Florida Power
CORPORATION

October 31, 1989
3F1089-23

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

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Technical Specification Change Request No. 174
Pressure/Temperature Limits
Generic Letter 88-11 Submittal

Dear Sir:

Florida Power Corporation (FPC) hereby submits Technical Specification Change Request No. (TSCRN) 174 Revision 0, requesting amendment to Appendix A of Operating License No. DPR-72. Proposed replacement pages for both the current CR-3 Technical Specifications and the Improved Technical Specifications are provided. Replacement pages for the associated bases are also provided.

This change request proposes Heatup, Cooldown and Inservice Leak and Hydrostatic Testing pressure/temperature limits based on 15 effective full power years (EFPY) of reactor operation. This submittal also proposes a low temperature overpressurization protection (LTOP) features Technical Specification for Crystal River Unit 3 (CR-3). Previous FPC commitments for administrative controls regarding LTOP are superseded by this submittal. The LTOP approach, discussed with the NRC staff in a May 16, 1989 meeting between FPC, the NRC, and Babcock and Wilcox (B&W), has been developed utilizing non-10CFR50 Appendix G methodology. The non-Appendix G approach is based on the low probability of occurrence for an LTOP-type event at CR-3 and is consistent with the NRC position in Generic Letter 88-11. The attached engineering document that supports the use of the non-Appendix G methodology has been determined to be proprietary to B&W in accordance with 10CFR2.790. The Affidavit of Mr. James H. Taylor, B&W's Manager of Licensing Services, which identifies the summary report as B&W proprietary, has been enclosed.

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Change: NRC POR - Non Prop Only
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A Florida Progress Company

October 31, 1989
3F1089-23

This submittal completes FPC's commitment on Generic Letter 88-11 "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations". (FPC letter 3F1188-15 to the NRC, dated November 23, 1988; modified in FPC letter 3F0889-10 to the NRC, dated August 14, 1989). In the Generic Letter 88-11 response, FPC committed to the use of Regulatory Guide 1.99 Revision 2, in the development of the 15 EFPY pressure/ temperature limits for CR-3. Both the Appendix G pressure/ temperature limits and the LTOP pressure/ temperature limit curve have been calculated utilizing the methods described in Regulatory Guide 1.99 Revision 2 to predict the effect of neutron irradiation on reactor vessel material properties.

FPC requires NRC review and approval of the proposed changes by January 1991 in order to continue to operate CR-3, as current 8 EFPY pressure/ temperature limits are projected to expire at that time. However, the current CR-3 Technical Specification Improvement Program lead plant schedule may necessitate NRC approval of the LTOP Technical Specification by June 1990. FPC requests this amendment become effective 30 days after issuance in order to allow for procedure changes and training.

Sincerely,

Gary Boldt

Gary Boldt, Vice President
Nuclear Production

GLB/BPW

Attachment

xc: Regional Administrator, Region II
Senior Resident Inspector

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

CERTIFICATE OF SERVICE

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

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

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

A copy of Technical Specification Change Request No. 174, Revision 0, requesting Amendment to Appendix A of Operating License No. DPR-72.

FLORIDA POWER CORPORATION

Gary Boldt

Gary Boldt, Vice President
Nuclear Production

SWORN TO AND SUBSCRIBED BEFORE ME THIS 31ST DAY OF OCTOBER 1989.

Anna Maria Di Lorna
Notary Public

Notary Public, State of Florida at Large
My Commission Expires: 10/19/92

NOTARY PUBLIC, STATE OF FLORIDA AT LARGE
MY COMMISSION EXPIRES OCT. 19, 1992
BONDED THROUGH ASHTON AGENCY INC

STATE OF FLORIDA

COUNTY OF CITRUS

Gary Boldt states that he is the Vice President, Nuclear Production for Florida Power Corporation; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Gary Boldt

Gary Boldt, Vice President
Nuclear Production

Subscribed and sworn to me, a Notary Public in and for the State and County above named, this 31st day of October, 1989.

Anna Maria Di Roma

Notary Public

Notary Public, State of Florida at Large
My Commission Expires: 10/19/92

NOTARY PUBLIC, STATE OF FLORIDA AT LARGE
MY COMMISSION EXPIRES OCT. 19, 1992
BONDED THROUGH ASHTON AGENCY INC

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302/ LICENSE NO. DPR-72
REQUEST NO. 174, REVISION 0
PRESSURE/TEMPERATURE LIMITS

A. LICENSE DOCUMENT INVOLVED: Technical Specifications

PORTION:	Index, Page V	/ TSIP Index Not affected
	Index, Page IX	/ TSIP Index Not affected
	3.4.12	/ TSIP Specification 3.3.8
	3.5.2	/ TSIP Specification 3.4.2
	3.5.3	/ TSIP Specification 3.4.3
	3.4.3.2	/ TSIP Specification 3.3.10 Not Affected
	3.1.2.4.1	/ TSIP Specification (N/A)
	N/A	/ TSIP Specification 3.3.6

DESCRIPTION OF REQUEST:

The proposed change would add a Limiting Condition for Operability (LCO) for Low Temperature Overpressurization Protection Features to Technical Specifications.

The change also allows two high pressure injection pumps to be deactivated in MODE 3 and MODE 4 in accordance with Limiting Condition for Operation (LCO) 3.4.12 Low Temperature Overpressure Protection Features. The submittal proposes adding a note to the MODE applicability of LCO 3.1.2.4.1 to allow two makeup/ HPI pumps to be deactivated in accordance with LCO 3.4.12 Low Temperature Overpressure Protection Features.

A note is also provided to LCO 3.4.3.2 to alert the operator that Pressure Operated Relief Valve (PORV) OPERABILITY for RCS pressure control is also required by LCO 3.4.12 Low Temperature Overpressure Protection Features.

The proposed change also re-defines the MODE APPLICABILITY section of Technical Specification Improvement Program (TSIP) LCO 3.3.6 Pressurizer Water Level to read MODES 1, 2, and MODE 3 with RCS temperature > 283°F.

REASON FOR REQUEST:

Low temperature overpressurization protection (LTOP) features ensure the reactor coolant pressure boundary (RCPB) is adequately protected at low reactor coolant system (RCS) temperatures. The RCPB is one of the primary boundaries to fission product release and protective features that insure its' integrity should be included in Technical Specifications. Additionally, proposed LTOP

features are based on non-10CFR50 Appendix G methodolgy that utilize reduced margins as part of the analysis. To ensure the plant is operated within the bounds of the analysis, the critical analysis assumptions are preserved in Technical Specifications.

With reactor coolant system (RCS) temperature less than 283°F, inadvertent high pressure injection (HPI) must be administratively precluded for low temperature overpressure protection (LTOP). HPI actuation could potentially pressurize the RCS in excess of the allowable LTOP fracture mechanics limits. Low temperature overpressure protection requires that two HPI trains be deactivated whenever RCS temperature is less than or equal to 283°F. This is in conflict with LCO 3.1.2.4.1 which requires two OPERABLE makeup pumps when RCS temperature is greater than 280°F.

The LTOP LCO also contains requirements on PORV OPERABILITY for RCS pressure control which must be preserved and appropriate actions to be taken in the event the requirements are not met. The PORV is one of the required redundant LTOP features and must be OPERABLE to provide overpressure protection in an LTOP-type event.

At RCS temperatures less than or equal to 283°F, LCO 3.4.12 requires additional restrictions on the pressurizer water level than provided in LCO 3.3.6. This is necessary to ensure the low temperature overpressure protection features are maintained.

EVALUATION OF REQUEST:

In Generic Letter 88-11, the Nuclear Regulatory Commission indicated it would consider the use of non-10CFR50 Appendix G methodology for developing low temperature overpressurization protection (LTOP) limits. The reduced analysis margins used in the non-Appendix G methodology were intended to make the amount of conservatism in the LTOP limits more representative of the risk. Florida Power Corporation (FPC) pursued this opportunity because relaxed LTOP limits result in more operational flexibility and generally lower operator burden. To justify use of the non-Appendix G method, FPC has 1) demonstrated the frequency of occurrence for a low temperature overpressure event that would exceed Appendix G limits is less than one per reactor lifetime, and 2) developed adequate safety justification in regards to the appropriate fracture toughness limits.

Crystal River Unit 3 (CR-3) has two design features/ operating practices which contribute to the low probability of occurrence for an LTOP-type event. The first of these is the operating practice of maintaining a gas or steam bubble in the pressurizer at all times (except system hydrotest). This provides a surge volume, which unlike the "water solid" system, can accommodate most pressure transients. The "water solid" system has the potential for almost instantaneous pressure increase for mass addition

transients and significantly faster pressure increases for energy addition transients. Operating experience throughout the nuclear power industry has shown that most all LTOP-type events occurred during heatup and cooldown during a "water solid" condition. Secondly, at low RCS pressures, other pressurized water reactor (PWR) designs are configured so that an inadvertent isolation of the Residual Heat Removal System will terminate letdown flow from the RCS. RCS pressure increases as the makeup pump continues to operate, and eventually the RCS is overpressurized. The CR-3 design does not route letdown flow through the Decay Heat Removal system, and is not susceptible to this type of event.

The basis for LCO 3.3.6 Pressurizer Water Level (TSIP format) requiring MODE 3 and MODE 4 with RCS temperature greater than or equal to 275°F is to prevent solid water RCS operation during plant heatup and cooldown. The cushion effect of this steam space prevents rapid RCS pressure rises due to normal plant perturbations. The revision to this LCO will require a more restrictive limit on pressurizer water level over the range of RCS temperatures from 283°F to 275°F and continue to maintain the remainder of the existing requirements. Lowering pressurizer water level to that required by LCO 3.4.12 (220 inches), at the RCS temperature of 283°F, results in a larger margin to solid water operation and a larger steam space to cushion the RCS against overpressure transients.

The LTOP LCO was written to ensure redundant LTOP subsystems are operational when the LTOP features are required for protection of the RCPB. The primary means of LTOP is operator action to terminate the event. For conservative purposes, the analysis demonstrates, and the plant is operated, such that the operator has at least 10 minutes from the time an alarm is received until RCS pressure exceeds the LTOP fracture toughness (non-Appendix G) limits. As a backup to operator action, the pressure operated relief valve (PORV), with the reduced pressure setpoint selected, is set to relieve at an RCS pressure less than that corresponding to the minimum LTOP fracture toughness limits. The LCO is written to 1) ensure the PORV is OPERABLE, and 2) place restrictions on plant operations that ensure 10 minutes is available for operator action. Remedial actions have been provided to ensure that in the event the LTOP LCO is violated, specific steps will be taken to place the plant in a non-applicable condition. Surveillance Requirements ensure the limitations are maintained.

With the PORV inoperable in MODES 1,2 or 3, LCO 3.4.3.2 action (a) directs the operator to restore the PORV to OPERABLE within one hour or close and remove power to the block valve. Failing this, the plant shall be placed in HOT STANDBY in the next 6 hours and COLD SHUTDOWN in the following 30 hours. This action places the plant into the LTOP LCO which has separate requirements for PORV OPERABILITY and appropriate actions to take if the PORV is

inoperable. The note added to the PORV LCO provides a cross-reference between the two specifications. Providing the operator with the note on other PORV operability requirements ensures the operator will expeditiously proceed to LCO 3.4.12 Low Temperature Overpressurization Protection Features and carry out the appropriate actions for an inoperable PORV in these MODES.

10CFR50 Appendix G requires the operator to maintain RCS temperature and pressure within the limits of LCO 3.4.9 Pressure/ Temperature Limits at all times. These limits are based on ASME Section III, Appendix G assumptions for normal occurrences and the plant must be operated within these limits. The non-Appendix G curve is only used to determine the 10 minute "window" and set PORV setpoint. The curve is contained in the engineering document supporting the non-Appendix G limits and does not appear in procedures or within Technical Specifications. This will eliminate any confusion which might occur as a result of two pressure/ temperature limit curves.

Two analyzed overpressure transients resulted in RCS pressure increases that exceed LTOP limits in less than 10 minutes. These transients are high pressure injection (HPI) and core flood tank (CFT) actuation. The two events must be administratively precluded from occurring, and this is the bases for requiring both trains of each system to be deactivated in the LTOP region. With the HPI and CFT actuation precluded, the limiting analysis case becomes the full-open failure of the makeup control valve. This event is the basis for PORV setpoint, pressurizer level, and allowable makeup tank level.

Technical Specification LCO 3.1.2.4.1 requires at least two OPERABLE makeup pumps in MODE 3 (RCS temperature greater than or equal to 280°F). LCO 3.4.12 Low Temperature Overpressure Protection Features requires two trains of HPI be deactivated whenever RCS temperature is less than or equal to 283°F to ensure an inadvertent Engineered Safeguards (ES) actuation does not result in overpressurizing the RCS. This creates a conflict between the two LCOs since the HPI pumps also serve as the normal makeup pumps, whereby deactivating the two trains of HPI leaves only one OPERABLE makeup pump. Deactivating the two trains of HPI at 283°F, also causes the operator to enter the action statement for LCO 3.1.2.4.1. This LCO provides 72 hours for the operator to restore at least two makeup pumps to OPERABLE or take action to place the plant in a safe MODE of operation where the LCO does not apply. The allowable out-of-service period ensures that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failure during the repair period. The amount of time the plant remains between the temperatures of 283°F and 280°F is expected to be much less than 72 hours and once the plant is in MODE 4 (RCS temperature less than 280°F) only one makeup pump is required, and LCO 3.1.2.4.1 is no

Current Technical Specifications allow the HPI injection isolation valve power supply breakers to be "racked out" in MODE 4. Based on the updated LTOP analysis, HPI should now be deactivated whenever RCS temperature is equal to or less than 283°F (MODE 3) and the reactor vessel head is not fully detensioned. "Racking out" the power supplies for the HPI isolation injection valves still allows manual initiation of HPI by "racking in" the breakers and then operating the valves (The footnote for current CR-3 LCO 3.5.3 has been reworded to clarify the meaning of "racking out" as it applies to the actual plant practice for deactivating HPI isolation injection valves). However, HPI would not be immediately available since operator action outside the control room is required. FPC has evaluated alternative methods for deactivating HPI that do not require "racking out" power supply breakers. These methods would provide the operator with the capability of restoring HPI without leaving the control room and could be used at the higher LTOP temperatures to make HPI more readily available in the event it is needed. To further balance the possible need for core cooling, LCO 3.4.12 does not require the makeup system to be deactivated. At the lower temperatures associated with LTOP, and the expected decay heat levels, the makeup system can provide adequate flow via the makeup control valve, or it can be used in the interim until HPI can be re-activated.

Raising the RCS temperature at which HPI is deactivated from 280°F to 283°F is considered acceptable. This conclusion is based on 1) the low probability of a loss of coolant accident requiring immediate HPI flow in this region of plant operation, 2) the limited amount of time the plant is actually operated between the RCS temperatures of 280 and 283F, 3) the ready accessibility of RCS makeup flow and HPI, and 4) the improved LTOP protection.

The LTOP fracture toughness limits are based on 21 Effective Full Power Years (EFPY) of reactor operation for determining neutron fluence values for the reactor vessel. Similar to the Appendix G heatup and cooldown limits, the vessel is the limiting RCS component in terms of LTOP fracture toughness. Regulatory Guide 1.99 Revision 2 was utilized to determine the shift in reference nil ductility transition temperature due to neutron irradiation. This method was endorsed in Generic Letter 88-11 as an acceptable method for predicting the shift in material behavior. The analysis used the heatup and cooldown rates assumed in generating the 15 EFPY Appendix G pressure/ temperature limit curves (listed in Pressure Temperture Limits LCO 3.4.9).

SHOLLY EVALUATION OF REQUEST:

Florida Power Corporation (FPC) proposes the addition of a Low Temperature Overpressurization Protection (LTOP) Features Technical Specification does not involve a significant hazards consideration. The addition of a Technical Specification requirement to maintain the LTOP features ensures the reactor coolant pressure boundary is protected against a non-ductile failure at low reactor coolant temperatures.

Based on the above, FPC concludes this change will not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated since there are currently no LTOP requirements in Technical Specifications. This change represents additional requirements necessary to preclude an LTOP event from occurring. These additional requirements also provide protection for all pressure and temperature combinations for which an LTOP event may be postulated. Overall, these requirements provide a level of protection greater than or equivalent to existing requirements.

2. Create the possibility of a new or different kind of accident from any previously evaluated because the addition of an LTOP Technical Specification does not require modification to the plant nor does it create a new mode of plant operation, with the exception of the small increase in reactor coolant temperature at which the high pressure injection system is deactivated. This is considered acceptable based on the small amount of time the plant is operated in this temperature region and the low probability of a loss of coolant accident (LOCA) requiring immediate high pressure injection flow at the reactor coolant temperatures in this region. In the unlikely event a LOCA does occur, high pressure injection would be available following operator restoration of the system. Reactor coolant makeup flow would be available in the interim to provide core cooling requirements.

3. Involve a significant reduction in the margin of safety. Any reduction in the margin of safety will be insignificant and offset by the safety benefit gained through the additional requirements placed on plant operation to preclude a low temperature overpressurization event.

TSCRN 174A
CURRENT TECHNICAL SPECIFICATION FORMAT

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 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% Δk/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.

* Two makeup/HPI pumps may be deactivated in accordance with Specification 3.4.12.

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 Specification 3.0.5, the PORV shall be demonstrated OPERABLE at least one per 18 months 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.

* PORV OPERABILITY for pressure control is required by Specification 3.4.12.

LOW TEMPERATURE OVERPRESSURIZATION PROTECTION FEATURES

LIMITING CONDITION FOR OPERATION

3.4.12 Low Temperature Overpressurization Protection (LTOP) features shall be OPERABLE and shall be comprised of:

a. Pressurizer level less than or equal to 220 inches,
b. OPERABLE power operated relief valve (PORV) with a setpoint of less than or equal to 555 psig,
c. Two trains of High Pressure Injection (HPI) deactivated, and
d. Two core flood tanks (CFT) isolated with the isolation valve closed and the power supply breakers fixed in the open position whenever CFT pressure is greater than or equal to the maximum allowable reactor coolant (RC) pressure for the existing RC temperature (per PT limits shown in Figures 3.4-2 and 3.4-3).

APPLICABILITY: MODE 3 with RCS temperature ≤ 283°F, MODE 4, MODE 5, and MODE 6 with the reactor vessel head not completely detensioned.

ACTION:

a. With pressurizer level greater than 220 inches, restore pressurizer level to less than or equal to 220 inches within one hour, or close and maintain closed the makeup control valve and its associated isolation valve and stop plant heatup within the next 12 hours.

b. With the PORV inoperable, restore the PORV to OPERABLE status within one hour or reduce makeup tank level to less than or equal to 70 inches and deactivate Low-Low makeup tank level interlock to the BWST suction valves within the next 12 hours.

c. With one or two HPI trains active, deactivate the HPI trains within one hour or close and remove power from the HPI injection valves (MUV-23,MUV-24,MUV-25, and MUV-26) within the next 12 hours.

d. With one or two CFTs not isolated when CFT pressure is greater than the maximum allowable RCS pressure for the existing RCS temperature, isolate the affected CFT(s) within one hour or within the next 12 hours, increase RCS temperature above 175°F or depressurize CFT to less than 555 psig.

e. With the LTOP features inoperable for reasons other than above, restore the LTOP features within one hour or depressurize the RCS to atmospheric pressure and establish an RCS vent equivalent to an orifice area of 0.75 square inches and verify both trains of HPI are deactivated and verify both CFTs are isolated with the isolation valve closed and the power supply breaker fixed in the open position within the following 12 hours.

f. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.12 The LTOP features shall be demonstrated OPERABLE:

a. By verifying pressurizer level is less than or equal to 220 inches at least once per 30 minutes during heatup and cooldown, otherwise verify pressurizer level is less than or equal to 220 inches at least once per 12 hours.

b. By verifying the PORV block valve is open at least once per 12 hours.

c. By verifying two HPI trains deactivated at least once per 12 hours.

d. By verifying two CFTs are isolated at least once per 12 hours.

e. By performing a CHANNEL FUNCTIONAL TEST of the PORV at least once per 31 days .

f. By performing a CHANNEL CALIBRATION of the PORV at least once per 18 months.

g. By verifying at least once per 12 hours that an RCS vent is open when required by ACTION "e" above.

ECCS SUBSYSTEMS - $T_{avg} \geq 280°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure injection (HPI) pump,*

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) on a safety injection signal and manually transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT STANDBY within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to the Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

* Two high pressure injection pumps may be deactivated in accordance with Specification 3.4.12

ECCS SUBSYSTEMS - $T_{avg} > 280°F$

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE high pressure injection (HPI) pump,*

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path** capable of taking suction from the borated water storage tank (BWST) on a safety injection signal and manually transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With one ECCS subsystem OPERABLE because of the inoperability of either the HPI pump or the flow path from the borated water storage tank, restore at least one ECCS subsystem to the OPERABLE status within one hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the decay heat cooler or LPI pump, restored at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 280°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and in 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystems shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2

* Two high pressure injection pumps may be deactivated in accordance with Specification 3.4.12.

** The high pressure injection isolation valves may be closed with their power supply breakers fixed in the open position in MODE 4.

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.

3/4.4.12 Low Temperature Overpressurization Protection Features

Low temperature overpressurization protection (LTOP) features ensure adequate overpressure protection for the reactor vessel at low reactor coolant system (RCS) temperatures and pressures. This is especially important since low carbon steels typical of reactor vessel materials, have reduced fracture toughness and are more susceptible to non-ductile failures at lower RCS temperatures.

Traditionally, 10CFR50 Appendix G heatup and cooldown pressure/ temperature limits have served as the LTOP pressure/ temperature limits as well. This meant LTOP administrative restraints were based on normal heatup and cooldown pressure/ temperature limits. 10CFR50 requires fracture toughness limits be established that provide protection of the reactor coolant pressure boundary during any condition of normal operation, including events anticipated to occur one or more times during the reactor lifetime. This includes normal heatup and cooldown of the RCS and as noted above, also included LTOP. A review of CR-3 plant design, plant operation, and historical data revealed the frequency of occurrence for a low temperature overpressure event that would exceed 10CFR50 Appendix G limits was considerably less than one per reactor lifetime. Based on the frequency of occurrence, the use of non-Appendix G limits (utilizing reduced design margins) was justified in order to establish appropriate plant limitations for LTOP.

The RCS temperature where low temperature overpressurization is required is determined based on the minimum pressurization temperature. Above this temperature, LTOP protection is not required since an overpressurization transient will be terminated by the pressurizer safety valves (PSV) before the LTOP pressure/ temperature limits are exceeded. Below the minimum pressurization temperature, LTOP pressure/ temperature limits would be exceeded before the PSV relieve. Reactor coolant pressure boundary protection is based on LTOP features in this region. Additionally, LTOP protection is not required when the reactor vessel head is completely detensioned, as overpressurization of an "open" RCS is not considered credible.

Operator action is assumed as the primary means for terminating RCS pressure increase due to a low temperature overpressurization event. To ensure a conservative amount of time for the operator to take action, administrative limits on plant operation are implemented which provide 10 minutes prior to exceeding LTOP pressure/ temperature limits. As a backup to operator action, the power operated relief valve (PORV) with reduced pressure setpoint, actuates to relieve RCS pressure increases before exceeding LTOP limits.

CR-3 is designed, except for system hydrotest, to be operated with a gas or steam bubble in the pressurizer at all times. The gas or steam space acts as a surge volume and limits the rate of RCS pressure increase in the event of an overpressurization event. Pressurizer level is a direct indication of the amount of steam space available and must be limited to ensure a 10 minute operator action time is preserved.

The maximum allowed PORV setpoint for LTOP is derived from the LTOP pressure/ temperature limits. Operation with a setpoint less than or equal to the minimum LTOP pressure/ temperature limit setpoint ensures that the non-Appendix G criteria will not be violated. System pressure overshoot, that is, increase in RCS pressure after pressure reaches the PORV setpoint, does not occur due to the rapid action of the PORV and the relatively slow rates of pressure increase due to the pressurizer steam bubble.

Two of the analyzed overpressurization transients must be administratively precluded, since the resultant RCS pressure increase exceeds the LTOP limits in less than 10 minutes. The two events are inadvertent actuation of high pressure injection (HPI) and core flood tank (CFT) actuation. With these transients precluded as credible events, a full open failure of the makeup control valve becomes the limiting overpressurization event on which the LTOP features are based.

The LCO permits HPI surveillance testing for components since pump or valve testing can proceed by alternating the system deactivation from pumps to valves. Testing must not permit HPI injection flow to enter the RCS.

The LTOP pressure/ temperature limits are based on 21 Effective Full Power Years (EFPY) of reactor operation and will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens, as required by 10CFR50 Appendix H. The RCS heatup and cooldown rates used to develop the limits are the same as listed in LCO 3.4.9 "Pressure Temperature Limits".

TSCRN 174A
TSIP FORMAT

3.3 REACTOR COOLANT SYSTEM (RCS)

3.3.6 Pressurizer Water Level

LCO 3.3.6 Pressurizer water level shall be ≤ 290 inches.

APPLICABILITY: MODES 1, 2 and
MODE 3 with RCS temperature > 283°F

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level > 290 inches.	A.1 Restore pressurizer water level to within limit.	1 hour
B. Required Action NOT met within required Completion Time.	B.1 Be in MODE 3. AND B.2 Be in MODE 3 with RCS temperature ≤ 283°F.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Verify pressurizer water level ≤ 290 inches.	12 hours

3.3 REACTOR COOLANT SYSTEM (RCS)

3.3.8 Low Temperature Overpressurization Protection Features

LCO 3.3.8 Low temperature overpressurization protection (LTOP) features shall be OPERABLE and shall be comprised of:

a. Pressurizer level ≤ 220 inches, and

b. OPERABLE power operated relief valve (PORV) with a setpoint of ≤ 555 psig, and

c. Two trains of high pressure injection (HPI) deactivated, and

d. Two core flood tanks (CFT) isolated with the isolation valve closed and the power supply breakers fixed in the open position.

----------NOTE----------
Provisions of LCO 3.0.3 are not applicable.

APPLICABILITY: RCS temperature ≤ 283°F and the reactor vessel head not completely detensioned.

----------NOTE----------
CFT isolation only required when CFT pressure is ≥ to the maximum allowable reactor coolant (RC) pressure for the existing RC temperature (in accordance with PT Limit Curves provided in the PRESSURE/TEMPERATURE LIMITS REPORT).

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LTOP features inoperable due to pressurizer level > 220 inches.	A.1 Restore pressurizer level to ≤ 220 inches.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action A.1 NOT met within required Completion Time.	B.1 Close and maintain closed the makeup control valve and its associated isolation valve. AND B.2 Stop plant heatup.	12 hours 12 hours
C. LTOP features inoperable due to PORV inoperability.	C.1 Restore PORV to OPERABLE status.	1 hour
D. Required Action C.1 NOT met within required Completion Time.	D.1 Reduce makeup tank level to ≤ 70 inches, AND D.2 Deactivate Low-Low makeup tank level interlock to the BWST suction valves.	12 hours 12 hours
E. LTOP features inoperable due to one or two HPI trains active.	E.1 Deactivate HPI trains.	1 hour
F. Required Action E.1 NOT met within required Completion Time.	F.1 Close and remove power from HPI injection valves (MUV-23, MUV-24, MUV-25, and MUV-26.)	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. LTOP features inoperable due to 1 or 2 CFTs not isolated when CFT pressure is > the maximum allowable RCS pressure for existing temperature.	G.1 Isolate affected CFTs.	1 hour
H. Required Action G.1 NOT met within required Completion Time.	H.1 Increase RCS temperature above 175°F.	12 hours
	OR	
	H.2 Depressurize CFTs to < 555 psig.	12 hours
I. LTOP features inoperable for reasons other than above.	I.1 Restore LTOP features.	1 hour
	OR	
	I.2.1 Depressurize RCS to atmospheric pressure and establish an RCS vent equivalent to an orifice area of 0.75 square inches.	12 hours
	AND	
	I.2.2 Verify two trains of HPI deactivated.	12 hours
	AND	
	I.2.3 Verify two CFTs are isolated with the isolation valve closed and the power supply breaker fixed in the open position.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Verify pressurizer level ≤ 220 inches	30 minutes during RCS heatup and cooldown AND 12 hours
SR 3.3.8.2	Verify PORV block valve is open.	12 hours
SR 3.3.8.3	Verify HPI deactivated.	12 hours
SR 3.3.8.4	Verify CFTs isolated.	12 hours
SR 3.3.8.5	----------NOTE---------- Required only when complying with Required Action I.2. ---------- Verify RCS vent open.	12 hours
SR 3.3.8.6	Perform CHANNEL FUNCTIONAL TEST of PORV.	31 days
SR 3.3.8.7	Perform CHANNEL CALIBRATION of PORV.	18 months

3.3 REACTOR COOLANT SYSTEM (RCS)

3.3.9 Pressurizer Safety Valves

LCO 3.3.9 Two pressurizer code safety valves shall be OPERABLE with lift settings ≥ 2475 psig and ≤ 2525 psig.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer code safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action NOT met within required Completion Time.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Verify pressurizer code safety valves OPERABLE in accordance with SR 3.0.5.	In accordance with SR 3.0.5.

3.3 REACTOR COOLANT SYSTEM (RCS)

3.3.10 Relief Valves

LCO 3.3.10 One power operated relief valve (PORV) and its block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. PORV inoperable.	A.1 Restore PORV to OPERABLE status.	1 hour
	OR	
	A.2 Close the block valve.	1 hour
B. Block valve inoperable.	B.1 Restore affected components to OPERABLE status.	1 hour
OR	OR	
PORV and block valve inoperable.	B.2.1 Close block valve.	1 hour
	AND	
	B.2.2 Remove power from block valve.	1 hour
	OR	
	B.3.1 Close PORV.	1 hour
	AND	
	B.3.2 Remove power from PORV.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action NOT met within required Completion Time.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	----------NOTE---------- Surveillance NOT required with block valve closed in accordance with the Required Actions of LCO 3.3.10. ---------- Operate block valve through one complete cycle of travel.	92 days
SR 3.3.10.2	Perform CHANNEL CALIBRATION for PORV.	18 months

3.3 REACTOR COOLANT SYSTEM (RCS)

3.3.11 RCS Leakage

LCO 3.3.11 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE;
b. 1 gpm UNIDENTIFIED LEAKAGE;
c. 10 gpm IDENTIFIED LEAKAGE from the RCS.

----------NOTE----------
Primary to secondary leakage addressed in LCO 3.3.12 is included in the 10 gpm IDENTIFIED LEAKAGE from the RCS.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS leakage outside limits for reasons other than PRESSURE BOUNDARY LEAKAGE.	A.1 Reduce leakage to within limits.	4 hours
B. Required Action A.1 NOT met within required Completion Time.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. PRESSURE BOUNDARY LEAKAGE exists.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.11.1	----------NOTE---------- Provisions of SR 3.0.4 are not applicable. ---------- Perform reactor coolant system water inventory balance.	72 hours during steady state operation.

3.3 REACTOR COOLANT SYSTEM (RCS)

3.3.12 RCS Primary to Secondary Leakage

LCO 3.3.12 RCS primary to secondary leakage shall be ≤ 1 gpm total through both steam generators.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS primary to secondary leakage > 1 gpm.	A.1 Reduce leakage rate to within limit.	4 hours
B. Required Action NOT met within required Completion Time.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.12.1	----------NOTE---------- Provisions of SR 3.0.4 are not applicable. ---------- Verify primary to secondary leakage ≤ 1 gpm.	72 hours during steady state operation.

3.3 REACTOR COOLANT SYSTEM (RCS)

3.3.13 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.3.13 Leakage for each RCS PIV listed below shall be ≤ 5.0 gpm.

1. CFV-1
2. CFV-3
3. DHV-1
4. DHV-2

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Leakage for one or more RCS PIV > 5.0 gpm.	A.1 Reduce leakage rate within limit.	4 hours
B. Required Action NOT met within required Completion Time.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.13.1	----------NOTE---------- Provisions of SR 3.0.4 are not applicable for entry into MODES 3 and 4 for the purpose of testing the isolation check valves. ---------- Verify leakage for each PIV ≤ 5.0 gpm.	Prior to entering MODE 2 whenever the plant has been in MODE 5 for 72 hours or more, and if leakage testing has not been performed in the previous 9 months. <u>AND</u> In accordance with SR 3.0.5.

3.3 REACTOR COOLANT SYSTEM (RCS)

3.3.14 RCS Leakage Detection Instrumentation

LCO 3.3.14 The following RCS leakage detection instruments shall be OPERABLE:

a. Containment atmosphere activity monitor, and
b. Containment sump level monitor.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required containment atmosphere activity monitor inoperable.	A.1 Take and analyze grab samples.	Once per 24 hours.
	AND	
	A.2 Restore monitor to OPERABLE status.	30 days
B. Containment sump level monitor inoperable.	B.1 Perform SR 3.3.11.1.	Once per 24 hours.
	AND	
	B.2 Restore monitor to OPERABLE status.	30 days
C. Required Action NOT met within required Completion Time.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.14.1	Monitor containment atmosphere activity.	12 hours
SR 3.3.14.2	Monitor containment sump level.	12 hours
SR 3.3.14.3	Perform a CHANNEL CHECK of containment atmosphere activity monitor.	12 hours
SR 3.3.14.4	Perform a CHANNEL FUNCTIONAL TEST of containment atmosphere activity monitor.	31 days
SR 3.3.14.5	Perform a CHANNEL CALIBRATION of containment atmosphere activity monitor.	18 months
SR 3.3.14.6	Perform a CHANNEL CALIBRATION of containment sump level monitor.	18 months

3.3 REACTOR COOLANT SYSTEM (RCS)

3.3.15 Specific Activity

LCO 3.3.15 The specific activity of the primary coolant shall be $\leq$ 1.0 microcurie/gram DOSE EQUIVALENT I-131, and $\leq$ 100/$\bar{E}$ microcuries/gram.

APPLICABILITY: MODES 1 and 2.
MODE 3 with $T_{avg} \geq$ 500°F.

----------NOTE----------
Provisions of LCO 3.0.4 are not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity of the primary coolant > 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the acceptable operation region of Figure 3.3.15-1.	A.1 Sample and perform isotopic analysis for iodine including I-131, I-133, and I-135.	Once per 4 hours
	AND	
	A.2 Restore specific activity to within limit.	48 hours
B. Required Action A.2 NOT met within the required Completion Time.	B.1 Sample and perform isotopic analysis for iodine, including I-131, I-133, and I-135.	Once per 4 hours
OR	AND	
Specific activity in unacceptable operation region of Figure 3.3.15-1.	B.2 Be in MODE 3 with T_{avg} < 500°F.	6 hours

(continued)

ACTIONS (continued)

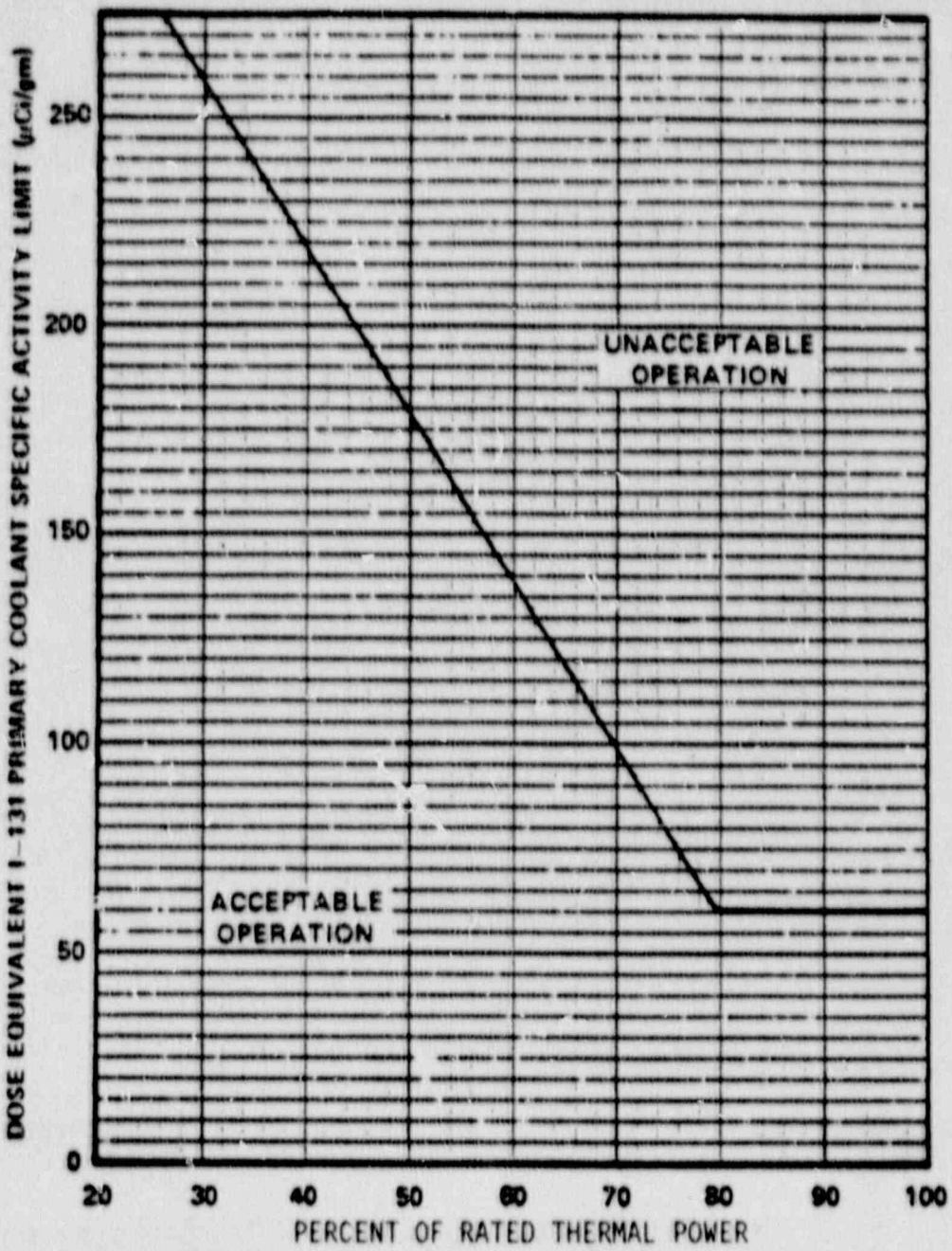
CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Specific activity of the primary coolant > 100/$\bar{E}$ microcuries/gram.	C.1 Sample and perform isotopic analysis for iodine, including I-131, I-133, and I-135.	Once per 4 hours
	AND	
	C.2 Be in MODE 3 with T_{avg} < 500°F.	6 hours
D. Specific activity of the primary coolant > 1.0 microcurie/gram DOSE EQUIVALENT I-131 and > 100/$\bar{E}$ microcuries/gram.	D.1 Sample and perform isotopic analysis for iodine, including I-131, I-133 and I-135.	Once per 4 hours
	AND	
	D.2 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.15.1	Verify specific activity of the primary coolant ≤ 1.0 microcuries/gram DOSE EQUIVALENT I-131.	72 hours
SR 3.3.15.2	----------NOTE---------- Provisions of SR 3.0.4 are not applicable. ---------- Verify specific activity of the primary coolant ≤ 100/$\bar{E}$ microcuries/gram.	-----NOTE------ Sample to be taken after a minimum of 2 EFPD and 20 days of power operation have elapsed since the reactor was last subcritical for 48 hours or longer. ---------- 184 days

Figure 3.3.15-1 (Page 1 of 1)

DOSE EQUIVALENT I-131 Primary Coolant Activity Versus Percent of RATED THERMAL POWER With Primary Coolant Specific Activity > 1.0 microcurie/gram DOSE EQUIVALENT I-131.



3.4 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.4.2 ECCS Trains - MODES 1, 2 and 3

LCO 3.4.2 Two ECCS trains shall be OPERABLE, with each train comprised of:

a. One OPERABLE high pressure injection (HPI) train,

b. One OPERABLE low pressure injection (LPI) train, and

c. One OPERABLE decay heat (DH) cooler.

----------NOTE----------
Two HPI pumps may be deactivated in accordance with LCO 3.3.8, Low Temperature Overpressurization Protection Features.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ECCS train inoperable.	A.1 Restore train to OPERABLE status.	72 hours
B. Required Action NOT met within required Completion Time.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	6 hours 12 hours

3.4 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.4.3 ECCS Trains - MODE 4

LCO 3.4.3 One ECCS train shall be OPERABLE with:

a. An OPERABLE high pressure injection (HPI) train,

b. An OPERABLE low pressure injection (LPI) train, and

c. An OPERABLE decay heat (DH) cooler.

----------NOTE----------

1. High pressure injection isolation valves may be closed with their power supply breakers locked in the open position.

2. Two HPI pumps may be deactivated in accordance with LCO 3.3.8, Low Temperature Overpressurization Protection Feature.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required ECCS train inoperable.	A.1 Initiate action to restore ECCS train to OPERABLE status.	15 minutes
	AND	
	A.2 Maintain MODE 4 by available means of heat removal.	Until restoration of required ECCS train to OPERABLE status.
	AND	
	A.3 Restore inoperable HPI train to OPERABLE status.	1 hour

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

abnormal transient operation, rather than a safety limit, the value for pressurizer level is nominal and is not adjusted for instrument error.

Evaluations performed for large break loss of coolant accident (LOCA), which assumed a higher maximum level than assumed for the LOFW event, have been made. The higher pressurizer level assumed for the LOCA is the bases for the volume of water or coolant released to the containment. The containment analysis performed using the mass and energy release demonstrated that the maximum resulting containment pressure was within design limits.

The maximum pressurizer water level limit satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 1) because it prevents exceeding the initial reactor coolant mass which is an input assumption of the ECCS analysis. The maximum water level also permits the pressurizer code safety valves to relieve steam for anticipated pressure increase transients, preserving their function for mitigation. Thus Selection Criteria 3 is also indirectly applicable.

LCOs

The purpose of the LCO is to ensure pressurizer OPERABILITY for pressure control for normal power operation and for anticipated design basis events as previously described. Compliance with the LCO also ensures that the analysis for LOCA will be met.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on reactor coolant system temperature resulting in the greatest effect on pressurizer level and RCS pressure control. Thus applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3 with RCS temperature greater than 283°F. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as RC pump startup. The temperature of 283°F has been designated as the cutoff for applicability because LCO 3.3.8, Low Temperature Overpressurization Protection Features, provides a requirement for pressurizer level at 283°F. With RCS temperature $\leq$ 283°F, LCO 3.3.8 Low Temperature Overpressurization Protection Features requires further restrictions on maximum pressurizer water level. LCO 3.3.8 also contains appropriate actions to be taken in the event water level cannot be maintained less than the limit.

(continued)

BASES (continued)

ACTIONS

A.1

With water level in excess of the maximum limit, action must be taken to restore pressurizer operation to within the bounds assumed in the analysis. This is done by restoring the pressurizer water level to within the limit. The one hour Completion Time is based on engineering judgement. It is considered to be a reasonable time for draining excess liquid.

B.1 and B.2

If the water level cannot be restored, reducing core power constrains heat input effects that drive pressurizer insurges that could result from an anticipated transient. By reducing power and reactor coolant temperature to at least MODE 3, the thermal energy of the reactor coolant mass is reduced which provides compensation for LOCA mass and energy releases.

The 6 hours allotted to reach MODE 3 is a reasonable time based on operating experience to reach MODE 3 from full power without challenging safety systems and operators. Further pressure and temperature reduction to an RCS temperature ≤ 283°F places the plant into a condition where the LCO is not applicable. The 12 hour time to reach the non-applicable MODE is reasonable based on operating experience.

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

This surveillance requires pressurizer water level to be verified within the maximum limits on a periodic basis. The surveillance is performed by observing indicated level. The 12 hour Frequency is based on engineering judgement and industry-accepted practice.

REFERENCES

1. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

B 3.3 REACTOR COOLANT SYSTEM (RCS)

B 3.3.8 Low Temperature Overpressurization Protection Features

BASES

BACKGROUND

The purpose of the low temperature overpressure protection (LTOP) features LCO is to limit reactor coolant pressure at low temperatures to levels which will not compromise reactor pressure boundary integrity. The reactor vessel is the limiting component for demonstrating that protection is provided. The reactor vessel material is less tough at reduced temperatures than at normal operating temperature. As reactor vessel neutron irradiation accumulates, the vessel material becomes less resistant to stress at low temperatures. Stresses are therefore maintained low and increased as temperature increases during plant heatup. Requirements for all plants were developed by the NRC (Ref. 1) which suggested pressure control measures and required that single equipment failures or single operator errors should not cause the limits of 10 CFR 50, Appendix G (Ref. 2) to be exceeded. Subsequent to the Appendix G requirement the NRC issued Generic Letter 88-11 (Ref. 3) which allows relaxation of the LTOP limits. System evaluations and stress analysis supporting relaxed LTOP limits (Ref. 4) have been developed for Crystal River Unit 3. Although the LTOP limits have been relaxed, plant operational maneuvering must be controlled so the 10CFR50 Appendix G heatup and cooldown PT limits are not violated.

Overpressure protection given by the LCO is provided by reducing the PORV setpoint, and ensuring that pressurizer level is maintained below a maximum limit. The level provides a vapor space (steam or nitrogen), which can accommodate insurges without rapidly increasing pressure. The level limit this provides for a time period to allow the operator to stop the cause of the increase. The PORV, with its reduced setpoint, is the overpressure protection device which provides backup to the operator in terminating increasing pressure events. The approach used to protect the vessel also requires deactivating HPI and CFTs because the PORV and pressurizer level are not fully capable of preventing overpressurization were these systems to be inadvertently actuated. With HPI deactivated the ability to provide core cooling is restricted. To balance the possible need for core cooling the LCO does not require the makeup system to be deactivated. At the lower pressures associated with LTOP and the expected decay heat levels, the makeup system can provide adequate flow via the makeup control valve, or it can be used in the interim until HPI can be reactivated.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSIS

Although analyses of LTOP events are not described in the FSAR, analyses have been performed in response to NRC requests to demonstrate that the reactor vessel is adequately protected against overpressurization during shutdown. Transients potentially capable of overpressurizing the reactor coolant system have been identified and evaluated. Postulated transients included inadvertent high pressure injection actuation; opening core flooding system discharge valves; energizing the pressurizer heaters; failing the makeup control valve open; temporary loss of decay heat removal; reactor coolant thermal expansion caused by RC pump start causing heat transfer from hot steam generators; and addition of nitrogen to the pressurizer.

Two transients which could result in exceeding LTOP limits in less than 10 minutes are inadvertent HPI actuation or inadvertent discharge of two core flood tanks. The analyses also show that the PORV cannot maintain RCS pressure below the LTOP limit if more than one HPI pump is started. Consequently the LCO requires that HPI be defeated at low temperatures and pressures. The CFTs are also isolated for similar reasons, however, the analyses show that the effects of a discharge of two CFTs are important over a narrower range (175°F and below) than the range of the LCO (283°F and below). For other events, operator action is assumed after 10 minutes to preclude overpressurization. Evaluations show that time for operator actions is adequate or the events are self limiting (i.e., will not exceed the LTOP limit).

Analyses for operator response time show that the pressurizer must be maintained at or below 220 inches to allow a 10 minute action time for correcting transients. The fracture mechanics analysis show that the vessel is adequately protected when reactor coolant pressure is maintained at or below 555 psig. Consequently the PORV overpressure protection setpoint has been fixed at 555 psig.

The applicability temperature of 283°F has been established by fracture mechanics analyses. Above this temperature reactor vessel pressure protection is provided by the pressurizer code safety valves. The pressure (555 psig) and temperature (283°F) have been determined for the vessel materials with irradiation accumulation equivalent to 21 effective full power years (EFPY) of operation.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The LTOP features are used to prevent pressure increase transients from exceeding allowable limits. Although a low temperature overpressure transient is not a design basis accident, Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 5) applies because prevention of transient overpressure events leading to non-ductile failure assures pressure boundary integrity.

LCOs

The LCO requires the pressurizer level to be maintained at or below 220 inches to provide time for operator action to prevent transients from exceeding the overpressure protection limit of 555 psig. The PORV is to be OPERABLE with a setpoint at the overpressure limit, and the block valve should be open to ensure a clear flow path.

Since inadvertent actuation of HPI or CFTs cannot be protected by the PORV in combination with pressurizer level, these systems must be deactivated. For the HPI system, the preferred method of deactivation to be used is to close the HPI injection valves and fix their power supply breakers in the open position. Deactivation of other components is also permitted by the LCO, but if powered components are used, power must be removed because this ensures positive prevention of inadvertent actuation. The LCO also permits HPI surveillance testing for components since pump or valve testing can proceed by alternating the system deactivation from pumps to valves. Testing must not permit HPI injection flow to enter the RCS.

The provisions of LCO 3.0.3 are not applicable. If the LCO is violated, changing to a lower MODE may be impractical and may not provide improved LTOP protection. If an LTOP feature is inoperable, the correct approach is to restore the LTOP feature or take other specific ACTIONS.

APPLICABILITY

The LCO is applicable whenever the RCS is at low temperature and can be subjected to pressure increases. Thus the LCO does not apply when the reactor vessel head is completely detensioned or removed. The value for the temperature is determined from fracture mechanics analyses. The LCO is not applicable for operating conditions above the 283°F temperature because the pressurizer code safety valves are able to provide overpressure protection.

(continued)

BASES

APPLICABILITY (continued)

CFT isolation is only required when the CFT pressure is greater than the allowable pressure for the existing RC temperature (i.e., heatup and cooldown pressure temperature limits per 10CFR50 Appendix G and shown in the PRESSURE/TEMPERATURE LIMITS REPORT). This note permits the CFT discharge check valve surveillance to be performed under these conditions. LTOP limits cannot be violated because the heatup and cooldown PT limits are more restrictive.

ACTIONS

A.1, B.1 and B.2

With the pressurizer level greater than 220 in. the time for operator action is reduced, and the postulated transient event which is most affected is a failure of the makeup control valve which permits relatively rapid filling of the pressurizer. Pressurizer level restoration is required in 1 hour. If that cannot be accomplished, the makeup control valve and its associated isolation valve must be closed and maintained closed within an additional 12 hours. This Required Action limits makeup, which is not required with a high pressurizer level and permits cooldown and depressurization to continue. Heatup should be curtailed because heat addition may cause reactor coolant density decrease and increasing pressurizer level. The Completion Times are based on engineering judgement and operating experience that these activities can be accomplished in these time periods, and on engineering evaluations (Ref. 4) that indicate an event requiring LTOP protection is not likely in the time allowed for the Required Actions.

C.1, D.1, and D.2

With the PORV inoperable, overpressure relieving capability is lost and restoration of the PORV in 1 hour is required. If that cannot be accomplished the ability of the makeup system to add water should be restricted within the next 12 hours. By reducing the makeup tank level to 70 inches and by deactivating the low low level interlock to the BWST, insufficient water volume is available to cause the LTOP limit to be exceeded by an inadvertent makeup control valve opening. The makeup system is not deactivated because it may be needed to continue to manage the RCS inventory. The Completion Times are based on engineering judgement and operating experience that these activities can be accomplished in these time periods, and on engineering evaluations (Ref. 4) that indicate an event requiring LTOP protection is not likely in the time allowed for the Required Actions.

(continued)

BASES

ACTIONS

C.1, D.1, and D.2 (Cont'd)

Some PORV testing or maintenance can only be performed at plant shutdown. These activities are permitted provided measures are taken to compensate for the PORV unavailability. With HPI and CFT deactivated per the LCO, the limiting transient which could cause LTOP limits to be exceeded is excessive makeup. Required Action D.1 requires restricting the volume of makeup available from the makeup tank or BWST to be less than that which could cause the RCS pressure to exceed the LTOP limits (due to pressurizer insurge and compression of the vapor space).

E.1 and F.1

With one or both HPI trains active, both actions require deactivation. Required Action E.1 allows deactivation using a variety of methods as may be needed to fit various operating configurations of the combined makeup-HPI system design. If powered components are used to accomplish deactivation, power should be removed to assure positive lockout so that inadvertent ES actuation cannot cause HPI. If Required Action E.1 cannot be accomplished in the Completion Time of 1 hour, Required Action F.1 specifically requires closing and removing power from the HPI injection valves within the next 12 hours. Failure to deactivate HPI per the LCO is not expected, however, since inadvertent actuation is the event of greatest significance (causes the greatest pressure increase in the shortest time), emphasis is placed on deactivation by these Required Action statements. The Completion Times are based on engineering judgement and operating experience that these activities can be accomplished in these time periods, and on engineering evaluations (Ref. 4) that indicate an event requiring LTOP protection is not likely in the time allowed for the Required Actions.

G.1 , H.1 and H.2

With the CFTs unisolated, Required Action G.1 requires isolation within 1 hour. If isolation cannot be accomplished Required Action H.1 provides two options, either of which should be accomplished in 12 hours. The CFT pressure of 600 psi cannot cause LTOP limits to be exceeded if the RCS temperature is greater than 175°F with a full discharge of both tanks. Depressurizing the CFTs below the LTOP pressure limit of 555 psig also prevents exceeding the LTOP limits.

(continued)

BASES

ACTIONS

G.1 , H.1 and H.2 (Cont'd)

The Completion Times are based on engineering judgement and operating experience that these activities can be accomplished in these time periods, and on engineering evaluations (Ref. 4) that indicate an event requiring LTOP protection is not likely in the time allowed for the Required Actions.

I.1, I.2.1, I.2.2, and I.2.3

With the LTOP features inoperable for reasons other than those cited in the above conditions, the system must be restored to an acceptable condition within 1 hour or in the next 12 hours the RCS must be depressurized and a vent size equivalent to an orifice of 0.75 in^2 must be opened. These actions are principally directed toward conditions where more than one of the LCO conditions is violated. Of the multiple conditions the combination of most interest for these Required Actions are A (pressurizer high level) and C (PORV inoperable).

The LCO is not applicable when the RCS is adequately vented. Single or multiple vents may be used. A vent size equivalent to an orifice area of 0.75 in_2 has been specified. The vent size has been calculated assuming 100 psi back pressure. It is considered less likely that HPI or the CFTs cannot be deactivated. Because makeup may be required the vent size stipulated has been developed to accommodate inadvertent full makeup system operation. A 0.75 in_2 vent area is capable of relieving the full flow of one makeup pump with a wide open control valve and preventing the LTOP pressure limit from being exceeded. The PORV, which has a larger area, may be used for venting by opening and locking it open. Removing the PORV for maintenance or testing also accomplishes venting. The vent must be accompanied by deactivating HPI and CFTs since neither the PORV nor the 0.75 in^2 orifice equivalent is capable of maintaining pressure below LTOP limits if these systems are inadvertently actuated.

The Completion Times are based on engineering judgement and operating experience that these activities can be accomplished in these time periods, and on engineering evaluations (Ref. 4) that indicate an event requiring LTOP protection is not likely in the time required for the Required Actions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

Verification of pressurizer level by observing control room indications (or equivalent) assures that a steam or nitrogen bubble of sufficient size is available to reduce the rate of pressure increase from potential transients. The 30 minute surveillance frequency during heatup and cooldown is to be performed for the LCO applicability period when temperature changes can cause pressurizer level variations and may be discontinued when definitions given in plant procedures for defining the end of these conditions are satisfied. Thereafter, surveillance is required at 12 hour intervals. The surveillance frequencies are based on engineering judgement and industry-accepted practice.

SR 3.3.8.2

Verification that the block valve is open ensures an open flow path to the PORV. The 12 hour Frequency is based on engineering judgement and industry-accepted practice.

SR 3.3.8.3 and SR 3.3.8.4

Verification that HPI is deactivated and the CFTs are isolated ensures that inadvertent injection or discharge will not cause a violation of the LTOP pressure limit. The 12 hour Frequency is based on engineering judgement and industry accepted practice.

SR 3.3.8.5

The RCS vent is to be verified open for relief protection when required per Required Action 1.2. The verification frequency of 12 hours is based on engineering judgement and industry accepted practice.

SR 3.3.8.6

Performance of a CHANNEL FUNCTIONAL TEST is required to ensure the PORV setpoints are proper prior to use of the PORV for LTOP. The Frequency permits testing at any time within 31 days of use and allows testing during cooldown prior to entry into LTOP applicability. The Frequency is based on engineering judgement and industry accepted practice.

(continued)

BASES

SURVEILLANCE REQUIRMENTS (continued)

SR 3.3.8.7

Surveillance Requirement 3.3.8.6 is the performance of a CHANNEL CALIBRATION every 18 months. The CHANNEL CALIBRATION for the LTOP setpoint ensures that the PORV will be actuated at the appropriate RCS pressure by verifying the accuracy of the instrument string. The calibration can only be performed during a shutdown. The Frequency is based on engineering judgement and industry-accepted practice.

REFERENCES

1. NRC letter dated October, 1976, J. F. Stolz to J. T. Rogers of Florida Power Corporation, "Transmittal of Analyses of Low Temperature Overpressurization Transients for Crystal River Unit 3."

2. 10 CFR 50, Appendix G, "Fracture Toughness Requirements."

3. Generic letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operation."

4. B&W Summary Report 51-1176431-01, "Crystal River 3, Reactor Vessel Low Temperature Overpressure Protection (LTOP)," September 1989.

5. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

B 3.3 REACTOR COOLANT SYSTEM (RCS)

B 3.3.9 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer code safety valves is to provide reactor coolant system (RCS) overpressure protection. Operating in conjunction with the reactor protection system two valves are used to assure that the Safety Limit of 2750 psig is not exceeded for analyzed transients during operation in MODES 1, 2, and 3. One code safety valve is adequate in MODES 4 and 5. Overpressure protection is also provided by operating procedures and low temperature overpressurization protection (LTOP) system equipment in MODES 4 and 5. LTOP is provided by reducing the PORV setpoint and isolating certain functions. For more detail see Bases B 3.3.8 (Low Temperature Overpressurization Protection Features). The code safety valves discharge steam from the pressurizer to a drain tank located in the containment. Should the drain tank or connecting piping be unable to accept the code valve discharge, valve operation would result in relief to the containment atmosphere. While discharge to the containment atmosphere is highly undesirable, reactor coolant pressure boundary integrity would remain protected.

APPLICABLE SAFETY ANALYSIS

All accident analyses in the FSAR which require safety valve actuation assume operation of both pressurizer code safety valves to limit increasing reactor coolant pressure. The overpressure protection analysis (Ref. 1) is also based on operation of both code safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 1%). These valves must accommodate pressurizer insurges which could occur during the startup, rod withdrawal, ejected rod, loss of main feedwater, and main feedwater line break accidents. The startup accident establishes the minimum code safety valve capacity. The startup accident is assumed to occur at less than 15% power in MODE 1 and could occur in the lower bound of MODE 2 when the control rod drive trip breakers are closed in the transition from MODE 3 to MODE 2. Single failure of a code safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME code. Compliance with this specification is required to assure that the accident analysis and design basis calculations remain valid.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The pressurizer code safety valves satisfy the requirements of Selection Criterion 3 of the Interim Policy Statement (Ref. 3) because operation of two valves within their allowed limit setting ensures that they will function to provide RCS overpressure protection for analyzed transients of the design basis. Failure to function could challenge the integrity of a fission product barrier.

LCOs

The two pressurizer code safety valves are set to open at the RCS design pressure (2500 psig) and within the ASME specified tolerance to avoid exceeding the maximum RCS design pressure Safety Limit, to maintain accident analysis assumptions, and to comply with ASME requirements. The upper and lower pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 2) for lifting pressures above 1000 psig. The limit protected by this specification is the reactor coolant pressure boundary Safety Limit of 110% of design pressure. Inoperability of one or both valves could result in exceeding the Safety Limit were a transient to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

OPERABILITY of the two valves is required in MODES 1 , 2, and 3 because their combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 is included although the listed accidents would not require both code safety valves to function for protection in MODE 3. It is conservatively included because the accident could occur near the lower bound of MODE 2. In MODES 4 and 5 one code safety valve is required to accommodate the pressurizer insurges which could result from all the sources identified (i.e. residual heat, pump energy, pressurizer heaters) and thus the LCO is not applicable in MODES 4 and 5. Overpressure protection is not required in MODE 6 with the reactor vessel head removed.

ACTIONS

A.1

With one pressurizer code safety valve inoperable, restoration must take place in 15 minutes. The 15 minutes to restore a safety valve to OPERABILITY is based on engineering judgement.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the Required Action cannot be met within the required Completion Time, the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in at least MODE 3 in six hours and in MODE 4 in 12 hours. The six hours allotted to reach MODE 3 is a reasonable time based on operating experience to reach MODE 3 from full power without challenging safety systems and operators. Similarly, the 12 hours allotted is a reasonable time to reach MODE 4 considering that a plant can easily cooldown to this MODE in this time frame. The change from MODES 1, 2 or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges and thereby removes the attendant need for overpressure protection by two pressurizer code safety valves.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Section XI of the ASME code (SR 3.0.5) requires testing of each pressurizer code safety valve at least once every 5 years. This surveillance includes setpoint testing and demonstration of OPERABILITY through non-popping (hydraulic assist) techniques. Because testing is done at reduced pressure, the test lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. The Frequency is based on engineering judgement and industry-accepted practice.

REFERENCES

1. B&W Topical Report BAW-10043, "Overpressure Protection for Babcock & Wilcox Pressurized Water Reactors, J. D. Carlton, May 1972.

2. ASME Boiler & Pressure Vessel Code, Section III, "Nuclear Vessels," and Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components.

3. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

B 3.3 REACTOR COOLANT SYSTEM (RCS)

B 3.3.10 Relief Valves

BASES

BACKGROUND

The pressurizer is equipped with three devices for pressure relief functions: two ASME code safety valves which are safety grade components and one power operated relief valve (PORV) which is not a safety grade device. The PORV is an electromagnetic pilot operated valve that is automatically opened at a specific set pressure when the pressurizer pressure increases and is automatically closed on decreasing pressure. The PORV may also be manually operated using controls installed in the control room.

An electric motor operated, normally open, block valve is installed between the pressurizer and the PORV. The function of the block valve is to isolate the PORV. Block valve closure is accomplished manually using controls in the control room and may be used to isolate a leaking PORV to permit continued power operation. Most importantly, the block valve is used to isolate a stuck open PORV small break loss of coolant accident (SBLOCA) to terminate the reactor coolant system (RCS) depressurization and coolant inventory loss.

The PORV, its block valve, and their controls are powered from normal power supplies but are also capable of being powered from emergency supplies. Power supplies for the PORV are separate from those for the block valve. Power supply requirements are defined in NUREG 0737, Paragraph III G. 1 (Ref. 1).

The PORV setpoint is set greater than the high pressure reactor trip setpoint and less than the opening setpoint for the pressurizer code safety valves. This setpoint was required by the NRC in IE Bulletin 79-05B (Ref. 2). The purpose of the relationship of these setpoints is to limit the number of transient pressure increase challenges which might open the PORV, which, if opened, could fail in the open position. The PORV setpoint is greater than the high pressure reactor trip setpoint. Consequently a pressure

(continued)

BASES

BACKGROUND (continued)

increase transient would cause a reactor trip, reducing core energy, and for many expected transients, prevent the pressure increase from reaching the PORV setpoint. The PORV setpoint thus limits the frequency of challenges from transients and limits the possibility of a SBLOCA from a failed open PORV. Placing the setpoint below the pressurizer safety valve opening setpoint reduces the frequency of challenges to the safety valves, which unlike the PORV cannot be isolated if they were to fail open. Accurate control of the PORV setpoint is therefore important for limiting the possibility of SBLOCA.

The primary purpose of this LCO is to ensure that the PORV, its setpoint, and the block valve are operating correctly so the potential for SBLOCA through the PORV pathway is minimized, or if a SBLOCA were to occur through a failed open PORV the block valve could be manually operated to isolate the path.

The PORV may also be manually operated to depressurize the RCS as deemed necessary by the operator in response to normal or abnormal transients. The PORV may be used for depressurization when the pressurizer spray is not available; a condition that would be encountered during loss of offsite power. Steam generator tube rupture is one event that may require use of the PORV if the sprays are unavailable.

The PORV may also be used for feed and bleed core cooling for multiple equipment failure events that are not within the design basis, such as total loss of feedwater.

The PORV functions as an automatic overpressure device and limits challenges to the code safety valves. Although the PORV acts as an overpressure protection device, safety analyses do not take credit for PORV actuation, but do take credit for the safety valves. The overpressure protection function of the PORV during MODES 1 and 2 is an operational function and is not addressed by Technical Specifications.

The PORV also provides low temperature overpressure protection (LTOP) during heatup and cooldown. LCO 3.3.8 addresses this function.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSIS

There are no explicit FSAR safety analyses of a SBLOCA through the PORV path. The PORV SBLOCA is not a design basis event, however the break size is bounded by the spectrum of piping breaks analyzed for plant licensing. Because the PORV SBLOCA is located at the top of the pressurizer the RCS response characteristics are different from RCS loop piping breaks; analyses have been performed to investigate these characteristics.

The possibility of a SBLOCA through the PORV is reduced when the PORV flow path is OPERABLE and the PORV opening setpoint is verified to be reasonably remote from expected transient challenges. The possibility is minimized if the flow path is isolated.

The PORV opening setpoint has been established in accordance with IE Bulletin 79-05B (Ref. 2). No specific safety analyses were performed to determine the setpoint, however, it has been set so expected RCS pressure increases from anticipated transients will not challenge the PORV, minimizing the possibility of SBLOCA through the PORV.

Overpressure protection analyses do not take credit for the PORV opening and therefore are not pertinent to the PORV.

The design basis accidents reported in the FSAR safety analyses do not take credit for the PORV for mitigation. However operational analyses that support the emergency operating procedures utilize the PORV to depressurize the RCS for mitigation of steam generator tube rupture (SGTR) when the pressurizer spray system is unavailable (loss of offsite power). FSAR safety analyses for SGTR have been performed assuming that offsite power is available and thus sprays are available.

The PORV and its block valve do not satisfy the requirements of the Selection Criterion of the NRC Interim Policy Statement (Ref. 3). This specification was evaluated using insights gained from reviewing representative probabilistic risk assessments. The PORV and its block valve are deemed important to risk.

(continued)

BASES (continued)

LCOs

The LCO requires the PORV and its block valve to be OPERABLE. By ensuring that the PORV opening setpoint is correct the PORV is not subject to frequent challenges from possible pressure increase transients and therefore the possibility of a SBLOCA through a failed open PORV is not an event of an undesirable frequency. The block valve is required to be OPERABLE so it may be used to isolate the flow path if the PORV is not OPERABLE. If the block valve is not OPERABLE, the PORV may be used for isolation.

APPLICABILITY

The PORV will automatically open when the RCS pressure increases to the PORV setpoint. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. Pressure increase transients can occur any time the steam generators are used for heat removal. The most rapid increases will occur at higher operating power and pressure conditions of MODES 1 and 2.

Pressure increases are less prominent in MODE 3, because the core input energy is reduced, but the RCS pressure is high. Therefore the applicability is pertinent to MODES 1, 2, and 3. When both pressure and core energy are decreased the pressure surges become much less significant and the LCO is not applicable in MODE 4, partly because the consequences are less severe and partly because the time spent in heatup and cooldown is short. The PORV setpoint is reduced for low temperature overpressurization protection (LTOP) at lower pressures during heatup and cooldown. LTOP is applicable during MODES 4, 5, and 6 with the reactor vessel head in place. As such, LCO 3.3.8 (LTOP Features) addresses the PORV requirements in these MODES.

Anticipated pressure increase transients caused by secondary system upsets which are pertinent to MODES 1, 2, or 3 include:

- Loss of electrical load.
- Turbine trip.
- Loss of main feedwater.
- Loss of condenser vacuum.
- Inadvertent closure of main steam isolation valve(s).

(continued)

BASES (continued)

ACTIONS

In general, the Required Actions for each of the Conditions (PORV inoperable, block valve inoperable, or both inoperable) utilize the same concept:

- Restore OPERABILITY, or (if that is not possible)

- Isolate the flow path (isolation ensures that a transient challenge will not cause the PORV to fail open resulting in a SBLOCA), or (if restoration and isolation are not possible)

- Reduce core power and RCS pressure (by reducing the energy level the pressure increase of a secondary side transient are less likely to challenge the degraded components in the flow path).

The Required Actions permit continued operation with either or both valves inoperable as long as the flow path is isolated.

A.1 and A.2

With the PORV inoperable, either the PORV must be restored or the flow path isolated within one hour. In this instance, as compared to Required Actions for other Conditions, the block valve should be closed but power need not be removed from the block valve. This Required Action is because the block valve is OPERABLE. The Completion Times are based on engineering judgement and plant operating experience.

B.1, B.2.1, B.2.2, B.3.1, and B.3.2

If the block valve is inoperable or the PORV and the block valve are inoperable, the inoperable components must be restored, or the flow path isolated and power supply removed. The Completion Times are based on engineering judgement and plant operating experience.

C.1 and C.2

If the Required Action cannot be met within the required Completion Time, the plant must be placed in a MODE in which the requirement does not apply. This is done by placing the plant in at least MODE 3 in six hours and in MODE 4 in 12 hours. The six hours allotted to reach MODE 3 is a reasonable time based on operating experience to reach MODE 3 from full power without challenging safety systems and operators.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2 (Cont'd)

Similarly, the 12 hours allotted is a reasonable time to reach MODE 4 considering that a plant can easily cooldown to this MODE in such a time frame. In MODE 4, RCS energy (core power and pressure) is reduced to a minimum and decay heat removal can now be provided by the decay heat removal system, which eliminates the possibility for secondary plant upsets.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

Block valve cycling verifies that it can be closed if needed. The basis for the frequency is ASME XI (Ref. 4). Block valve cycling is not performed when it is closed for isolation; cycling could increase the hazard of an existing degraded flow path.

SR 3.3.10.2

Surveillance Requirement 3.3.10.2 is the performance of a CHANNEL CALIBRATION every 18 months. The CHANNEL CALIBRATION ensures the PORV setpoint is appropriately established above the RCS high pressure trip setpoint and thus remote from transient pressure challenges. The calibration also ensures that the PORV will open below the pressurizer code safety valve setpoint, thus limiting challenges to the code safety valves. The calibration can only be performed during shutdown. The Frequency is based on engineering judgement and industry-accepted practice.

REFERENCES

1. NUREG 0737 "Clarification of TMI Action Plan Requirements," November, 1980.

2. NRC IE Bulletin 79-05B, 4/21/79.

3. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

4. ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

B 3.3 REACTOR COOLANT SYSTEM (RCS)

B 3.3.11 RCS Leakage

BASES

BACKGROUND

The RCS is comprised of components whose joints are made by welding, bolting, rolling and pressure loading. The RCS is isolated from other plant systems by valves. During plant life, these interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS leakage LCO is to permit system operation in the presence of leakage from these sources in amounts which do not compromise safety. The LCO defines the types of leakage and allowable limits for leakage. This LCO is required to protect the reactor coolant pressure boundary against degradation, which ensures the RCS integrity for maintaining core cooling. Leakage monitoring is an indicator of RCS integrity and can be performed frequently during operation. Leakage monitoring is complementary to inservice inspections which are performed periodically at outages.

Other related LCOs give limits for leakage at specific locations. LCO 3.3.12 (RCS Primary to Secondary Leakage), specifies limits for steam generator tube leakage, and LCO 3.3.13 (RCS Pressure Isolation Valve (PIV) Leakage), specifies valve seat leakage limits for certain valves that isolate the high pressure RCS from other low pressure systems. LCO 3.3.14 (Leakage Detection Instrumentation), specifies the requirements for the monitoring equipment used to detect leakage into the containment.

APPLICABLE SAFETY ANALYSIS

Except for primary to secondary leakage (LCO 3.3.12) safety analyses for design bases accidents do not address leakage. Some design basis accidents, particularly those with an emphasis for offsite dose calculations such as steam generator tube rupture, assume a pre-existing 1 gpm primary to secondary leak and consequent activity in one generator as an initial analysis condition.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

Leakage is an indication of possible degradation of the RCS boundary. Thus Selection Criterion 3 of the NRC Interim Policy Statement (Ref. 1) is satisfied.

LCOs

a. PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE, defined as leakage through a non-isolable fault in a RCS component body, pipe, or vessel wall (excluding reactor coolant pump (RCP) shaft seals, packing, and steam generator tube leakage), is not allowed in any amount because it would be indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the reactor coolant pressure boundary.

b. UNIDENTIFIED LEAKAGE

UNIDENTIFIED LEAKAGE is defined as reactor coolant leakage which is not identified. Up to 1 gpm of UNIDENTIFIED LEAKAGE is considered allowable on the basis that it is a reasonable minimum detectable level for the containment air monitoring and containment sump level monitoring equipment.

c. IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE is defined as leakage into closed systems connected to the RCS that is captured and recovered. Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because it provides for leakage from known sources which do not interfere with normal operation and which is well within the capability of the make-up system to replenish. This leakage includes leakage to the containment from sources that are specifically known and located, but does not include pipe or vessel wall leakage or RCP controlled bleed off (which is a normal process function and not considered leakage). Violation of this LCO could result in impairment of reactor coolant inventory control.

Leakage past the seats of pressure isolation valves (LCO 3.3.13) and primary to secondary leakage (LCO 3.3.12) are included in IDENTIFIED LEAKAGE.

(continued)

BASES (continued)

APPLICABILITY

The RCS leakage LCO applies to MODES 1, 2, 3, and 4 to ensure that the applicable accident analysis assumptions remain valid and to minimize pressure boundary degradation from continued leakage. Leakage limits are not provided for MODES 5 and 6 because the reactor coolant pressure is far lower, making leakage less likely and less difficult to control; and because the mechanisms for offsite release have been reduced or eliminated. Accordingly, the potential consequences of reactor coolant leakage are far lower during these MODES.

ACTIONS

A.1

The general activities associated with the Condition A are:

- Quantifying or verifying the leakage rate.
- Identifying the source of any leakage.
- Determining what repair might be appropriate, and determining if the repair can be carried out at pressure or whether shutdown is required.

Of the activities associated with the Condition A, one of great importance is determining if PRESSURE BOUNDARY LEAKAGE exists. If the leakage source cannot be identified, the leak is UNIDENTIFIED, and if there is doubt of the location a conservative assumption is that the leakage is from the pressure boundary.

With RCS leakage outside limits for reasons other than PRESSURE BOUNDARY LEAKAGE, the leakage must be reduced. The 4 hour Completion Time may permit repair or isolation depending on the source and the complexity of the repair. The 4 hours is based on engineering judgement, and the time permits a reasonably stable period at operating pressure to identify the source and verify the quantity of leakage by inventory balance. Inventory balance calculations require a stable condition; pressure reduction may lessen the ability for source identification.

B.1 and B.2

If RCS leakage (other than PRESSURE BOUNDARY LEAKAGE) cannot be reduced to the permissible limit within the required Completion Time, reducing power and pressure in sequence to MODE 3 and then MODE 5 are required. By doing so the hazard associated with the leak is reduced.

(continued)

BASES

ACTIONS

B.1 and B.2 (Cont'd)

In this condition, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 in 36 hours. The six hours allotted to reach MODE 3 is a reasonable time based on operating experience to reach MODE 3 from full power without challenging safety systems and operators. Similarly, the 36 hours allotted is a reasonable time to reach MODE 5 considering the plant can easily cooldown in such a time frame on one safety system train. In MODE 5, pressure is reduced to the lowest possible value and temperature to below the boiling point at atmospheric pressure. The Required Action reduces the driving force for the leak and eliminates the possibility for contaminated steam leaks to the containment atmosphere and the possible deleterious cutting action of steam on the material surrounding the leak.

C.1 and C.2

If any reactor coolant PRESSURE BOUNDARY LEAKAGE is detected, the reactor must be placed in HOT STANDBY in 6 hours and MODE 5 in the next 36 hours. This action reduces the leakage, reduces the factors which tend to degrade the pressure boundary, and most importantly reduces the potential for RCS piping or vessel failure. The Bases for the Completion Times are the same as those for Required Actions B.1 and B.2. As such, the Bases for Required Actions B.1 and B.2 are applicable to Required Actions C.1 and C.2.

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1

An inventory balance is the most precise method for quantifying leakage rate. Nominal instrument indications of the parameters used to make the inventory balance calculation are used without adjustments for instrument error. Although other methods exist, and may be used, they give results that are less certain, and are not required to meet this specification. Seventy-two hours permits a reasonable interval for trending. Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and the surveillance is not required unless steady state is established. For purposes of leakage determination by inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank level, constant makeup and letdown and RC

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1 (Cont'd)

pump seal injection and return flows. As such, the Completion Time is based on engineering judgement and industry-accepted practice.

The NOTE provides an exception to 3.0.4 permitting entry into MODES with stable conditions so the surveillance can produce accurate results.

In addition to the required periodic steady state surveillance, an inventory balance is required to be made if the containment sump level leakage monitoring instrumentation is inoperable (Required Action B.1 of LCO 3.3.14).

REFERENCES

1. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

B 3.3 REACTOR COOLANT SYSTEM (RCS)

B 3.3.12 RCS Primary to Secondary Leakage

BASES

BACKGROUND

The purpose of the primary to secondary leakage LCO is to permit system operation in the presence of steam generator tube leakage in amounts which do not compromise safety. Refer to the Bases for LCO 3.3.11 for definitions of the leakage terms used. Leakage into the secondary side of the steam generators has two effects: 1) it indicates a degradation of the RCS boundary (experience indicates that in most cases the source of primary to secondary leakage is a single steam generator tube flaw), and 2) reactor coolant fission product activity is present in the secondary system. Because the hot well is common to all feedwater trains, activity from any primary to secondary leak is mixed and spread throughout the entire secondary system. The value of 1 gpm allowable leakage has been established because it is a practical limit within the capability for detection and quantification. The importance of existing secondary leakage and activity is that a primary barrier to fission product confinement has been breached and a pathway to public release via the condenser air ejector(s) exists. Monitoring for secondary leakage can be performed during operation thus providing an indication of the condition of the barrier; it is complementary to the inservice tube inspection program that is performed during outages. Leakage can be detected in a variety of ways including condenser offgas radiation monitors, secondary water chemistry analysis, or steam line radiation monitors.

The 1 gpm allowable primary to secondary leakage is included in the 10 gpm total allowable IDENTIFIED LEAKAGE rate. It is not in addition to the 10 gpm permissible IDENTIFIED LEAKAGE rate.

APPLICABLE SAFETY ANALYSIS

The limit of 1 gpm primary to secondary leakage is assumed as a pre-existing condition for design basis event safety analyses. The leak is assumed to exist in only one generator. However, because of feedwater train mixing in the hot well, the fission products are spread throughout the secondary

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

system. Of the analyses that are performed with the 1 gpm pre-existing leak, the steam line break (outside of containment) event is the most limiting for site radiation releases.

All of the inventory of the generator with the broken steam line and a portion of the inventory of the unaffected generator is released (rapid isolation terminates releases from the intact generator). Results show site boundary doses to be within acceptable limits.

Steam generator tube rupture (SGTR) is also analyzed with a concurrent 1 gpm leak in the opposite steam generator. The tube rupture flow rate is significantly greater than the 1 gpm rate and the effect is relatively inconsequential. Safety analyses for operating B&W plants have not been required to assume a loss of the condenser for SGTR. However best estimate SGTR releases with no condenser available have been evaluated to support emergency operating procedures. These analyses indicate that doses would be within limits and industry operating experience (such as the Ginna event) confirms that actual SGTR releases are much lower than predicted using conservative safety analysis methods.

Stress analyses have been performed for the once through steam generators (OTSG). Predictions are that the higher stresses caused by harsh transients such as the steam line break will not cause loss of tube integrity even for tubes thinned to the plugging limit.

This specification satisfies Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 1) because the 1 gpm leak rate limit is used as an input assumption for safety analysis.

LCOs

Primary to secondary leakage, assumed at the rate of 1 gpm, produced acceptable offsite doses in the steamline failure accident analysis. Violation of this LCO could void the accident analysis offsite dose calculations for the steamline failure.

The leakage also shows a breach of the pressure boundary exists and indicates that further tube weakening may occur.

(continued)

BASES (continued)

APPLICABILITY

The primary to secondary leakage LCO applies to MODES 1, 2, 3, and 4 to ensure that leakage is detected and operation terminated in the event steam generator tube leakage exceeds the amount assumed in the steam line failure accident analysis and ensures that the accident analysis remains valid. Leakage limits are not provided for MODES 5 and 6 because the reactor coolant pressure is far lower, reducing the primary to secondary pressure differential and thus the leak rate. Because the potential for offsite release is likely to be much lower in these MODES the potential consequences of primary to secondary reactor coolant leakage are far lower.

ACTIONS

A.1

With primary to secondary leakage in excess of 1 gpm the Required Action is to reduce the leak rate below 1 gpm in 4 hours. The Completion Time is based on engineering judgement. Four hours provides a stable period at operating pressure to verify the leak quantity, determine which generator is affected and prepare plans for remedial action. Activities to assess the leak rate should be done while the plant is in stable steady state conditions and cannot be done precisely if the plant is maneuvered.

B.1 and B.2

If the leak cannot be reduced, reducing power and pressure in sequence to MODE 3 and then MODE 5 are required. This reduces the leakage, and reduces or eliminates the mechanisms for offsite release. In this condition, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 in 6 hours and in MODE 5 in 36 hours. The six hours allotted to reach MODE 3 is a reasonable time based on operating experience to reach MODE 3 from full power without challenging safety systems and operators. Similarly, the 36 hours allotted is a reasonable time to reach MODE 5 considering the plant can easily cooldown in such a time frame on one safety system train. The orderly cooldown also permits tube stresses to be held to a minimum, limiting the potential for further tube degradation.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

Detection and location of primary to secondary leakage may be done by several methods. The inventory balance is the most precise method for quantifying leakage and may be done in conjunction with other methods which may also include analyzing the secondary chemistry and activity. Seventy-two hours permits a reasonable interval for trending, and is consistent with the time period for assessing leakage at other locations.

Stable steady state conditions must exist to obtain an accurate indication of the leakage rate using the inventory conditions. Other methods also require steady state. For this surveillance steady state means stable RCS pressure, temperature, steady reactor power with equilibrium xenon, constant boron concentration in the reactor coolant and stable secondary side conditions (temperature, pressure, flow and chemical consistency). Stable and constant makeup, letdown RC pump seal injection and seal return flows and stable makeup tank and pressurizer levels are also required.

The NOTE provides an exception to 3.0.4 permitting entry into MODES with stable conditions so the surveillance can produce accurate results.

REFERENCES

1. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 3, 1987.

B 3.3 REACTOR COOLANT SYSTEM (RCS)

B 3.3.13 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

This specification applies to the four series check valves (two per line) that isolate the RCS from low pressure portions of the decay heat removal (DHR) system outside the containment. A normally closed, high pressure rated, motor operated gate valve is upstream of the two check valves. The selection of valves is based on information presented in Reference 1 which requires testing of two in-series check valves used for isolation of high pressure to low pressure systems when leakage of one valve could go undetected for a substantial length of time.

The purpose of the RCS PIV LCO is to permit system operation in the presence of leakage through these valves in amounts which do not compromise safety. PIV leakage limits (leakage ≤ 5 gpm) apply to leakage rates for individual valves. The total IDENTIFIED LEAKAGE rate of 10 gpm given by LCO 3.3.11 also applies and leakage through these valves are to be included in the total allowable leakage.

Although the specification provides limits in the form of allowable leakage rates, the important purpose of the specification is to prevent overpressure failure of the low pressure portions of the DHR system caused by high RCS pressure. The leakage limits are indications that the boundary (check valves) between the RCS and the DHR system is degraded or becoming degraded. Failure of the check valves could lead to overpressure of the DHR piping or components. Failure consequences could be a LOCA outside of containment, with the possibility of being unable to recirculate from the containment emergency sump after the initial BWST injection is exhausted. A failure of portions of the DHR system can also degrade the ability for low pressure ECCS injection, although analyses indicate that other remaining injection paths would successfully maintain core cooling (when injection water is available).

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSIS	Pressure isolation valve leakage is not considered in any design basis accident analyses. This specification provides for monitoring the condition of the reactor coolant pressure boundary to detect degradation which could lead to accidents. Therefore, Selection Criterion 1 of the NRC Interim Policy Statement (Ref. 2) is satisfied.
LCOs	Maximum isolation valve leakage is usually on the order of drops per minute. The source of the leakage limits is an NRC letter resulting from the Reactor Safety Study (WASH 1400) which identified the inter-system loss of coolant accident as a significant contributor to core melt risk (Ref. 1). Violation of this LCO could result in continued degradation of a pressure isolation valve.
APPLICABILITY	This LCO apply to MODES 1, 2, 3, and 4 to ensure that pressure isolation valve leakage is detected to minimize pressure boundary degradation from continued leakage. Leakage limits are not provided for MODES 5 and 6 because the reactor coolant pressure is far lower, making leakage less likely. Accordingly, the potential for the consequences of reactor coolant leakage are far lower during these MODES.
ACTIONS	A.1 With leakage in excess of the allowable limits 4 hours are provided to reduce leakage. The period permits operation to continue under stable conditions while leakage is assessed and corrective actions are being taken. Leakage assessment requires a stable plant pressure condition. The 4 hour time is based on engineering judgement that actions to reseat and verify the leakage quantity in this period can be reasonably performed. The 4 hour period is also based on a subjective judgement that operation for longer periods increases the hazard for overpressurizing the low pressure portions of the DHR outside containment.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

The reactor must be placed in a MODE in which the requirement does not apply if leakage cannot be reduced. This is done by placing the plant in MODE 3 within 6 hours and MODE 5 within the next 36 hours. This action reduces the leakage and also reduces the factors which tend to further degrade the isolation valves. The six hours allotted to reach MODE 3 is a reasonable time based on operating experience to reach MODE 3 from full power without challenging safety systems and operators. Similarly, the 36 hours allotted is reasonable to reach MODE 5 considering that a plant can easily cooldown in such a time frame on one safety system train.

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1

Performance of leakage testing on each reactor coolant pressure isolation valve is required to verify that leakage is below the specified limits and to detect leaking valves. The 5 gpm limit is to be applied to each valve. Testing is performed at least every refueling or prior to startup after a 72 hour outage if a recent test (within 9 months) does not exist. The 72 hour outage allowance is based on engineering judgement. These Surveillance Requirements were specified by the NRC (Items 1 and 2) in Ref. 1 and are in accordance with ASME XI (Item 3) (Ref. 3).

SR 3.0.4 is exempted to permit leak testing at high valve differential pressures with stable conditions than are possible in the lower MODES.

REFERENCES

1. NRC Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves (all plants) dated 4/20/81. Includes Technical Evaluation Report "Primary Coolant System Pressure Isolation Valves," prepared by the Franklin Research Center.

2. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

3. ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants."

B 3.3 REACTOR COOLANT SYSTEM (RCS)

B 3.3.14 Leakage Detection Instrumentation

BASES

BACKGROUND

General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR 50, "General Design Criteria for Nuclear Power Plants, (Ref. 1)" requires that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

A limited amount of leakage is expected from the reactor coolant system (RCS) and from auxiliary systems within the containment. Some leakage will occur from valve packing, pump seals, vessel/closure head seals and safety and relief valves. If leakage occurs via these paths it is detected, collected to the extent practical, and isolated from the containment atmosphere so as not to mask any potentially serious leak should it occur. These leakages are IDENTIFIED LEAKAGE and may be piped to tanks or sumps so flow rate can be established and monitored during plant operation.

Uncollected leakage to the containment atmosphere from other sources increases the humidity of the containment. The moisture condensed from the atmosphere by air coolers together with any associated liquid leakage to the containment is UNIDENTIFIED LEAKAGE and is collected in tanks or sumps where the flow rate is established and monitored during plant operation. A small amount of UNIDENTIFIED LEAKAGE may be impractical to eliminate, but it should be reduced to a small flow rate, to permit the leakage detection systems to positively and rapidly detect a small increase in flow rate. Thus a small UNIDENTIFIED LEAKAGE rate that is of concern will not be masked by a larger acceptable IDENTIFIED LEAKAGE rate.

Leakage detection systems should detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for a gross pressure boundary failure. Some cracks might develop and penetrate the RCPB wall, exhibit very slow growth, and afford ample time for a safe and orderly plant shutdown.

(continued)

BASES

BACKGROUND (continued)

The leakage detection monitors used are of two different principles: containment sump level and atmospheric activity monitor. The atmospheric activity monitoring instrumentation detects gaseous and iodine radioactivity.

Industry practice has shown that water flow rate changes of from 0.5 to 1.0 gpm can readily be detected in containment sumps by monitoring changes in sump water level, in flow rate, or in the operating frequency of pumps. Sumps and tanks used to collect UNIDENTIFIED LEAKAGE and air cooler condensate are instrumented to alarm for increases of the normal flow rates. This sensitivity provides an acceptable performance for detecting increases in UNIDENTIFIED LEAKAGE.

Reactor coolant activity released to the containment can be detected by radiation monitoring instrumentation. Instrument sensitivities of 10^{-9} micro Ci/cc radioactivity for air particulate monitoring and of 10^{-6} micro Ci/cc radioactivity for gaseous monitoring are practical for these leakage detection systems. Typical ranges are 10-10^{1}6 cpm. Radioactivity monitoring systems are included because of their sensitivity and rapid response to leaks from the RCPB.

In addition to the instrumentation cited by the LCO, other leakage detection means may be used. Humidity changes or pressure and temperature changes may provide indications of leakage.

APPLICABLE SAFETY ANALYSIS

The safety significance of leaks from the RCPB vary widely depending on the source of the leak as well as the leakage rate and duration. Therefore, the detection and monitoring of reactor coolant leakage into the containment area is necessary. Separating the identified sources of leakage from unidentified sources is necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action should a leak occur that is detrimental to the safety of the facility.

RCS leakage detection instrumentation satisfies Selection Criterion 1 of the NRC Interim Policy Statement (Ref. 2). As such, these variables are retained in the RCS Leakage Detection Instrumentation LCO.

(continued)

BASES (continued)

LCO

One method of protection against RCPB leakage failure is the ability of instrumentation to rapidly detect extremely small leaks. This LCO requires that instrumentation of two diverse principles be OPERABLE to provide a high degree of confidence that extremely small leaks are detected in time to allow actions to place the plant in a safe condition when leakage indicates possible RCPB degradation.

The LCO is satisfied when monitors of diverse measurement means are available. Thus the containment sump monitor in combination with the containment atmosphere radioactivity monitor (iodine and gaseous channels) provides an acceptable minimum.

APPLICABILITY

Leakage detection systems are only required to be OPERABLE in MODES 1, 2, 3, and 4. In these MODES the RCS temperature is greater than 200°F and pressure is greater than atmospheric. With the plant in MODES 5 or 6, temperature is less than or equal to 200°F and pressure is low. Below 200°F any leakage would be liquid and atmospheric monitors are less effective. Since the design of the RCPB is able to withstand temperatures and pressures far greater than those allowed in MODES 5 or 6, and the pressure differential across the RCPB is low, leakage is almost impossible. Therefore, the LCO is not applicable in MODES 5 and 6.

ACTIONS

A.1 and A.2

With the containment atmosphere radioactivity monitor (gaseous and iodine activity channels) inoperable, grab samples shall be taken and analyzed to provide alternate periodic information. Provided these samples are obtained and analyzed every 24 hours, the plant may continue operation for up to 30 days. The 24 hour sampling interval is based on engineering judgement and plant operating experience. The 30 day Completion Time for restoration is based on engineering judgement and recognizes that multiple forms of leak detection are available, but extended periods of operation while using alternative monitoring is not prudent since the original design monitoring capability is not being met.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

With the containment sump level monitor inoperable, no form of grab sample could provide the equivalent information. However the atmospheric activity monitors provide indications of changes in leakage. Restoration is required to regain the function of the sump monitor. As an alternate to the sump monitor and in conjunction with atmospheric monitors the periodic surveillance, SR 3.3.11.1, for RCS inventory balance is to be performed at an increased frequency of once per 24 hours. The 24 hours is based on engineering judgement and is compatible with the required Frequency for grab samples. The 30 day Completion Time for restoration recognizes that multiple forms of leakage detection are available. The Completion Time is based on engineering judgement and plant operating experience.

C.1 and C.2

If the Required Action cannot be met within the required Completion Time, the plant must be placed in a MODE in which the LCO does not apply. This is done by placing the plant in at least MODE 3 in six hours and in MODE 5 in 36 hours. The six hours allotted to reach MODE 3 is a reasonable time based on operating experience to reach MODE 3 from full power without challenging safety systems and operators. Similarly, the 36 hours allotted is a reasonable time to reach MODE 5 considering that a plant can easily cooldown in such a time frame on one safety system train.

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

This Surveillance requires the periodic monitoring of containment atmosphere activity. This provides assurance that leakage which would indicate reactor coolant pressure boundary degradation would be detected. The surveillance Frequency (12 hours) is based on engineering judgement and industry-accepted practice.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.14.2

This Surveillance requires the periodic monitoring of the containment sump level. This provides assurance that leakage which would indicate reactor coolant pressure boundary degradation would be detected. The surveillance Frequency (12 hours) is based on engineering judgement and industry-accepted practice.

SR 3.3.14.3

Surveillance Requirement 3.3.14.3 is the performance of a CHANNEL CHECK of the containment atmosphere (gaseous and iodine) activity monitor. The CHANNEL CHECK gives reasonable confidence that the channels are within specification with respect to their alarm setpoints. The surveillance Frequency (12 hours), is based on engineering judgement and industry-accepted practice.

SR 3.3.14.4

Surveillance Requirement 3.3.14.4 is a performance of a CHANNEL FUNCTIONAL TEST for the containment atmosphere (gaseous and iodine) activity monitor. This test ensures that the monitor can perform its function in the desired manner. The CHANNEL FUNCTIONAL TEST verifies the alarm setpoint and relative accuracy of the instrument strings. The Frequency is based on engineering judgement and industry-accepted practice.

SR 3.3.14.5 and SR 3.3.14.6

Surveillance Requirements 3.3.14.5 and 3.3.14.6 are the performance of CHANNEL CALIBRATIONS of the containment atmosphere activity monitor and containment sump level monitor every 18 months. The CHANNEL CALIBRATION verifies the accuracy of the instrument string. The calibration includes the calibration of instruments located inside containment. The Frequency is based on engineering judgement and industry-accepted practice.

REFERENCES

1. Title 10 Code of Federal Regulations, Part 50, Appendix A, General Design Criteria for Nuclear Power Plants.

2. 52 FR 3788, NRC Interim Policy Statement on Technical Improvements for Nuclear Power Reactors, February 6, 1987.

3.3 REACTOR COOLANT SYSTEM (RCS)

B 3.3.15 Specific Activity

BASES

BACKGROUND

The purpose of the reactor coolant system (RCS) Specific Activity LCO is to limit the concentration of radionuclides in the reactor coolant and the resultant offsite dose consequences in the event of a steam generator tube rupture (SGTR). The sequence of this event includes a brief release of steam to the atmosphere through the safety and atmospheric dump valves followed by cooldown and depressurization using the turbine condenser.

This LCO contains both iodine and total specific activity limits. The iodine isotopic activities are expressed in terms of a DOSE EQUIVALENT I-131 per gram of reactor coolant. Total specific reactor coolant activity is limited on the basis of the weighted average beta and gamma energy levels in the coolant. The allowable levels are intended to limit the 2-hour dose at the site boundary to a small fraction of the 10 CFR 100 limit.

APPLICABLE SAFETY ANALYSIS

The limitation on the specific activity of the primary coolant ensures that the resulting 2-hour doses at the site boundary will not exceed an appropriately small fraction of the 10 CFR 100 limit following an SGTR with an assumed pre-existing steady state primary to secondary steam generator leakage rate of 1.0 gpm to the unaffected steam generator. The pre-existing leak has the effect of contaminating the secondary side prior to the SGTR. Operation with iodine specific activity levels greater than the LCO value is allowed provided the isotopic concentration does not exceed the limit in Figure 3.3.15-1. The 1.0 microcurie/gram limit may be exceeded temporarily (48 hours) during non-steady state conditions when power or reactor coolant pressure changes cause iodine spiking. Operation outside this limit for a

restricted time period provides time to reduce the temporarily increased iodine concentration and DOSE EQUIVALENT I-131 to within its limit. The activity levels allowed by Figure 3.3.15-1 increase the dose at the site boundary by a factor of up to twenty following a postulated SGTR. Use of Figure 3.3.15-1 is believed to be acceptable because the probability of an SGTR occurring during this short time interval is low enough to justify a higher dose limit.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)	LCO 3.3.15, Specific Activity, satisfies the requirements of Selection Criterion 2 of the NRC Interim Policy Statement (Ref. 1) because it helps ensure that reactor coolant activity will be within the initial conditions assumed in the accident analysis.
LCOs	The specific iodine activity is limited to 1.0 microcurie per gram DOSE EQUIVALENT I-131 and the total specific activity in the primary coolant is limited to the number of microcuries per gram equal to 100 divided by $\bar{E}$ (average disintegration energy of the sum of the average beta and gamma energies of the coolant nuclides). These values represent a reasonable operating capability rather than a specific analytical result. Adherence to these limits is required to restrict the 2-hour offsite dose following a SGTR to a small fraction of 10CFR100 limit. Violation of this LCO may result in coolant activity levels which exceed the generically applicable dose objective for the SGTR. The DOSE EQUIVALENT I-131 may exceed 1.0 microcurie per gram for up to 48 hours during one continuous time interval, provided that the concentration does not exceed the limit shown on Figure 3.3.15-1. This accommodates iodine spiking.
APPLICABILITY	LCO 3.3.15, Specific Activity, is applicable in MODES 1 and 2, and in MODE 3 with $T_{avg} \geq 500°F$. If a SGTR were to occur, in conjunction with a reactor trip, the energy in the primary system is sufficient to cause the secondary safety valves and the atmospheric dump valves to open in MODE 1, and possibly in MODE 2. For conservatism, the applicability has been extended to include MODE 3 with $T_{avg} \geq 500°F$. At this temperature secondary atmospheric releases are not possible because the saturation pressure at 500°F is considerably below the opening setpoint. Thus the applicability is a conservative range that bounds the possibility for site boundary doses for SGTR. The LCO is not applicable in MODE 3 with $T_{avg} \leq 500°F$ or MODES 4, 5, or 6 because releases cannot occur through the steam supply system flow paths.

(continued)

BASES (continued)

ACTIONS

A.1, A.2, B.1, B.2, C.1, C.2, D.1, and D.2

The Required Actions taken in response to high reactor coolant activity include sampling and analysis for determination of the DOSE EQUIVALENT I-131. Samples are analyzed at least every 4 hours for I-131, I-133, and I-135 if the limits of LCO 3.3.15 is exceeded. If the limits on Figure 3.3.15-1 are exceeded, or if the LCO 3.3.15 limit on iodine activity is exceeded for more than 48 hours during one continuous time interval, the reactor is placed in MODE 3 (below 500°F) within six hours. The Completion Time of 48 hours to reduce the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with processing systems and is based on engineering judgement.

Similarly, if reactor coolant specific activity exceeds the 100/$\bar{E}$ limit, the reactor is placed in MODE 3 (below 500°F) within 6 hours. This Required Action reduces the possibility of a release of activity through the main steam safety or atmospheric vent valves because 500°F is below the saturation temperature required to open these valves.

Six hours is a reasonable maximum time based on operating experience to reach MODE 3 from full power without challenging safety systems and operators, and 4 hours is a reasonable time between reactor coolant sample analyses. As such, the Completion Times are based on engineering judgement.

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1

The purpose of this periodic surveillance is to assure the coolant activity remains within the allowable limits. The surveillance is performed by drawing samples of coolant and performing a radiochemical analysis. By maintaining activity within limits, site boundary doses from a SGTR would be expected to be a small fraction of the 10CFR100 limit. The 72 hour Frequency is based on engineering judgement.

Performing the surveillance routinely while steady state conditions exist (constant power and RCS temperature) allows trends of average coolant activity to be evaluated and inferences about fuel condition to be made.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.15.2

The purpose of this surveillance is also to assure the coolant activity remains within limits to ensure that SGTR site boundary doses are a small fraction of the 10CFR100 limit. The surveillance provides more details of the isotopic content than the iodine surveillance (SR 3.3.15.1) but is performed at a less frequent interval (184 days). The Frequency is based on engineering judgement and industry-accepted practice. The Frequency NOTE has the effect of requiring a stable coolant condition free from any power change effects that would cause activity spikes or anomalies that would not represent the usual condition of the coolant.

REFERENCES

1. 52 FR 3788, NRC Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors, February 6, 1987.

BASES (continued)

LCOs

These LCOs establish the minimum equipment required to be available to accomplish the core cooling safety function following accidents which render the steam generators effectively unavailable, such as a large LOCA. Two independent (and redundant) ECCS trains are required to be OPERABLE to ensure that at least one is available assuming a single failure in the other train. Additionally, individual components within the ECCS trains may be called upon to mitigate the consequences of other transients and accidents.

During an event requiring ECCS actuation, a flow path is provided to ensure an abundant supply of water from the BWST to the RCS via the HPI and LPI pumps and their respective discharge flow paths to each of the four cold leg injection nozzles and the reactor vessel. In the long term, this flow path may be manually transferred to take its supply from the containment sump. The flow path for each train must maintain its designed independence to ensure that no single failure can disable both ECCS trains. In MODE 4, one OPERABLE ECCS train is acceptable without single failure consideration on the bases of the stable reactivity condition of the reactor and the limited core cooling requirements. In this condition one HPI and one LPI train provide sufficient injection flow to meet ECCS requirements.

In order to preclude a low temperature ovepressurization event in MODE 3 with RCS temperature $\leq$ 283°F and in MODE 4, the high pressure injection isolation valves may be closed with their power supply breakers locked in the open position. Two HPI pumps may also be deactivated when low temperature overpressurization is a concern. Regardless of the method used to deactivate HPI, operator action is then required to initiate high pressure injection. This is also considered acceptable on the bases of the stable reactivity condition of the reactor and the limited core cooling requirements.

The requirements of these LCOs are derived principally from events involving a loss of coolant inventory, and particularly the Appendix K (Ref. 5) evaluation. Failure to meet these requirements could result in the inability to match core heat generation, leading to fuel melting, increased clad metal-water reaction, and potential for alteration of core geometry for such low probability, high consequence events.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.6 (Cont'd)

This surveillance has two notes associated with it. Note 1 requires the HPI flow balance surveillance be performed in MODE 3. This surveillance energizes the high pressure injection (HPI) pumps to inject water into the reactor vessel. In order to minimize the possibility of overpressurizing the reactor vessel at low temperatures as a result of HPI operation, this surveillance is performed at plant conditions in which low temperature overpressurization is not a concern.

MODE 6 is not utilized for the testing due to the potential for overfilling the fuel transfer canal and increasing the reactor building's airborne contamination. As a result of Note 1, Note 2 makes the provision of 3.0.4 not applicable for the HPI flow balance surveillance to allow it to be performed in MODE 3 after reaching the applicable MODE.

SR 3.4.2.7

This surveillance ensures that the ECCS flow path from the BWST to the RCS is properly aligned by requiring a verification of the line up of those valves which could be inadvertently repositioned. Failure of one of these valves will affect only one ECCS train. Therefore, a monthly frequency has been established based on engineering judgement, and has been shown to be acceptable through operating experience.

SR 3.4.2.8

This surveillance ensures that the automatic isolation and interlock function of the DH system from the RCS will function if challenged. This interlock will prevent excessive RCS pressure from being exerted on the DH/LPI system through an open DH suction line from the RCS hot leg. The interlock will prevent opening and will automatically close isolation valves in the DH line when RCS pressure is above the setpoint. Excessive pressures in the DH system potentially could result in a loss of coolant accident outside of the containment. The interlock setpoint is based on preventing excessive pressure from being exerted on the DH/LPI system from the RCS. A surveillance frequency of 18 months has been established based on engineering judgement. This frequency has been shown to be acceptable through operating experience.

(continued)

TSCRN 174B

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302/LICENSE NO. DPR-72
REQUEST NO. 174, REVISION 0
PRESSURE/TEMPERATURE LIMITS

B. LICENSE DOCUMENT INVOLVED: Technical Specifications

PORTIONS: 3.4.9.1
Figure 3.4-2
Figure 3.4-3
Figure 3.4-4

DESCRIPTION OF REQUEST:

This submittal requests that reactor coolant system pressure and temperature limits be revised to allow for reactor operation during the first 15 effective full power year (EFPY) service period. The maximum heatup and cooldown rates assumed in the analyses and implemented in Technical Specifications (TS) have been revised. References to the 10CFR50 Appendix G criticality limit curve have been deleted from the TS and Bases. The Bases have also been revised to reflect the most recent surveillance capsule results. RT_{NDT} values at the 1T/4 and 3T/4 flaw locations for 21 EFPY and the latest information on chemical composition of the limiting reactor coolant system (RCS) components were added.

REASON FOR REQUEST:

Title 10 of the Code of Federal Regulations, Part 50, Appendix G requires reactor vessel pressure/ temperature limits be established in order to ensure fracture toughness requirements are satisfied for the current operational service period. Current Crystal River Unit 3 (CR-3) pressure/ temperature limit curves are conservative for the first 8 EFPY. CR-3 is approaching the end of this service period and is submitting revised pressure/ temperature limits to allow for continued operation.

The maximum allowable heatup and cooldown rates assumed in the fracture mechanics analyses have been revised to more accurately reflect actual plant capabilities. Previously assumed values for heatup and cooldown rate were not representative of actual plant operating conditions.

Administrative changes were required to make the TS and bases consistent with the current CR-3 licensing basis. References to the 10CFR50 Appendix G Criticality Limit curve have been deleted. While this curve is still calculated (as required by 10CFR50 Appendix G) the requirement to include it in TS was deleted by CR-3 License Amendment No. 82, dated September 23, 1985. The Bases are revised to include the most recent reactor vessel surveillance capsule data and reflect the use of Regulatory Guide 1.99 Revision Number 2 in the preparation of the 15 EFPY CR-3 pressure/ temperature limit curves.

EVALUATION OF REQUEST:

The Heatup, Cooldown, and Inservice Leak and Hydrostatic Testing pressure/ temperature limit curves provide a substantial margin between fracture toughness limits and actual plant operating conditions. This ensures that the reactor coolant pressure boundary (RCPB) is protected against non-ductile failures due to anticipated mass or energy inputs to the RCS. After the first several EFPY, the reactor vessel becomes the most limiting component of the RCPB in terms of fracture toughness. Fast-neutron irradiation causes a decrease in the ability of the vessel to absorb stresses and resist fracture. The concern is a crack that is undetected during inservice inspection will propagate and result in a non-ductile failure of the reactor vessel. This failure of the RCPB would have a serious impact on safety, particularly at rated RCS pressure and temperature. Therefore it is imperative that fracture mechanics limits be provided that are conservative for the period of operation they are in use.

This request submits updated pressure/ temperature limit curves for the first 15 EFPY. The 15 EFPY curves (reported in BAW-2091) are calculated for the predicted fluence the vessel will be exposed to at the end of 15 EFPY and are conservative until that time. The fluence values are based on the latest surveillance capsule results for CR-3 and are published in BAW-2049 "Analysis of Capsule CR3-F Florida Power Corporation Crystal River Unit-3" September 1988. The pressure/ temperature limit curves are prepared in accordance with 10CFR50 Appendix G and an NRC-approved methodology documented in BAW-10046A, Rev. 2 "Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G", June 1986. Regulatory Guide 1.99 Revision 2 "Radiation Embrittlement of Reactor Vessel Materials" May 1988 was used to predict the shift in reference temperature (RT_{NDT}) as a function of fluence and vessel chemistry. This methodology was endorsed by the NRC Staff in Generic Letter 88-11 as an acceptable method for predicting the effect of neutron irradiation on reactor vessel materials. The change in methodology necessitates a revision to the bases to include information pertinent to the new methods and remove superseded data based on previous revisions of Regulatory Guide 1.99.

The maximum allowable heatup and cooldown rates are consistent with the assumptions used in the fracture mechanics analysis. Changes in RCS temperature result in thermal stresses to the vessel. The larger the allowable rate of change of temperature, the larger the magnitude of the resultant thermal stress. This request reduces the allowable heatup and cooldown rates to make the rates more consistent with actual plant capabilities. The lower rates result in lower thermal stresses to the vessel. These limits ensure that the operator will not heatup or cooldown the RCS at an unanalyzed rate.

SHOLLY EVALUATION OF REQUEST:

Florida Power Corporation (FPC) proposes the revision to the pressure/ temperature limit curves and the reduction in the maximum allowable heatup and cooldown rates does not involve a significant hazard consideration. The revision of the pressure/ temperature limit curves is necessary to ensure conservative protection of the RCPB for the first 15 EFPY of reactor operation. Furthermore, the limit curves were prepared in accordance with an NRC-approved methodology. Technical Specifications will continue to require the plant to be operated within the limits of the curves and allowable heatup and cooldown limitations and include appropriate actions to be taken in the event the limits are violated.

FPC concludes this change will not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated because the revision of the pressure/ temperature limit curves and the reduction in allowable heatup and cooldown rates has no influence or impact on the probability of a Design Basis Accident (DBA) occurrence.

   CR-3 Technical Specifications will continue to require operation of the RCS within the pressure/ temperature and heatup/ cooldown limits. The reduction in the magnitude of the allowable heatup and cooldown rates results in generally lower thermal stresses applied to the reactor vessel than are currently allowed. The methodology used in preparation of the curves has been reviewed and approved by the NRC and is unchanged from previous pressure/ temperature limit revisions (With the exception of the current usage of Regulatory Guide 1.99 Revision 2). Regulatory Guide 1.99 Revision 2 has been endorsed by the NRC Staff as providing improved empirical correlations for predicting the effects of irradiation on the properties of reactor vessel materials.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated because updating the pressure/ temperature limit curves and reducing the allowable RCS heatup and cooldown rates introduces no new mode of plant operation nor does it require a physical modification to the plant.

3. Involve a significant reduction in the margin of safety because the CR-3 15 EFPY pressure/ temperature limits are prepared using the same NRC-approved methodology used to generate the current 8 EFPY pressure/ temperature limit curves. The decrease in reactor vessel fracture toughness

(CONTINUED)

SHOLLY EVALUATION OF REQUEST (CONTINUED):

(due to accumulated neutron exposure) has been accounted for in the preparation of the 15 EFPY pressure/ temperature limit curves using the methodology of NRC Regulatory Guide 1.99 Revision 2. This has resulted in lower allowable RCS pressures for given RCS temperatures in order to maintain acceptable levels of stress within the vessel and ensure the margin of safety is not decreased.