during Refuel I.

The implementation of low leakage fuel cycles at most B&W-designed plants, including Crystal River Unit 3, caused a corresponding decrease in the estimated end-of-life (EOL) fluences for affected reactor vessels. If capsule withdrawal continues according to the current schedule, a major portion of the capsules currently undergoing irradiation will accumulate fluences well in excess of estimated vessel EOL. Thus, the data obtained from evaluation of these capsules will be excessively conservative for evaluation of corresponding vessel integrity. Table 1 for Crystal River Unit 3 illustrates the impact of this fuel cycle design change. The revised schedule will assure acquisition of representative capsule data which can be related to the irradiated condition of the vessel.

Safety Analysis

The new pressure temperature limits of the reactor coolant pressure boundary were established in accordance with the requirements of 10 CFR 50, Appendix G. The methods and criteria employed that establish operating pressure and temperature limits are described in topical report BAW-10046. The revised pressure and temperature limits correspond to the design safety analysis for avoiding brittle fracture. Thus, this Technical Specification change will not degrade plant safety.

The withdrawal schedule as specified in ASTM Specification E-185-79, referenced in 10CFR50, Appendix H, requires that capsules be withdrawn at designated intervals such that the capsule fluence will correspond to specified conditions of the reactor vessel with respect to irradiation damage. Although the B&W Integrated Reactor Vessel Materials Surveillance Program was designed prior to E-185-79, regulations require that the program be maintained, to the extent practical, with the updated requirements. Therefore, to insure that capsules are withdrawn in a manner consistant with the required compatibility of capsule fluence and reactor vessel fluence, it is necessary to revise the current withdrawal sequence.

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-3, and 3.4-4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any one hour period,

b. For the temperature ranges specified below, the cooldown rates should be as specified (in any one hour period):

i.	$T > 270^{\circ}F$	$100^{\circ}F/Hr$,
ii.	$270^{\circ}F > T > 170^{\circ}F$	$50^{\circ}F/Hr$,
iii.	$170^{\circ}F > T > 70^{\circ}F$	$10^{\circ}F/Hr$,

and

c. A maximum temperature change of less than or equal to 5°F in any one hour period during hydrostatic testing operations above system design pressure.

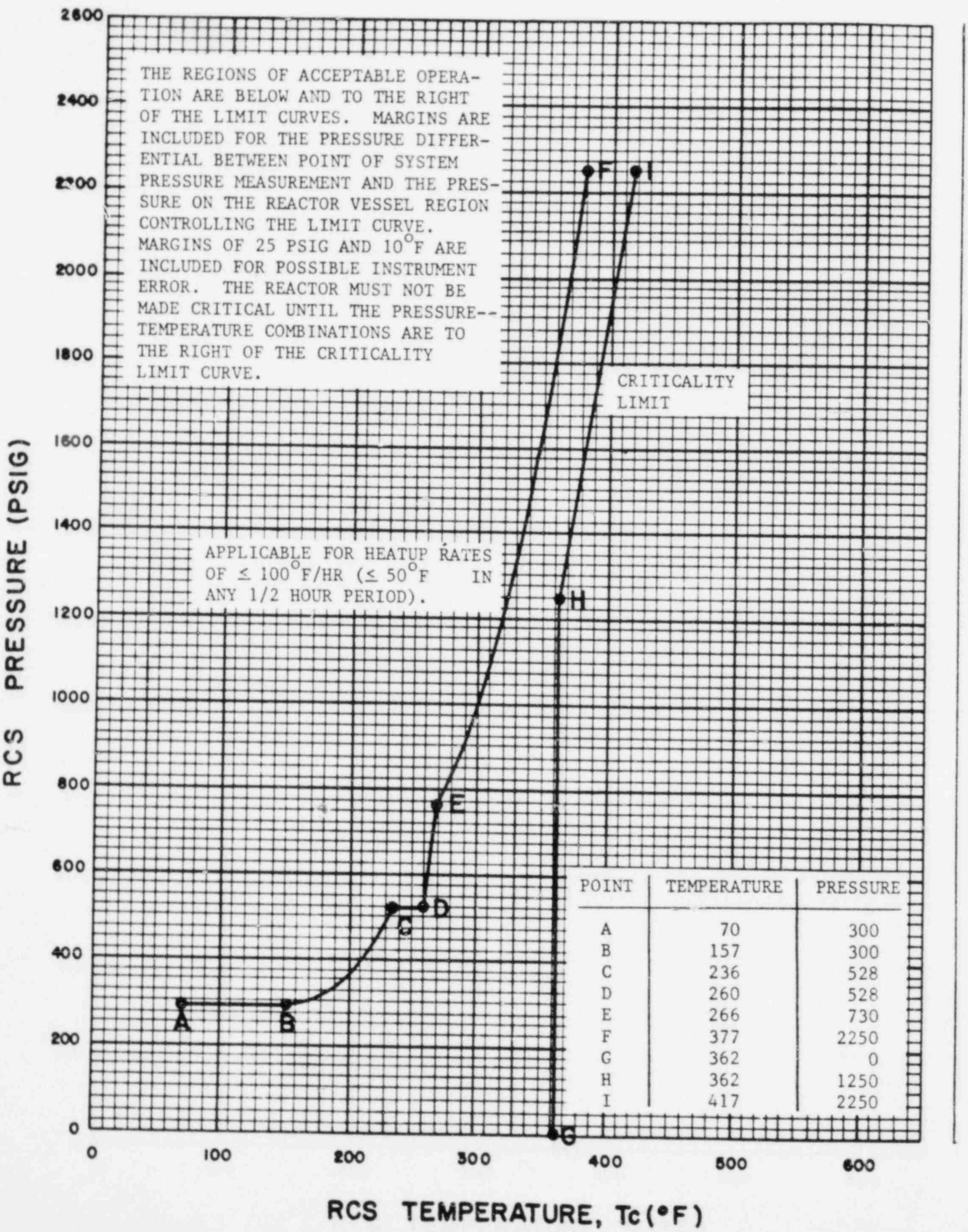
APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce RCS T avg and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

FIGURE 3.4-2

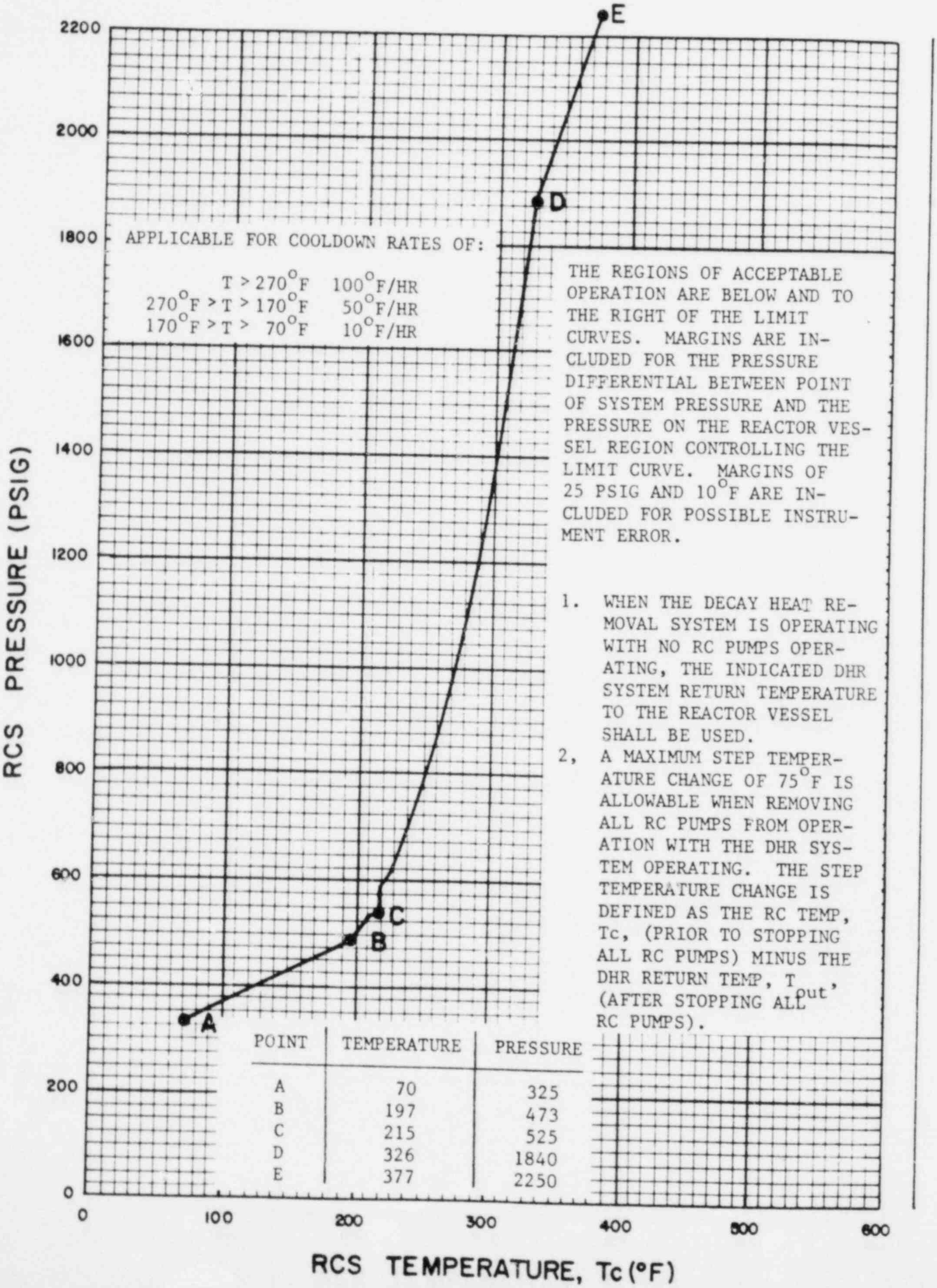
REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR HEATUP FOR FIRST 8 EFPY



POINT	TEMPERATURE	PRESSURE
A	70	300
B	157	300
C	236	528
D	260	528
E	266	730
F	377	2250
G	362	0
H	362	1250
I	417	2250

FIGURE 3.4-3

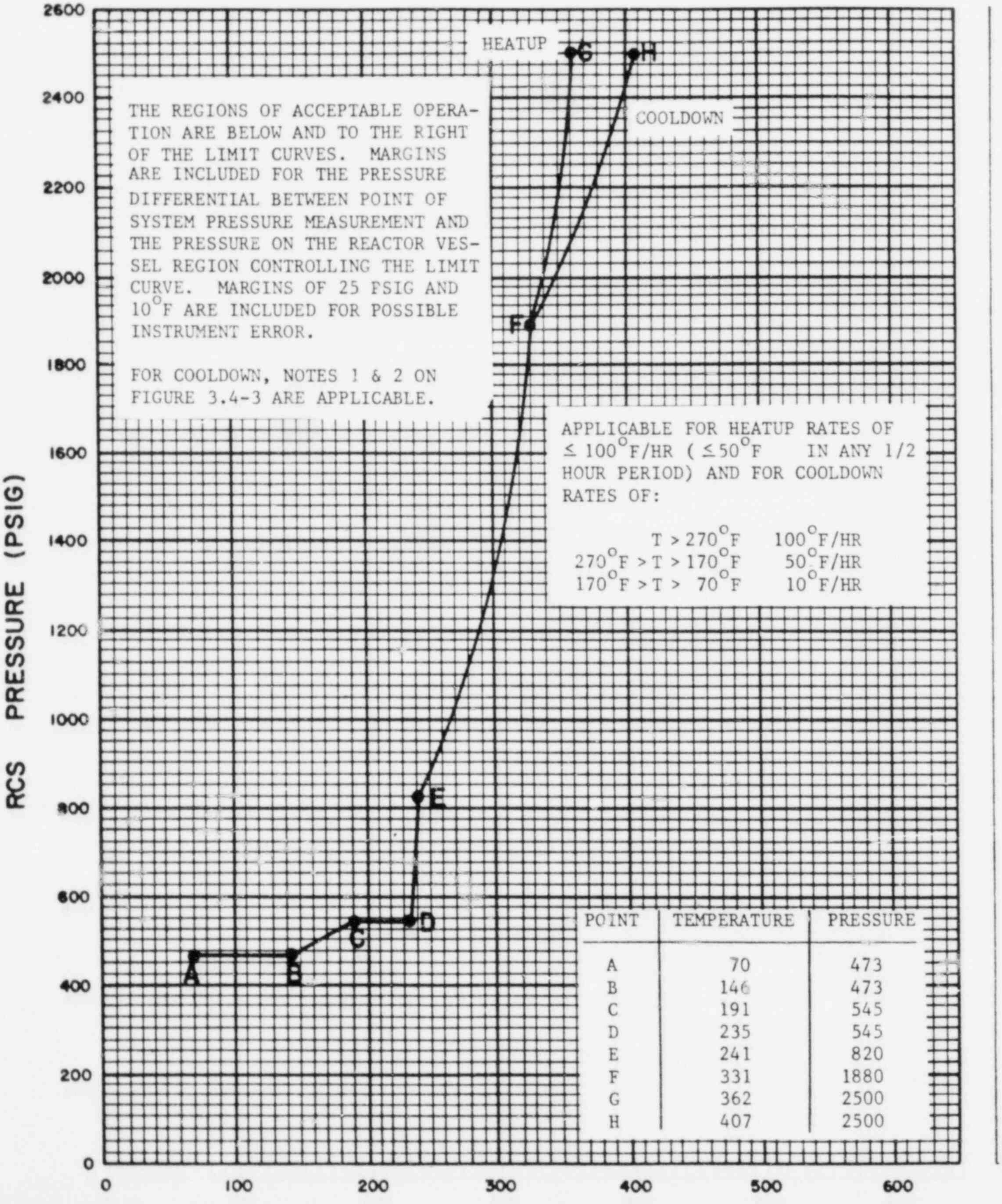
REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS FOR COOLDOWN FIRST 8 EFPY



POINT	TEMPERATURE	PRESSURE
A	70	325
B	197	473
C	215	525
D	326	1840
E	377	2250

FIGURE 3.4-4

REACTOR COOLANT SYSTEM PRESSURE - TEMPERATURE LIMITS FOR HEATUP & COOLDOWN LIMITS FOR INSERVICE LEAK AND HYDROSTATIC TESTS FOR FIRST 8 EFPY



POINT	TEMPERATURE	PRESSURE
A	70	473
B	146	473
C	191	545
D	235	545
E	241	820
F	331	1880
G	362	2500
H	407	2500

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

CAPSULE	INSTALLATION	REMOVAL
CR3-A	At 269 EFPD of First Cycle	End of Ninth Cycle
CR3-B	Initial Fuel Load	At 269 EFPD of First Cycle
CR3-C	At 269 EFPD of First Cycle	End of Sixth Cycle
CR3-D	Initial Fuel Load	End of Fourth Cycle
CR3-E	At 269 EFPD of First Cycle	End of Ninth Cycle
CR3-F	At 269 EFPD of First Cycle	End of Seventh Cycle

recalculated when the RT_{NDT} determined from the surveillance capsule is different from the calculated RT_{NDT} for the equivalent capsule radiation exposure.

The closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt pre-load). This region largely controls the pressure-temperature limitations of the first several service periods. The outlet nozzles of the reactor vessel also affect the pressure-temperature limit curves of the first several service periods. This is due to the high local stresses at the inside corner of the nozzle which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the RT_{NDT} temperature of the beltline region materials is high enough so that the beltline region of the reactor vessel will start to control the pressure-temperature limitations of the reactor coolant pressure boundary. For the service period for which the limit curves are established, the maximum allowable pressure as a function of fluid temperature is obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and beltline region. The maximum allowable pressure is taken to be the lower pressure of the three calculated pressures. The calculated pressure temperature limit curves are then adjusted by 25 psi and 10^{o}F for possible errors in the pressure measurement and the limiting component for all operating reactor coolant pump combinations. The limit curves were prepared based upon the most limiting adjusted reference temperature of all the beltline region materials at the end of the eighth effective full power year.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside the radius are essentially identical, the measured transistion shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The limit curves must be recalculated when the RT_{NDT} determined from the surveillance capsule is different from the calculated RT_{NDT} for the equivalent capsule radiation exposure. The first surveillance program capsule, Capsule B, was withdrawn at 269 EFPD and new limit curves were prepared based upon the analysis of Capsule B. The limit curves are applicable until the end of eight EFPY.

The unirradiated transverse impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determined for those materials for which sufficient amounts of material were available. The unirradiated impact properties and residual elements of the beltline region materials are listed in Bases Table 4-1. The adjusted reference temperature is calculated by adding the predicted radiation-induced delta RT_{NDT} and the unirradiated. The predicted delta RT_{NDT} is calculated using the respective neutron fluence and copper and phosphorus contents. Bases Figure 4-1 illustrates the calculated peak neutron fluence, at several locations through the reactor vessel beltline region wall and at the center of the surveillance capsules as a function of exposure time.

Bases Figure 4-2 illustrates the design curves for predicting the radiation induced delta RT_{NDT} as a function of the material's copper and phosphorus content and neutron fluence. The adjusted RT_{NDT}'s of the beltline region materials at the end of the fifth full power year are listed in Bases Table 4-1. The adjusted RT_{NDT}'s are given for the 1/4T and 3/4T (T is wall thickness) vessel wall locations. The assumed RT_{NDT} of the closure head region and of the outlet nozzle steel forgings is $60^{\circ}F$.

During cooldown at the higher temperatures, the limits are imposed by thermal and loading cycles on the steam generator tubes. These limits are the vertical segments of the limit lines on Figures 3.4-3 and 3.4-4, respectively. These limits will not require adjustments due to the neutron fluences.

Figure 3.4-2 presents the pressure-temperature limit curve for normal heatup. This figure also presents the core criticality limits as required by Appendix G to 10 CFR50. Figure 3.4-3 presents the pressure temperature limit curve for normal cooldown. Figure 3.4-4 presents the pressure-temperature limit curves for heatup and cooldown for inservice leak and hydrostatic testing. These figures are discussed in greater detail in BAW-1679, Rev. 1, June 1982, Analysis of Capsule CR3-B from Florida Power Corporation, Crystal River Unit 3 Reactor Vessel Materials, Surveillance Program.

BASES

All pressure-temperature limit curves are applicable up to the eighth effective full power year. The protection against non-ductile failure is assured by maintaining the coolant pressure below the upper limits of Figures 3.4-2, and 3.4-3, and 3.4-4.

The pressure and temperature limits shown on Figures 3.4-2 and 3.4-4 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFT Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTUAL INTEGRITY

The inspection programs for ASME Code Class 1, 2, and 3 components, except steam generator tubes, ensure that the structual integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves 1) ensure OPERABILITY, 2) ensure that the valves are not stuck open during normal operation, and 3) demonstrate that the valves are fully open at the forces assumed in the safety analysis.

IV. PORV EMERGENCY POWER SURVEILLANCE

Proposed Change

Delete surveillance requirement 4.4.3.2.3 from the page 3/4.4-4a and change "18 months" to "refueling cycle".

Reasons for the Proposed Change

Surveillance requirement 4.4.3.2.3 was inadvertantly included in the Crystal River Unit 3 Technical Specification on July 15, 1982, in Amendment 55. The PORV and block valves do not have specific emergency power supplies. On a Loss of Offsite Power Event, the Emergency Diesel Generators automatically supply power to the safety-related buses powering the PORV and block valves. The NRC was informed of this capability in a letter dated November 17, 1979, and concurred with this system in a letter to Florida Power dated May 5, 1980.

Safety Analysis

This change is only editorial. Florida Power Corporation is deleting this surveillance requirement because the equipment required to be checked never existed. This change does not affect plant safety.

REACTOR COOLANT SYSTEM

POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.2 The power operated relief valve (PORV) and its associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the PORV inoperable, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the block valve inoperable, within 1 hour either restore the block valve to OPERABLE status or close the block valve and remove power from the block valve or close the PORV and remove power from the associated solenoid valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 In addition to the requirements of Specifications 4.0.5, the PORV shall be demonstrated OPERABLE at least once per Refueling Cycle by performance of a CHANNEL CALIBRATION.

4.4.3.2.2 The block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

capability should the PORV become inoperable.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief.

The low level limit is based on providing enough water volume to prevent a pressurizer low level or a reactor coolant system low pressure condition that would actuate the Reactor Protection System or the Engineered Safety Feature Actuation System as a result of a reactor trip. The high level limit is based on maximum reactor coolant inventory assumed in the safety analysis.

The power operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valve minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensures that the structual integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant

V. MOTOR OPERATED EMERGENCY FEEDWATER PUMP INCLUSION AND FLOW PATH VERIFICATION

Proposed Change

Add Surveillance Requirements for the motor driven Emergency Feedwater pump to Technical Specification 3.7.1.2

Change the emergency feedwater system flow path verification to also include verifying the position of locked valves, also.

Change "18 months" to "refueling cycle".

Reasons for Proposed Changes

The surveillance requirements for the motor driven Emergency Feedwater pump are being added to the Technical Specifications at the request of the NRC staff (by letter to FPC, dated February 10, 1983).

The additional requirements of flow path verification are also made at the request of the Nuclear Regulatory Commission. These surveillance requirements were requested in a letter to Florida Power dated August 19, 1982.

Safety Analysis

These changes will not degrade plant safety. The additional surveillance requirements are already being performed. This change is merely administrative in nature.

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent steam generator emergency feedwater pumps and associated flow paths shall be OPERABLE with:

a. One emergency feedwater pump capable of being powered from an OPERABLE emergency bus, and

b. One emergency feedwater pump capable of being powered from an OPERABLE steam supply system.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one emergency feedwater pump and/or associated flow path inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that the steam turbine driven pump develops a discharge pressured greater than or equal to 1100 psig on recirculation flow when the secondary steam supply pressure is greater than 200 psig.*

2. Verifying that the motor driven pump develops a discharge pressure of greater than or equal to 1100 psig on recirculation flow.

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* When not in MODES 1, 2, or 3, surveillance shall be performed within 24 hours after entering MODE 3 and prior to entering MODE 2.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each valve in the flow path is in its correct position.

4. Verifying that the emergency feedwater ultrasonic flow rate detector is zero-checked.

b. At least once per Refueling Cycle, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on an emergency feedwater actuation test signal.

2. Verifying that the steam turbine driven pump and the motor driven pump start automatically:

a. Upon receipt of an emergency feedwater actuation OTSG A and B level low-low test signal, and

b. Upon receipt of an emergency feedwater actuation main feedwater pump turbines A and B control oil low test signal.

3. Verifying that the operating air accumulators for FWV-39 and FWV-40 maintain greater than or equal to 27 psig for at least one hour when isolated from their air supply.

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEMS

The OPERABILITY of the emergency feedwater systems ensures that the Reactor Coolant system can be cooled down to less than 280°F from normal operating conditions in the event of a total loss of offsite power.

Each emergency feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1144 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 280°F where the Decay Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 280°F in the event of a total loss of offsite power or of the main feedwater system. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 24 hours with steam discharge to atmosphere concurrent with loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1)minimize the

VI. HYDRAULIC SNUBBER INSPECTION SCHEDULE

Replace page 3/4 7-35 of Appendix A with proposed page 3/4 7-35 enclosed.

Proposed Change

Change the second footnote of Table 4.7-4 on page 3/4 7-35 to read as follows:

** The required inspection interval shall not be lengthened more than one step at a time. Following the 1983 refueling outage, the first inservice visual inspection of snubbers shall be performed after 4 months but within 10 months of commencing POWER OPERATION. Subsequent intervals shall be determined by the above Table.

Reason for Proposed Change

The failure rate associated with snubber inspection and testing over the last operating cycle was high enough to require a 124 day visual inspection interval following the 1983 refueling outage. However, snubber design modifications and maintenance changes were implemented during the 1983 refueling outage to eliminate the causes of the failures encountered during the previous operating cycle.

The proposed change will extend the required inspection interval from 124 days to between 4 and 10 months. This extension is justified by the design modifications and maintenance changes implemented during the 1983 refueling outage.

Safety Analysis

Plant safety will not be compromised by the proposed Technical Specification change. The maintenance performed and the design changes implemented on the Crystal River Unit 3 hydraulic snubbers over the last two refueling outages provide a high assurance of snubber operability. The maintenance and design changes have also provided essentially "new" snubbers. Florida Power Corporation is confident that the past problems encountered with the operability of these snubbers have been eliminated.

TABLE 4.7-4

HYDRAULIC SNUBBER INSPECTION SCHEDULE

NUMBER OF SNUBBERS FOUND INOPERABLE DURING INSPECTION OR DURING INSPECTION INTERVAL*	NEXT REQUIRED INSPECTION INTERVAL**
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3 or 4	124 days ± 25%
5, 6, or 7	62 days ± 25%
Greater than or equal to 8	31 days ± 25%

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* Snubbers may be categorized into two groups, "accessible" and "inaccessible". This categorization shall be based upon the snubber's accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule

** The required inspection interval shall not be lengthened more than one step at a time. Following the 1983 refueling outage, the first inservice visual inspection of snubbers shall be performed after 4 months but within 10 months of commencing POWER OPERATION. Subsequent intervals shall be determined by the above Table.

ATTACHMENT C: NUREG 0737 REQUIRED TECHNICAL SPECIFICATIONS

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 of a 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% FULL 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 was omitted from this submittal. Additionally, rather than state the nominal trip setpoints for these new reactor trips, the generic trip setpoint was specified. The nominal trip setpoints are stated in the Bases.

To remain consistant with other similar trips, the MODES in which surveillance required includes Modes 1 and 2 rather than just Mode 1 as was suggested by Generic Letter 82-16.

Because the 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) (2)	More than one pump drawing $\leq$3900 kw or $\geq$9000 kw	More than one pump drawing $\leq$3900 kw or $\geq$9000 kw
9.	Reactor Containment Vessel Pressure High	$\leq$ 4 psig	$\leq$ 4 psig
10	Anticipatory Reactor Trip - Main turbine	> 20% of RATED THERMAL POWER with a Main Turbine Trip	> 20% of RATED THERMAL POWER with a Main Turbine Trip
11	Anticipatory Reactor Trip - Both Main Feedwater Pumps.	> 20% of RATED THERMAL POWER with Both Main Feedwater Pumps Tripped	> 20% of RATED THERMAL POWER with Both Main Feedwater Pumps Tripped

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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, and
c. The Shutdown Bypass is removed when RCS Pressure > 1800 psig.

(2) Trip may be manually bypassed when reactor power is less than or equal to 2300 MWt.

(3) Operation with three reactor coolant pumps with RCPPM's bypassed is not permitted.

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, b)	2 per pump	1, 2	25
15.	Anticipatory Reactor Trip - Main Turbine	4	2(c)	3	1, 2	3#
16.	Anticipatory Reactor Trip - Both Main Feedwater Pumps	4 per pump	2 per pump(c)	3 per pump	1, 2	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.

(b) Trip may be manually bypassed when reactor power is less than or equal to 2475 MWT and 4 reactor coolant pumps are operating.

(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 *	$\leq$ 0.266 seconds
3.	RCS Outlet Temperature - High	Not Applicable
4.	Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE *	$\leq$ 1.79 seconds
5.	RCS Pressure - Low	$\leq$ 0.44 seconds
6.	RCS Pressure - High	$\leq$ 0.44 seconds
7.	Variable Low RCS Pressure	Not Applicable
8.	Pump Status Based on RCPPMs*	$\leq$ 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, 2
16.	Anticipatory Reactor Trip - Both Main Feedwater Pumps	S	R	M	1, 2

Reactor Containment Pressure - High

The Reactor Containment Pressure-High Trip Setpoint of 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 120% or is less than or equal to 70% of the nominal operating power. The nominal operating power decreases from when a pump is first started during heatup and is pumping dense fluid (typically 7500KW) to when a pump is operating at full reactor power and is pumping less dense fluid (typically 5500KW). In order to avoid spurious trips during normal operation, the 120% trip setpoint (9000KW) is based on the nominal operating power for a pump during heatup and the 70% trip setpoint (3900KW) is based on the nominal operating power for a pump operating at full reactor power. 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 signal will prevent this instrumentation from providing the protective action (i.e., a trip signal).

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). Thus a reactor trip (above 20% RATED THERMAL POWER) occurs if the main turbine is not operating or if both main feedwater pumps are not operating before undercooling begins.

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, sixteen additional gaseous sampling (WS) valves, and four additional hydrogen purge (LR) valves.

The liquid and gaseous sampling valves will be a part of the Post Accident Sampling System and as such will be 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-eight (28) new containment isolation valves accommodating the new Post-Accident Sampling System and Reactor Building Hydrogen Purge System. In the event of an accident, these systems will provide information about radiological conditions within the Reactor Building or allow purging of hydrogen from 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 positions. Therefore, the entry into other OPERATIONAL MODES should not be prohibited because the valves asterisked will be in their isolation positions and would not contribute to an accident if containment isolation was 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 *	is. 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)
4.	CIV-41*	iso. CI sys. fr. RB	60
	CIV-40*	"	60
	CIV-34*	"	60
	CIV-35*	"	60
5.	DHV-93 check	iso. DH system fr. pzr.	NA
	DHV-91*	"	60
	DHV-43 #	iso. DH sys. fr. RB sump	120
	DHV-42 #	"	120
	DHV-4# & 41#	iso. DH sys. fr. RC	120
	DHV-6 #	iso. DH system from Reactor Vessel	60
	DHV-5 #	"	60
6.	DWV-162 check	iso. DW system fr. RB	NA
	DWV-160 (A/B)*	"	60
7.	FWV-44 check #	iso. feedwater from RCSG-1A	NA
	FWV-45 check #	"	NA
	FWV-43 check #	iso. feedwater from RCSG-1B	NA
	FWV-46 check #	"	NA
8.	LRV-70 *	iso. H_2 purge sys. from RB	NA
	LRV-71 *	"	NA
	LRV-72 *	"	NA
	LRV-73 *	"	NA
9.	MSV-130 #(A/B)*	iso. MDT-1 from RCSG-1A	60
	MSV-148 #(A/B)*	iso. MDT-1 from RCSG-1B	60
	MSV-411 # *	iso. main steam lines from RCSG-1A	60
	MSV-412 # *	iso. main steam lines from RCSG-1A	60
	MSV-413 # *	iso. main steam lines from RCSG-1B	60
	MSV-414 # *	iso. main steam lines from RCSG-1B	60

TABLE 3.6-1 (continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
10.	MUV-40 (A)*	iso. MU system from RC	60
	MUV-41 (A)*	"	60
	MUV-49 (B)	"	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
11.	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-109 #	iso. NSCCC from DRRD-1	60
	SWV-110 #	"	60
12.	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

TABLE 3.6-1 (continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
	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
13.	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-38*	"	NA
	WSV-39*	"	NA
	WSV-34*	"	NA
	WSV-35*	"	NA
	WSV-32*	"	NA
	WSV-33*	"	NA
	WSV-41*	"	NA
	WSV-40*	"	NA
	WSV-42*	"	NA
	WSV-43*	"	NA

B. CONTAINMENT PURGE AND EXHAUST

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

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	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 two solenoid controlled valves and one manual block valve 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 is expected to be operable prior to restart for Cycle V. This system will provide the capability to vent noncondensible gases from the Reactor Coolant System which may inhibit core cooling during natural circulation.

Safety Analysis

An NRC Draft Generic Letter, dated January 28, 1983, established preliminary 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 will be addressed separately at a later date. To avoid putting Reactor Coolant System water into the containment during operation or shutdown, two of the proposed surveillance requirements, 4.4.11.1 and 4.4.11.2.3, have been deleted.

Performance of surveillance requirement 4.4.11.1 could lead to a small break loss-of-coolant-accident if one of the vent valves failed open during testing. With one vent valve even partially open, the other vent valve would vent the RCS upon cycling. This same situation arises during the flow test required by 4.4.11.2.3. This flow test would also require that the RCS be pressurized to verify correct flow during operation.

The requirement to shutdown if one vent path is inoperable for more than 30 days has also been deleted. Deletion of this requirement will not increase the probability or consequences of an accident. In the event of an accident requiring an inoperable vent to release noncondensible gases, a safe plant cooldown is possible using the other ventable coolant loop. If two of the vents are inoperable, the plant is required to be in hot standby within 78 hours.

The addition of this new vent system can help to reduce the probability and effects of an accident. The operability and surveillance requirements for the Reactor Coolant System Vents assure that gases which may inhibit core cooling during natural circulation may be vented from of the Reactor Coolant System. This new Technical Specification will not decrease plant safety.

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 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 one of the above reactor coolant system vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of the vent valve and block valve in the inoperable vent path.

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

SURVEILLANCE REQUIREMENTS

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

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

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

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.

IV. REACTOR BUILDING HIGH RADIATION MONITOR

Proposed Changes

Change Specification 3.3.3.1 and the Bases to reflect the addition of a new Reactor Building Area Monitor.

This monitor will provide high range data of radiological conditions within the Reactor Building during a large radioactive release type accident.

Reasons for the Proposed Change

This change is being made in response to NUREG-0737, Item II.F.1, Attachment 3. This system is expected to be operable prior to restart for Cycle V.

The Reactor Building High Radiation Monitor will provide operators with an indication of containment radiological conditions in the event of an accident.

Safety Analysis

The Reactor Building High Radiation Monitor requirements are consistent with and the January 28, 1983, Draft Generic Letter recommendations. This monitor is a post-accident instrument (i.e., wide range as opposed to operating or normal range) and will be used in making dose projections following radiological accidents. The addition of this monitor to the Technical Specifications will not degrade plant safety.

Should both channels of the Reactor Building High Radiation Monitor be inoperable, Florida Power Corporation presently has the capability to monitor the area over a range up to 10^5 rad/hr (RM-G19).

Additional Information

The initial in situ post-installation calibration of these monitors will utilize a "derived or secondary standard". This standard will be derived using a Condenser-R Meter traceable to the National Bureau of Standards. In addition, a uniform radiation field (at least one decade below 10 R/hr) will be created by placing the derived standard an appropriate distance from the new monitors. Immediately following the calibration using the derived standard, a laboratory standard will be placed in direct contact with the new monitor. The point of contact will be noted and the monitor response (corresponding to at least one decade below 10 R/hr) recorded for future calibrations. These future calibrations will be performed every refueling by placing the laboratory standard in direct contact with the monitor and verifying the response.

In situ calibration by electronic signal substitution is used for all range decades above 10 rad/hr.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool area Criticality Monitor	1	*	15 mr/hr	10^{-1}-10^{4}mr/hr	14
b. Reactor Building High Radiation Monitor	2	1,2,3,4	10^{2} rad/hr	1-10^{8} rad/hr	30
2. PROCESS MONITORS					
a. Fuel Storage Pool Area					
i. Gaseous Activity-Ventilation System Isolation	1	**	2 x background	10^{1}-10^{6} cpm	16
b. Reactor Building					
i. Gaseous Activity-					
a) Purge Exhaust Duct Isolation	1	6	***	10^{1}-10^{6} cpm	17
b) RCS Leakage Detection	1	1,2,3,4	Not Applicable	10^{1}-10^{6} cpm	15
ii. Iodine Activity-RCS Leakage Detection	1	1,2,3,4	Not Applicable	10^{1}-10^{6} cpm	15
c. Control Room					
i. Iodine Activity-Ventilation System Isolation/recirculation	1	All Modes	2 x background	10^{1}-10^{6} cpm	18

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

*** Determined by requirements of appendix "B", Tech. Specs., Section 2.4.2 - Crystal Piver 3 Operating License No. DPR-72.

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.12.

ACTION 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 18 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 30 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirements.

1) Either restore the inoperable Channels(s) to OPERABLE status with 7 days of the event, or

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area Criticality Monitor	S	R	M	*
b. Reactor Building High Radiation Monitor	S	R	M	1,2,3,4
2. PROCESS MONITORS				
a. Fuel Storage Pool Area				
i. Gaseous Activity - Ventilation System isolation	S	R	M	**
b. Reactor Building				
i. Gaseous Activity -				
a) Purge Exhaust Duct isolation	S	Q	M	6
b) RCS Leakage Detection	S	R	M	1,2,3,4
ii. Iodine Activity - RCS Leakage Detection	S	R	M	1,2,3,4
c. Control Room				
i. Iodine Activity - Ventilation System Isolation/ Recirculation	S	R	M	All Modes

* With fuel in the storage pool or building
** With irradiated fuel in the storage pool

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

The CHANNEL CALIBRATION of the Reactor Building High Radiation Monitor is performed *in situ* for at least one decade below 10 rad/hr. *In situ* calibration by electronic signal substitution is used for all range decades above 10 rad/hr.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. See Bases Figure 3-1 and 3-2 for examples of acceptable minimum incore detector arrangements.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event so that the response of those features important to safety may be evaluated. This capability is required to permit comparison of the measured response to that used in the design basis for the facility. This instrumentation is consistent with the recommendations of Safety Guide 12 "Instrumentation for Earthquakes), March 1971.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation dose to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs", February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of

V. REACTOR BUILDING WIDE RANGE PRESSURE MONITOR AND REACTOR BUILDING FLOOD LEVEL MONITOR

Proposed Changes

Add two new post-accident monitoring instruments to Specification 3.3.3.6. These new instruments are:

a. Reactor Building Wide Range Pressure Monitor

b. Reactor Building Flood Level Monitor.

The wider range for the pressure monitor is indicated under instrument #2 and the Flood Level is a new instrument #18.

Also, add a new column to Table 3.3-10, MINIMUM RECORDERS OPERABLE, that specifies which instruments must have recorders.

Reasons for the Proposed Change

These changes are being made in response to NUREG-0737, Item II.F.1, Attachments 4 and 5. These systems are being installed during Refuel IV and are expected to be cperable prior to restart for Cycle V.

Safety Analysis

The addition of these two post-accident monitoring instruments will upgrade plant safety. The new monitors can help operators mitigate the consequences of an accident should the normal operating monitors go off scale.

The Draft Generic Letter recommended specifications for the Wide Range Pressure Monitor and Flood Level Monitor. The specifications for these monitors are consistent with those in the Draft Generic Letter.

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels and recorders shown in Table 3.3-10 shall be OPERABLE with readouts on all channels in the control room.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-7.

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE	MINIMUM RECORDERS OPERABLE
1.	Power Range Nuclear Flux	0-125%	2	1
2.	Reactor Building Pressure	0-70 psia	2	1
		0-300 psia	2	1
3.	Source Range Nuclear Flux	10^{-1} to 10^{6} cps	2	1
4.	Reactor Coolant Outlet Temperature	520^{o} F - 620^{o} F	2 per loop	1 per loop
5.	Reactor Coolant Total Flow	0-160 x 10^{6} lb./hr.	1	1
6.	RC Loop Pressure	0-2500 psig	2	1
		0- 600 psig	1	0
		1700-2500 psig	2	1
7.	Pressurizer Level	0-320 inches	2	1
8.	Steam Generator Outlet Pressure	0-1200 psig	2/steam generator	1/steam generator
9.	Steam Generator Operating Range Level	0-100%	2/steam generator	1/steam generator
10.	Borated Water Storage Tank Level	0-50 feet	2	1
11.	Startup Feedwater Flow	0-1.5x10^{6} lb./hr.	2	0
12.	Reactor Coolant System Subcooling Margin Monitor	-685F^{o} to +668F^{o}	1	0
13.	PORV Position Indicator (Primary Detector)	N/A	1	0
14.	PORV Position Indicator (Backup Detector)	N/A	0	0
15.	PORV Block Valve Position Indicator	N/A	0	0
16.	Safety Valve Position Indicator (Primary Detector)	N/A	1/Valve	0
17.	Safety Valve Position Indicator (Backup Detector)	N/A	0	0
18.	Reactor Building Flood Level	0-10 feet	2	1

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	CHANNEL CALIBRATION
1.	Power Range Nuclear Flux	M	Q*
2.	Reactor Building Pressure	M	R
3.	Source Range Nuclear Flux	M	R*
4.	Reactor Coolant Outlet Temperature	M	R
5.	Reactor Coolant Total Flow Rate	M	R
6.	RC Loop Pressure	M	R
7.	Pressurizer Level	M	R
8.	Steam Generator Outlet Pressure	M	R
9.	Steam Generator Level	M	R
10.	Borated Water Storage Tank Level	M	R
11.	Startup Feedwater Flow Rate	M	R
12.	Reactor Coolant System Subcooling Margin Monitor	M	R
13.	PORV Position Indicator (Primary Detector)	M	R
14.	PORV Position Indicator (Backup Detector)	M	R
15.	PORV Block Valve Position Indicator	M	R
16.	Safety Valve Position Indicator (Primary Detector)	M	R
17.	Safety Valve Position Indicator (Backup Detector)	M	R
18.	Reactor Building Flood Level	M	R

*Neutron detectors may be excluded from CHANNEL CALIBRATION

VI. CONTAINMENT HYDROGEN MONITORS

Proposed Changes

Change the Specification 3.6.4.1 to require "two independent containment hydrogen monitors." The existing specification is more specific and requires one analyzer and a gas chromatograph. Recent plant modifications have installed two hydrogen analyzers on the waste sampling system.

Also, revise the Action Statement to state the action required when both hydrogen monitors are inoperable.

Reasons for the Proposed Change

This change is being made to reflect the addition of two independent, in-place hydrogen monitors and in response to NUREG 0737, Item II.F.1, Attachment 6.

The new in-place hydrogen monitors are expected to be operable prior to restart for Cycle V.

Safety Analysis

The change to the Hydrogen Monitor Specification will not degrade plant safety.

The NRC Draft Generic Letter, dated January 28, 1983, included recommendations for operability and surveillance requirements for the Hydrogen Monitoring system. The vised Hydrogen Monitor Specification is consistent with the recommended requirements except the exclusion of an ANALOG CHANNEL OPERATIONAL TEST. The requirement to perform a CHANNEL FUNCTIONAL TEST has been included in this specification. This surveillance requirement should provide adequate testing to assure monitor operability.

The requirement to commence plant shutdown if one hydrogen analyzer is inoperable for more than 30 days was also excluded. Two portable hydrogen analyzing units are available in case one of the required hydrogen monitors is inoperable. Thus, the ability to monitor hydrogen concentration in the Reactor Building by two independent monitoring units is available.

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.4.1 Two independent containment hydrogen monitors shall be OPERABLE

APPLICABILITY: MODES 1 and 2

ACTION:

With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen monitor shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gas containing:

a. One volume percent hydrogen, balance nitrogen.

b. Four volume percent hydrogen, balance nitrogen.

BASES

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. The purge system is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

In addition to the two inplace hydrogen monitors, there are two portable hydrogen analyzing units. In the event that one hydrogen monitor is inoperable, one of the portable units may be used to monitor the hydrogen concentration in the Reactor Building.

VII. TOXIC GAS DETECTION SYSTEM

Proposed Changes

Add a new Specification, 3.3.3.7, that specifies the operability and surveillance requirements for the Toxic Gas Detection System.

The Toxic Gas System monitors ammonia, sulfur dioxide and chlorine in the Control Room ventilation system to assure habitability requirements for the operators are satisfied.

Reasons for the Proposed Change

This change is being made in response to NUREG-0737, Item III.D.3.4. This new system has been installed and operating since Refuel III. It is being modified during Refuel IV to improve its reliability.

Safety Analysis

This change will upgrade plant safety. The Toxic Gas System is designed to protect the operators from toxic amounts of ammonia, sulfur dioxide, and chlorine by placing the ventilation system in the recirculation mode.

The Toxic Gas system specification is consistent with the recommendations of the January 28, 1983, Draft Generic Letter, with the exception that the Crystal River system monitors ammonia and sulfur dioxide as well as chlorine.

Another deviation from the recommended specification is the exclusion of an ANALOG CHANNEL OPERATIONAL TEST. To replace this requirements, a CHANNEL FUNCTIONAL TEST shall be performed every 31 days. A CHANNEL FUNCTIONAL TEST will provide adequate assurrance of monitor operability.

TOXIC GAS SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.8 Two independent Toxic Gas systems shall be operable with their alarm/trip setpoints adjusted to actuate at:

(a) chlorine concentration of less than or equal to 5 ppm,

(b) sulfur dioxide concentration at less than or equal to 5 ppm,

(c) ammonia concentration of less than or equal to 100 ppm.

APPLICABILITY: ALL MODES

ACTION:

a. With one toxic gas system inoperable, restore the inoperable detection system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the Control Room emergency ventilation system in the recirculation mode of operation.

b. With both toxic gas systems inoperable, within 1 hour initiate and maintain operation of the Control Room emergency ventilation system in the recirulation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each toxic gas system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per Refueling Cycle.

REMOTE SHUTDOWN INSTRUMENTATION (Continued)

HOT STANDBY of the facility from locations outside of the Control Room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of Appendix "A", 10 CFR 50.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant paramenters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumenation for Light-Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident", December 1975.

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

The OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is returned to service.

3/4.3.3.8 TOXIC GAS SYSTEMS

The OPERABILITY of the Toxic Gas System ensures that the Control Room operators will be adequately protected against the effects of the accidental release of toxic gas and that the plant can be safety operated or shutdown under such conditions.

While Florida Power Corporation is not committed to Regulatory Guides 1.78 and 1.95 these have provided specific guidance for practices that may mitigate the hazards to the Control Room operators from an accidental chemical release.

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 (4.6.3.1.3) requiring that these valves be verified closed every 31 days when in MODES 1, 2, 3 and 4.

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. The Crystal River Unit 3 purge valves are closed and verified closed every 31 days when in MODES 1, 2, 3 and 4. 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.

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.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per Refueling Cycle by:

a. Verifying that on a containment isolation test signal, each automatic isolation valve actuates to its isolation position.

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.

TABLE 3.6-1 (continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
	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
13.	WSV-3	iso. containment monitoring system from RB	60
	WSV-4	"	60
	WSV-5	"	60
	WSV-6	"	60

B. CONTAINMENT PURGE AND EXHAUST

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

C. MANUAL

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

TABLE 3.6-1 (continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
D. PENETRATIONS REQUIRING TYPE B TESTS		
Blind Flange 119	iso. RB	NA
Blind Flange 120	"	NA
Blind Flange 202	"	NA
Blind Flange 348	iso. fuel tranfer tube from Transfer Canal	NA
Blind Flange 436	"	NA
Equipment Hatch	iso. RB	NA
Personnel Hatch	"	NA

\# 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