Florida Power
CORPORATION

October 31, 1978

P

Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

Dear Sir:

Enclosed are three (3) originals and forty (40) copies of Technical Specification Change Requests No. 27 and 34 requesting amendment to Appendix A of Operating License No. DPR-72. As part of this request, proposed replacement pages for Appendix A are enclosed.

Also enclosed is FPC's licensing fee, in the amount of eight thousand dollars ($8,000) and a signed copy of Certificate of Service for Technical Specification Change Requests No. 27 and 34 to the Chief Executive of Citrus County, Florida.

Very truly yours,

FLORIDA POWER CORPORATION

W.P. Stewart

W.P. Stewart

WPS/ECS/emf
M03 (10/30)

cc: Office of Inspection & Enforcement
U.S. Nuclear Regulatory Commission
101 Marietta Street, Suite 3100
Atlanta, Ga 30303
File: 3-0-3-a-3

7811080165 P

STATE OF FLORIDA

COUNTY OF PINELLAS

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

W.P. Stewart

W.P. Stewart

Subscribed and sworn to before me, a Notary Public in and for the State and County above named, this 31st day of October, 1978.

Charles K Pope

Notary Public

Notary Public, State of Florida at Large,
My Commission Expires: July 25, 1980

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

CERTIFICATE OF SERVICE

W.P. Stewart deposes and says that the following has been served on the 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

An original copy of Technical Specification Change Requests No. 27 and 34 requesting amendment to Appendix A of Operating License No. DPR-72.

FLORIDA POWER CORPORATION

W. P. Stewart

W.P. Stewart
Director, Power Production

SWORN TO AND SUBSCRIBED BEFORE ME THIS 31st DAY OF OCTOBER, 1978.

Charles R Page

Notary Public

Notary Public State of Florida at Large
My Commission Expires: July 25, 1980

(NOTARIAL SEAL)

CRYSTAL RIVER UNIT NO. 3
DESIGN INFORMATION AND TECHNICAL SPECIFICATIONS
FOR REACTOR COOLANT PUMP POWER MONITORS

The Reactor Coolant Pump Power Monitor (RCPPM) is designed to anticipate a loss or reduction of the reactor coolant flow by monitoring reactor coolant pump power and detecting abnormal power conditions indicative of an inoperable pump. The status of each pump will be transmitted by the RCPPM to each of four Reactor Protection System (RPS) channels. Two RCPPMs will be supplied to provide redundant pump status information to each RPS channel. Logic in the RPS will act on the pump status information and take appropriate action as follows:

TABLE

	RPS ACTION		
No. of Pumps Inoperable	100%	78%	51.2%
1	Action by flux/flow comparator	None	None
2 (Diff. Loops)	Trip Reactor	Trip Reactor	Trip Reactor
2 (Same Loops)	Trip Reactor	Trip Reactor	Trip Reactor
3	Trip Reactor	Trip Reactor	Trip Reactor
4	Trip Reactor	Trip Reactor	Trip Reactor

As stated in the Accident Analyses of the CR#3 FSAR, in the event that a loss of reactor coolant flow due to failure of one or more of the RC pumps was to occur at the present licensed power level or 2452 MWt, the transient is terminated by the present RPS flux-flow trip. The present RPS action is quick enough to preclude the minimum DNB ratio from going below 1.30 for the four pump coastdown transient and below 1.00 for the locked rotor transient.

However, at thermal power levels greater than 2500 MWt, RPS action by the flux-flow comparator is not fast enoubh in event of loss of more than one RC pump to preclude the minimum DNB ration from going below the acceptance criteria. Therefore, for power levels above 2500 MWt, Nuclear Overpower based on Reactor Coolant Pump Power Monitors must be added to the RPS trip functions as this will reduce the response time of the Reactor Protection System from 650 ms to 300 ms and thereby terminate the transient quick enough to insure that the minimum DNB ratio limits are not violated.

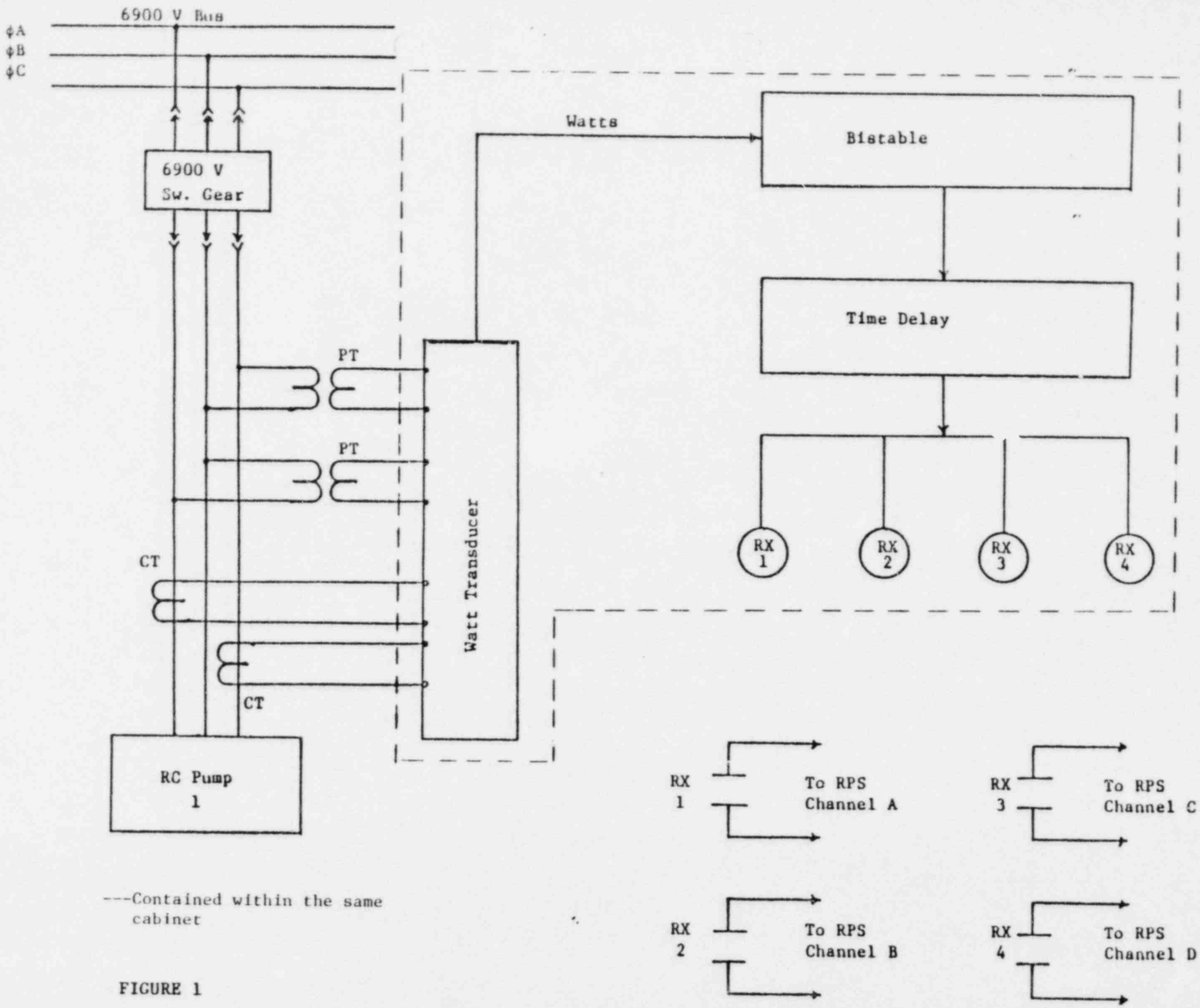
Figure 1 shows the proposed scheme for one RCPPM string. Two current transformers and a 3-phase potential transformer measure the current and voltage on the RCP pump power feed lines. The transformers input into a Hall effect watt transducer which produces an output signal proportional to real power. This power signal is fed into a bistable which provides a contact output for selected overpower and underpower setpoints. The bistable output contact is wired to a time delay relay to provide a time delay function sufficient to permit normal pump bus transfer without reporting a "pump inoperable" status to the RPS. The output of the timer actuates four separate relays. A contact from each relay is wired to its respective RPS channel. Thus, one pump monitor string provides status information for one pump to each of four RPS channels. An identical, redundant string (not shown in Figure 1) using separate transformers and monitoring equipment again provides status information for the same pump to the four RPS channels. In the event of a failure of one string, all four RPS channels would still have the necessary pump status information via the redundant string.

Figure 2 shows a block diagram of the complete RCPPM system. The system will be packed such that equipment belonging to redundant strings will be in enclosures separated by barriers. Contact outputs from the RCPPM cabinets to the four RPS channels will be arranged to provide adequate physical separation and electrical isolation between each channel. Cable and equipment separation for this installation will be in accordance with IEEE 384-1977 and Regulatory Guide 1.75. Where separation cannot be maintained, physical barriers will be included.

RCPPM cabinets and equipment specified will be seismically qualified and located in a Class I structure. All supports for engineered safeguards cable trays and conduits are designed for OBE and SSE using the acceleration floor response spectra developed for applicable levels of the containment building, auxiliary building, intermediate building and the control complex.

The current and potential transformers will not be seismically qualified. However, separation of the cables carrying redundant transformer outputs to the RCPPM cabinets will be provided in accordance with the separation criteria stated above. The current and potential transformers are not seismically qualified because they are not required to safely shutdown the reactor. The loss of the current or poential transformers will result in a "pump inoperable" signal to the RPS system. Upon the receipt of two such signals, whatever the cause, the RPS will trip the reactor.

Attached are the proposed revisions to the Technical Specifications for CR#3 as a result of the installation of Reactor Coolant Pump Power Monitors.



---Contained within the same cabinet

FIGURE 1

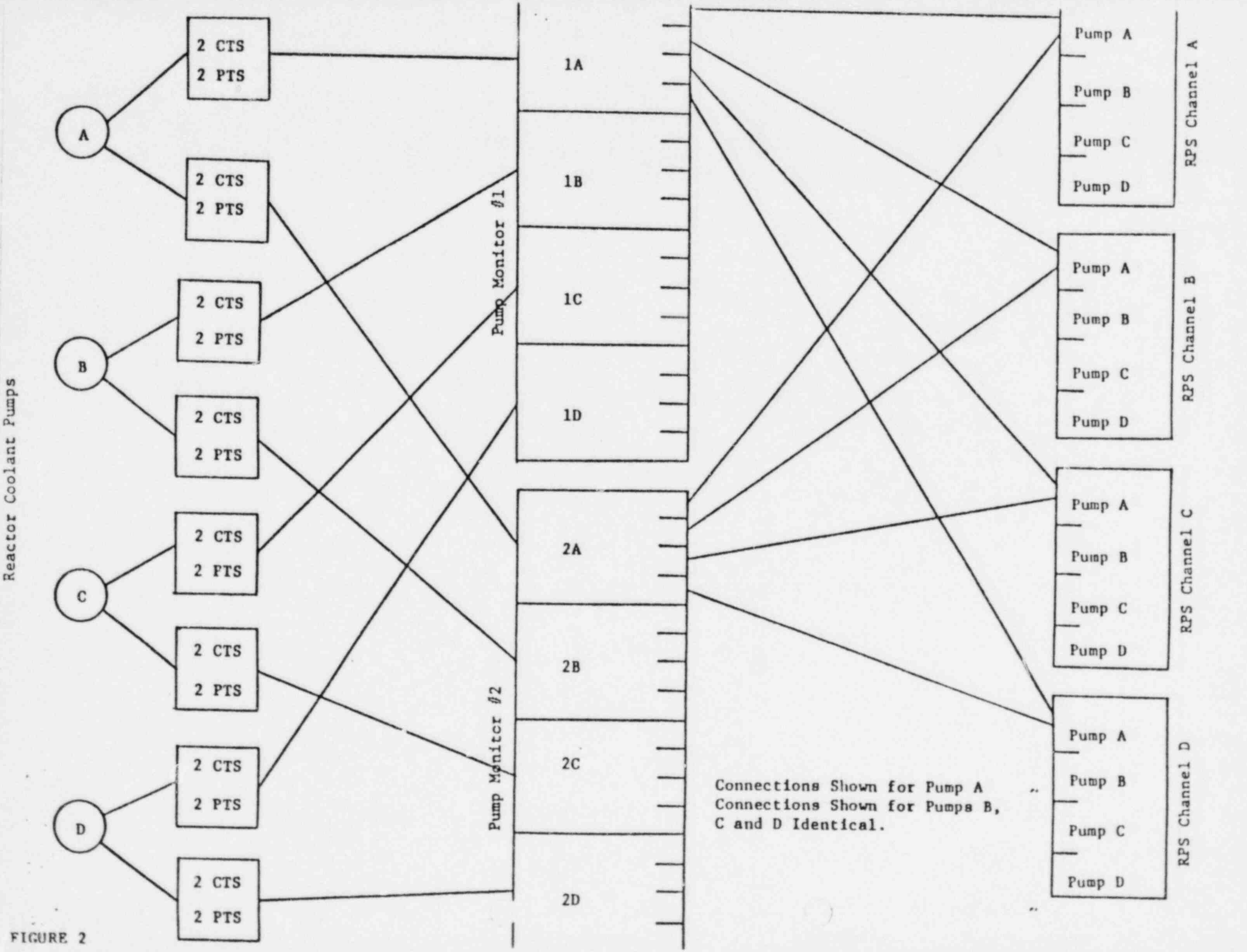

Reactor Coolant Pumps
A
B
C
D
2 CTS
2 PTS
Pump Monitor #1
1A
1B
1C
1D
Pump Monitor #2
2A
2B
2C
2D
RPS Channel A
RPS Channel B
RPS Channel C
RPS Channel D
Pump A
Pump B
Pump C
Pump D
Connections Shown for Pump A
Connections Shown for Pumps B, C and D Identical.

FIGURE 2

TECHNICAL SPECIFICATION CHANGE REQUEST NO. 27

Replace pages 2-6, B 2-7, 3/4 3-2, 3/4 3-6, and 3/4 3-7 with the attached revised pages 2-6, B 2-7, B 2-8, 3/4 3-2, 3/4 3-6, and 3/4 3-7.

Proposed Change

The above pages are being revised to include Reactor Coolant Pump Power Monitors (RCPPM) in the CR#3 Technical Specifications. The pump monitors will be installed during the first refueling outage of CR#3. It is requested that this technical specification change become effective upon completion of installation and testing of the monitors.

Reason for Proposed Change

Florida Power Corporation is presently discussing with the NRC staff our intent to request that the power level of CR#3 be increased from 2452 MWt to the FSAR ultimate core power level of 2544 MWt. It is our intent to receive approval from the NRC for this power increase in order that the upgrade can be accomplished at the first refueling outage of CR#3. As part of this power upgrade, B&W has indicated that Reactor Coolant Pump Power Monitors (RCPPM) must be installed at CR#3 to preclude going below the minimum DNB ratio limits in the event of a loss of coolant flow accident at CR#3.

The Reactor Coolant Pump Power Monitor is designed to anticipate a loss or reduction of reactor coolant flow by monitoring reactor coolant pump power and detecting abnormal power conditions indicative of an inoperable pump. The status of each pump will be transmitted by the RCPPM to each of four Reactor Protection System (RPS) channels. Two RCPPMs will be supplied to provide redundant pump status information to each RPS channel. Logic in the RPS will act on the pump status information and take appropriate action as follows:

TABLE

No. of Pumps Inoperable	RPS ACTION 100%	78%	51.2%
1	Action by flux/flow comparator	None	None
2 (Diff. Loops)	Trip Reactor	Trip Reactor	Trip Reactor
2 (Same Loops)	Trip Reactor	Trip Reactor	Trip Reactor
3	Trip Reactor	Trip Reactor	Trip Reactor
4	Trip Reactor	Trip Reactor	Trip Reactor

As stated in the Accident Analyses of the CR#3 FSAR, in the event that a loss of reactor coolant flow due to failure of one or more of the RC pumps was to occur at the present licensed power level or 2452 MWt, the transient is terminated by the present RPS flux-flow trip. The present RPS action is quick enough to preclude the minimum DNB ratio from going below 1.30 for the four pump coastdown transient and below 1.00 for the locked rotor transient.

However, at thermal power levels greater than 2500 MWt, RPS action by the flux-flow comparator is not fast enough in event of loss of more than one RC pump to preclude the minimum DNB ratio from going below the acceptance criteria. Therefore, for power levels above 2500 MWt, Nuclear Overpower based on Reactor Coolant Pump Power Monitors must be added to the RPS trip functions as this will reduce the response time of the Reactor Protection System and thereby terminate the transient quick enough to insure that the minimum DNB ratio limits are not violated. The pump power monitors are being added in addition to the present flux-flow comparator.

Safety Analysis Justifying Proposed Change

The licensing submittal in support of the proposed power level increase will be submitted as part of the reload report for the first refueling for CR#3. Included in this licensing submittal will be a reanalysis of the loss of coolant flow accident due to four pump coastdown and the locked rotor cases. This safety analysis will demonstrate that the addition of the Reactor Coolant Pump Power Monitors at CR#3 will prevent violation of the minimum DNB ratio limits for the four pump coastdown and locked rotor cases.

TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIOn NIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Reactor Containment Vessel Pressure High	$\leq$ 4 psig	$\leq$ 4 psig
9. Nuclear Overpower based on Pump Monitors(1)	$\leq$ 0.00% of RATED THERMAL POWER with one pump operating in each loop	$\leq$ 0.28% of RATED THERMAL POWER with one pump operating in each loop
	$\leq$ 0.00% of RATED THERMAL POWER with two pump operating in one loop and no pump operating in the other loop	$\leq$ 0.28% of RATED THERMAL POWER WITH two pumps operating in one loop and no pump operating in the other loop
	$\leq$ 0.00% of RATED THERMAL POWER with no pumps operating or only one pump operating	$\leq$ 0.28% of RATED THERMAL POWER with no pumps operating or only one pump operating

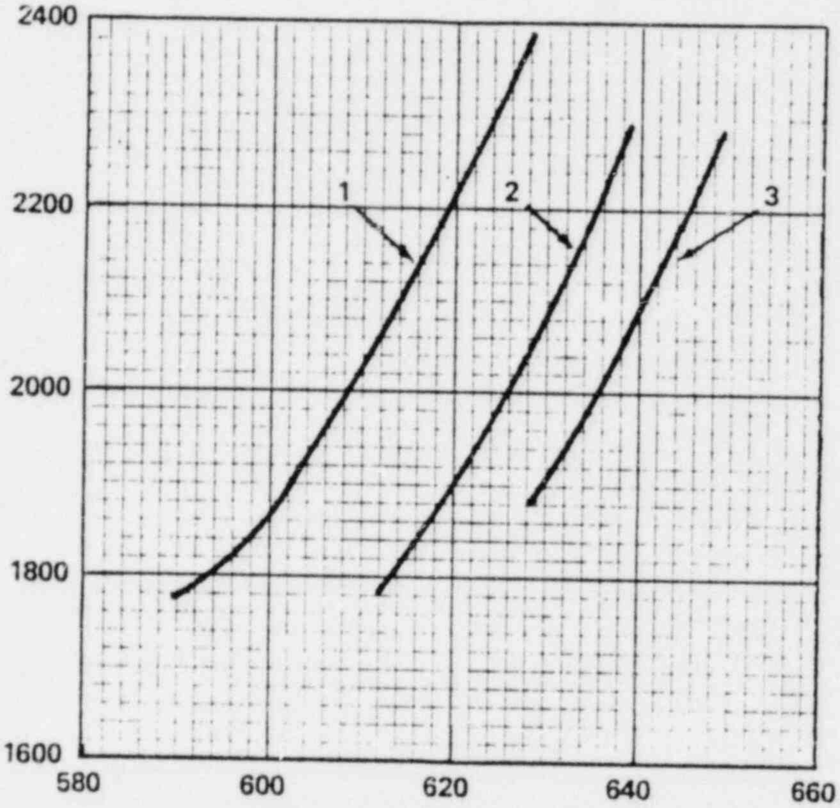
(1) Trip may be manually bypassed when RCS pressure $\leq$ 1720 psig by actuating Shutdown Bypass provided that:

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

BASES

Nuclear Overpower Based on Pump Monitors

In conjunction with the power/imbalance/flow trips, the Nuclear Overpower Based On Pump Monitors trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.



Curve	Reactor Coolant Flow (LBS/HR)	Power	Pumps Operating (Type of Limit)
1	131.3×10^6 (100%)	112%	Four Pumps (DNBR Limit)
2	98.1×10^6 (74.7%)	84%	Three Pumps (DNBR Limit)
3	64.4×10^6 (49.0%)	57%	One Pump in each loop(Quality Limit)

Pressure/Temperature Limits at Maximum
Allowable Power for Minimum DNBR

BASES Figure 2.1

TABLE 3.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

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

TABLE 3.3-2

REACTOR PROTECTION SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIMES
1. Manual Reactor Trip	Not Applicable
2. Nuclear Overpower*	≤ 0.3 seconds
3. RCS Outlet Temperature--High	Not Applicable
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE*	≤ 1.4 seconds
5. RCS Pressure--Low	≤ 0.5 seconds
6. RCS Pressure--High	≤ 0.5 seconds
7. Variable Low RCS Pressure	Not Applicable
8. Reactor Containment Pressure--High	Not Applicable
9. Nuclear Overpower Based on Pump Monitor*	≤ 0.47 seconds

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

TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

Technical Specification Change Request No. 34 [Appendix A]

Replaces pages IX, 3/4 4-24, 26, 27, 28, B3/4 4-6, 7, 8, 9, 10, 11, 12 and 13 with the attached revised pages IX, 3/4 4-24, 26, 27, 28, B3/4 4-6, 7 and 8.

Proposed Change

Revisions to the heatup, cooldown, and inservice leak and hydrostatic test pressure-temperature curves are proposed as well as the supporting bases.

Reason For Proposed Change

B&W has advised that as a result of chemical analysis performed on archive weldments in conjunction with the B&W owner's group program for evaluation of reactor vessel material properties, it was discovered that weld filler wire atypical of the submerged arc weld filler wires used by B&W in the construction of nuclear pressure vessels was unknowingly mixed by the supplier with a shipment of Mn-Mo-Ni filler wire. The atypical weld wire has high Silicon and low Nickel contents which are outside of the typical range for the Mn-Mo-Ni filler wires specified by B&W; specifically, the Ni content was 0.1% (typically 0.6%) and the Si content was 1.0% (typically 0.5%). There are no directly applicable irradiation data for the atypical weldment although other applicable data exists, and furthermore, welds of this wire possess a higher than normal unirradiated reference temperature, RT_{NDT}. Weldments prepared from the atypical weld wire exhibit a very adequate Charpy Upper Shelf Energy.

This wire mixture (Mn-Mo-Ni filler wire plus the atypical filler wire) may have been used in the construction of seven B&W reactor vessels. The weld locations of interest in the B&W commercial vessels where the wire mixture may have been used are divided into three categories which are the Closure Head-to-Flange, Outlet Nozzle-to-Nozzle Belt, and Beltline Region.

The seven B&W reactor vessels were constructed in accordance with ASME B&PV Code, Section III, 1964 Edition, Addenda through Summer 1967. With the information available at the time of construction, the reactor vessels met the ASME Code.

A technical evaluation has been performed on B&W reactor vessels assuming the atypical material is in each of the three RV locations mentioned above. This fracture mechanics evaluation demonstrates that the structural integrity of the reactor vessel has not been compromised by the possible presence of the atypical material. However, for those vessels with the atypical weld material assumed to be present, the future operation of the plant may be governed by more restrictive pressure-temperature operating limits.

The analytical procedures of BAW-10046A, Rev. 1 have been employed with the exception of circumferential welds in the beltline region. For the circumferential welds, the postulated flaw as specified in BAW-10046A has been oriented in the circumferential direction. The component of stress due to pressure used in the fracture mechanics analysis is then the longitudinal stress. The hoop stress component is used for postulated flaws in the longitudinal seams.

Weld Location Where Weld Wire 72105 Has Been

Used That May Contain Atypical Weld

Contract	Location	Date	Possible % Of Mixed Weld in Wire
NSS-7 [CR3]	Upper Shell-Lower Shell	12-69	100%
	Core Flood & Outlet Nozzle to Nozzle Belt	1-70	100%
	Surveillance [Specimens]	6-71 & 3-73	100%

FPC committed in a letter to the USNRC dated August 31, 1978 to propose a Technical Specification Change incorporating these revised operating curves into the Technical Specifications prior to November 1, 1978.

Safety Analysis Justifying Proposed Change

As stated above, this change was developed using the same assumptions and analytical methods as the original pressure-temperature operating curves except the atypical weld material was assumed to be present. The presence of the atypical weld material has the effect of reducing the conservatism included in the present pressure-temperatue curves for CR#3. However, this reduction is not of sufficient magnitude such that the resultant change in the operating limits violates the present technical specification curves or any previous safety analyses. As the pressure-temperature curves being submitted in this change request are based on the assumption that the atypical weld material is present, the implementation of these curves will add conservatism to the safety analysis and provide the same level of safety margins as were present in the original curves based on typical weld wire material. Therefore, this change does not involve the changing of any safety analysis assumptions for Crystal River #3.

ECS/ldd/F01

BASES

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

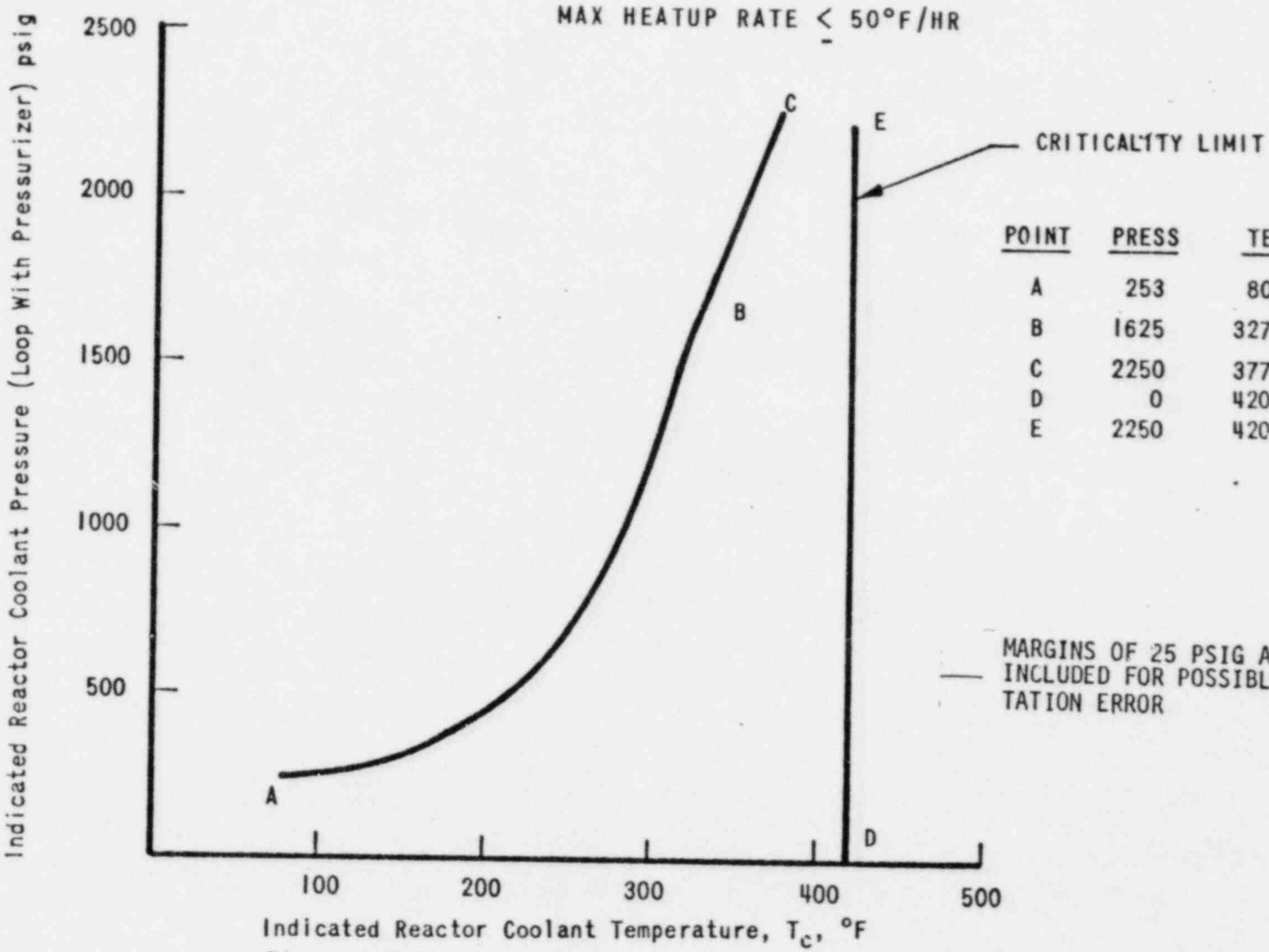
APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce RCS T_{avg} and pressure to less than 200°F and 145 psig, respectively, within the following 30 hours.

CR-3

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS FOR HEATUP FOR THE FIRST 5 EFPY



POINT	PRESS	TEMP
A	253	80°F
B	1625	327°F
C	2250	377°F
D	0	420°F
E	2250	420°F

MARGINS OF 25 PSIG AND 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENTATION ERROR

Figure 3.4-2

CR-3

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS
FOR COOLDOWN FOR THE FIRST 5 EFPY

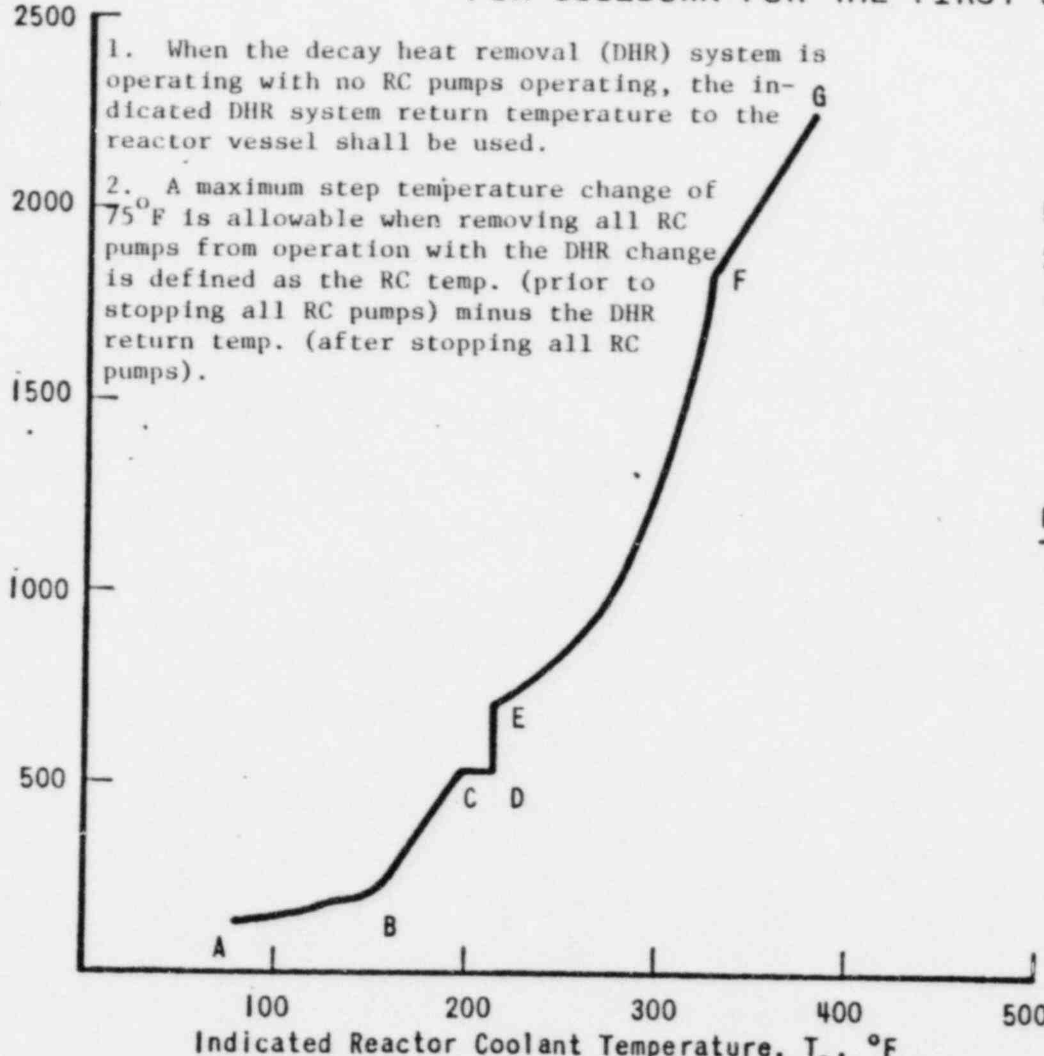


Figure 3.4-3

COOLDOWN RATE

TEMP	MAX RATE
550-280	100°F/HR
280-150	50°F/HR
150-	25°F/HR

POINT	PRESS	TEMP
A	145	80°
B	250	160°
C	525	195°
D	525	215°
E	720	215°
F	1850	330°
G	2250	377°

MARGINS OF 25 PSIG AND 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENTATION ERROR

CRYSTAL RIVER 3

INSERVICE LEAK AND HYDROSTATIC TEST (5 EFPY) HEATUP AND COOLDOWN

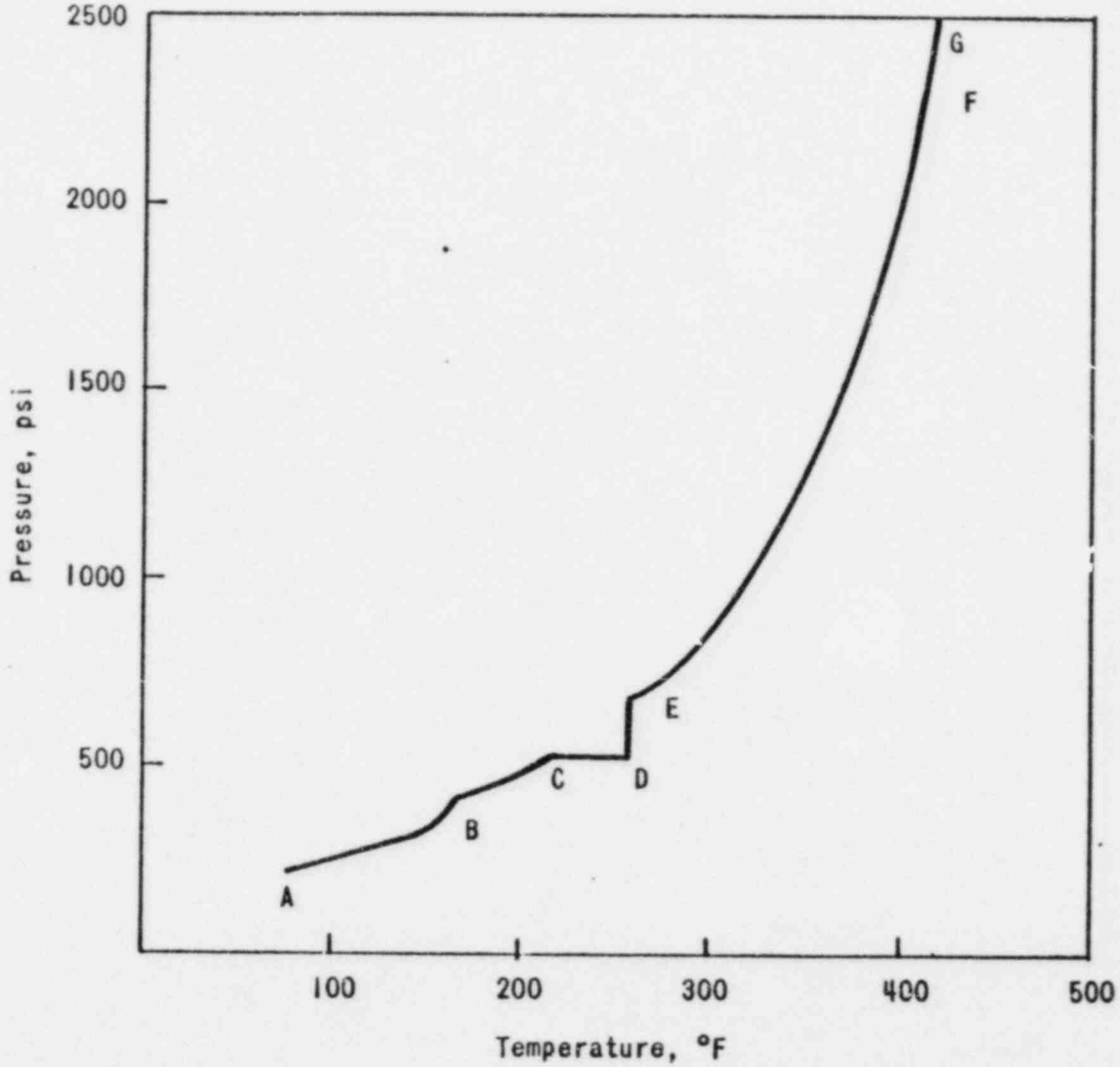


Figure 3.4-4

MAX HEATUP RATE $\leq$ 50°F/HR

MAX COOLDOWN RATES:

TEMP	C.D. MAX RATES
550-280	100°F/HR
280-150	50°F/HR
150-	25°F/HR

POINT	PRESS	TEMP
A	220	75
B	420	165
C	525	213
D	525	255
E	675	256
F	2250	410
G	2500	420

MARGINS OF 25 PSIG AND 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENTATION ERROR

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity >1.0 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accomodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 10 percent of the unit's yearly operating time since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to <500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4 4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.2.4 of the FSAR. During heatup and cooldown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The heatup limit curve, Figure 3.4-2 is a composite curve which was prepared by determining the most conservative case, with either the 1/4T or 3/4T wall location controlling, for any heatup rate up to the indicated maxima per hour. The cooldown limit curve, Figure 3.4-3, is a composite curve which was prepared based upon the same type analysis.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper, and phosphorous content of the material in question has been calculated. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 account for predicted adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The normal heatup, normal cooldown and inservice leak and hydrostatic test curves must be recalculated with the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt pre-load). The outlet nozzles of the reactor vessel affect the pressure-temperature limit curves of the first several service periods. For the service period for which the limit curves are established, the maximum allowable pressure as a function of fluid temperature is obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and beltline region. The maximum allowable pressure is taken to be the lower pressure of the three calculated pressures. The calculated pressure temperature limit curves are then adjusted by 25 psi and 10°F for possible errors in the pressure and temperature sensing instruments. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations. The limit curves were prepared based upon the most limiting adjusted reference temperature of all the beltline region materials at the end of the fifth effective full power year.

The unirradiated transverse impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determined for those materials for which sufficient amounts of material were available. The adjusted reference temperature is calculated by adding the predicted radition-induced ΔRT_{NDT} and the unirradiated. The predicted ΔRT_{NDT} are calculated in accordance with Regyulatory Guide 1.99 using the respective neutron fluence and copper and phosphorus contents.

The assumed unirradiated reference temperature of the beltline region controlling material is 120°. At the end of five effective full power years this increases to 252°F at 1/4T and 185°F at 3/4T. The RT_{NDT} at the outlet juncture is taken to be 120°F. The RT_{NDT} in the closure head is assumed to be 60°F. Pressure - temperature constraints are computed in accordance with BAW 10046A with postulated flaws taken to be fully contained within the material region of interest.

Figure 3.4-2 presents the pressure-temperature limit curve for normal heatup. This figure also presents the core criticality limits as required by Appendix G to 10 CFR 50. Figure 3.4-3 presents the pressure temperature limit curve for normal cooldown. Figure 3.4-4 presents the pressure-temperature limit curves for heatup and cooldown for inservice leak and hydrostatic testing.

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

The pressure and temperature limits shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

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

3/4.4.10 STRUCTURAL INTEGRITY

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

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