Docket No. 50-302

June 4, 1990

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Mr. Percy M. Beard, Jr.
Senior Vice President,
Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Operations
Licensing
P. O. Box 219-NA-2I
Crystal River, Florida 32629

Dear Mr. Beard:

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT RE: EMERGENCY FEEDWATER PUMP AND LOW PRESSURE INJECTION SYSTEM BLOCK/TRIP AND ESF RESPONSE TIMES (TAC NO. 76135)

The Commission has issued the enclosed Amendment No. 130 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). This amendment consists of changes to the Technical Specifications (TS) in response to your application dated February 13, 1990.

This amendment changes the TS to allow elimination of the automatic simultaneous operation of the motor-driven emergency feedwater pump and the low pressure injection system when offsite power is not available, and changes response times for the low pressure and high pressure injection systems.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Harley Silver, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 130 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

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cc:
Mr. A. H. Stephens
General Counsel
Florida Power Corporation
MAC - A5D
P. O. Box 14042
St. Petersburg, Florida 33733

Mr. P. F. McKee, Director
Nuclear Plant Operations
Florida Power Corporation
P. O. Box 219-NA-2C
Crystal River, Florida 32629

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
1700 Rockville Pike, Suite 525
Rockville, Maryland 20852

Senior Resident Inspector
Crystal River Unit 3
U.S. Nuclear Regulatory Commission
15760 West Powerline Street
Crystal River, Florida 32629

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street N.W., Suite 2900
Atlanta, Georgia 30323

Mr. Jacob Daniel Nash
Office of Radiation Control
Department of Health and
Rehabilitative Services
1317 Winewood Blvd.
Tallahassee, Florida 32399-0700

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

State Planning and Development
Clearinghouse
Office of Planning and Budget
Executive Office of the Governor
The Capitol Building
Tallahassee, Florida 32301

Chairman
Board of County Commissioners
Citrus County
110 North Apopka Avenue
Inverness, Florida 32650

Mr. Rolf C. Widell, Director
Nuclear Operations Site Support
Florida Power Corporation
P.O. Box 219-NA-2I
Crystal River, Florida 32629

Mr. Gary L. Boldt
Vice President, Nuclear Production
Florida Power Corporation
P. O. Box 219-SA-2C
Crystal River, Florida 32629

DATED: June 4, 1990

AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-72-CRYSTAL RIVER UNIT 3

Docket File
NRC & Local PDRs
PDII-2 Reading
S. Varga, 14/E/4
G. Lainas, 14/H/3
H. Berkow
D. Miller
H. Silver
G. Hsii 8/E/23
J. Stewart 7/E/12
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
G. Hill (4), P-137
Wanda Jones, P-130A
J. Calvo, 11/F/23
J. Miller, 11/F/23
ACRS (10)
GPA/PA
OC/LFMB
M. Sinkule, R-II
J. Stewart
G. Hsii

cc: Plant Service list

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 130
License No. DPR-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power Corporation, et al. (the licensees) dated February 13, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

    Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 130, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: June 4, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 130

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove	Insert
3/4 3-10	3/4 3-10
3/4 3-14	3/4 3-14
--	3/4 3-14a
3/4 3-17	3/4 3-17
3/4 7-4	3/4 7-4

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint Value.

b. With an ESFAS instrumentation channel inoperable, take the action shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION					
a. High Pressure Injection					
1. Manual Initiation	2	1	2	1, 2, 3, 4	13
2. Reactor Bldg. Pressure High	3	2	2	1, 2, 3	9#
3. RCS Pressure Low	3	2	2	1, 2, 3*	9#
4. RCS Pressure Low-Low	3	2	2	1, 2, 3**	9#
5. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10
b. Low Pressure Injection					
1. Manual Initiation	2	1	2	1, 2, 3, 4	13
2. Reactor Bldg. Pressure High	3	2#####	2	1, 2, 3	9#
3. RCS Pressure Low-Low	3	2	2	1, 2, 3**	9#
4. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10
2. REACTOR BLDG. COOLING					
a. Manual Initiation	2	1	2	1, 2, 3, 4	13
b. Reactor Bldg. Pressure High	3	2	2	1, 2, 3	9#
c. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. REACTOR BLDG. ISOLATION					
a. Manual Initiation	2	1	2	1, 2, 3, 4	13
b. Reactor Bldg. Pressure High	3	2	2	1, 2, 3	9#
c. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10
d. Deleted					
e. RCS Pressure Low (HPI Isolation)	3	2	2	1, 2, 3*	13
f. Automatic Actuation Logic (HPI Isolation)	2	1	2	1, 2, 3, 4	10

TABLE 3.3-3 (Continued)

TABLE NOTATION

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

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

***Trip function may be bypassed in this MODE with steam generator pressure below 750 psig. Bypass shall be automatically removed when steam generator pressure exceeds 750 psig.

#The provisions of Specification 3.0.4 are not applicable.

##Trip automatically bypassed below 20 percent of RATED THERMAL POWER.

###Trip function may be bypassed below 10 percent of RATED THERMAL POWER.

Manual trip function occurs if two channels in the same train are actuated.

Trip function for LPI pump start is automatically bypassed when off-site power is not available. Bypass is automatically removed when off-site power is restored.

ACTION STATEMENTS

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

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

ACTION 11 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 12 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the minimum channels OPERABLE required is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for Surveillance testing per Specification 4.3.2.1.1.

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

ACTION 14 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour, one additional channel may be bypassed for up to 2 hours for Surveillance testing per Specification 4.3.2.1.1.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
5. REACTOR BLDG. ISOLATION		
a. ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	$\leq$ 4 psig	$\leq$ 4 psig
3. Automatic Actuation Logic	Not Applicable	Not Applicable
4. Deleted		
5. RCS Pressure Low (HPI Isolation)	$\geq$ 1500 psig	$\geq$ 1500 psig
6. Automatic Actuation Logic (HPI Isolation)	Not Applicable	Not Applicable

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TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS*
1. Manual	
a. High Pressure Injection	Not Applicable
b. Low Pressure Injection	Not Applicable
c. Reactor Building Cooling	Not Applicable
d. Reactor Building Isolation	Not Applicable
e. Reactor Building Spray	Not Applicable
f. Reactor Building Purge Isolation	Not Applicable
g. MFW and MSL Isolation	
1. Emergency Feedwater Actuation	Not Applicable
2. Feedwater Isolation	Not Applicable
3. Steam Line Isolation	Not Applicable
h. Emergency Feedwater Actuation	Not Applicable
2. Reactor Building Pressure-High	
a. High Pressure Injection	35*
b. Low Pressure Injection	35*
c. Reactor Building Cooling	25*
d. Reactor Building Isolation	60*
3. Reactor Building Pressure High-High (with HPI signal)	
a. Reactor Building Spray	56*
4. RCS Pressure Low	
a. High Pressure Injection	35*
b. HPI Isolation	60*
5. RCS Pressure Low-Low	
a. High Pressure Injection	35*
b. Low Pressure Injection	35*
6. Low Steam Generator Pressure	
a. Feedwater Isolation	34
b. Steam Line Isolation	5
c. Emergency Feedwater Actuation	Not Applicable

TABLE 4.7-1

STEAM LINE SAFETY VALVES

VALVE NUMBER	LIFT SETTING (± 1%) (psig)	ORIFICE SIZE (inches)
STEAM GENERATOR 3A		
Main steam line A1		
MSV - 34	1050	4.515
MSV - 38	1070	4.515
MSV - 43	1090	4.515
MSV - 40	1100	3.750
Main steam line A2		
MSV - 33	1050	4.515
MSV - 37	1070	4.515
MSV - 42	1090	4.515
MSV - 46	1100	4.515
STEAM GENERATOR 3B		
Main steam line B1		
MSV - 35	1050	4.515
MSV - 39	1070	4.515
MSV - 44	1090	4.515
MSV - 47	1100	4.515
Main steam line B2		
MSV - 36	1050	4.515
MSV - 41	1070	4.515
MSV - 45	1090	4.515
MSV - 48	1100	3.750

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

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

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

** Except when off-site power is not available and Reactor Coolant System pressure is less than 500 psig and Low Pressure Injection is not bypassed.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 130 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

INTRODUCTION

By letter dated February 13, 1990, Florida Power Corporation (FPC or the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). The proposed amendment would change the Technical Specifications (TS) associated with installation of the low pressure injection (LPI) actuation and the motor-driven emergency feedwater (EFW) pump block/trip feature. The block/trip feature is being installed in order to prevent electrical loads on the emergency diesel generators (EDGs) from being greater than the 30-minute rating during certain design basis accident scenarios which require LPI operation. These changes would result in elimination of the automatic, simultaneous operation of the motor-driven emergency feedwater pump (EFP-1) and the LPI system when offsite power is not available. This is accomplished by:

(a) eliminating the LPI pump start on 1500 psig, decreasing reactor coolant system (RCS) pressure;

(b) eliminating the LPI pump start on 4 psig reactor building pressure only when offsite power is unavailable; and,

(c) tripping or preventing the start of the motor-driven EFW pump when RCS pressure is below 500 psig and offsite power is unavailable.

The TS changes consist of (1) adding a note to TS Table 3.3-3 indicating that the reactor building pressure high actuation function of LPI is automatically bypassed when offsite power is not available, and (2) adding a note to the limiting condition for operation (LCO) 3.7.1.2.a indicating that the EFP-1 need not be operable when offsite power is not available and RCS pressure is less than 500 psig and the LPI is not bypassed. In addition, the licensee also proposed to increase the response times for the high pressure injection (HPI) and LPI in TS Table 3.3-5 from 25 to 35 seconds. This is to take credit for available margin between the time delay assumed in the safety analysis and the current TS limit.

The licensee has performed an evaluation of effects of the proposed TS changes. The staff evaluation of the proposed TS changes as well as licensee's evaluation follows.

EVALUATION

The LPI system, which operates independently of, and in addition to, the HPI system, is designed to maintain core cooling for large-break loss-of-coolant accidents (LOCAs). The EFW system is designed to provide a reliable source of feedwater to the steam generators to ensure a heat sink for decay heat removal during abnormal and accident conditions when the reactor is shut down.

The proposed TS change to bypass the reactor building pressure high (4 psig) actuation function of the LPI essentially eliminates possible actuation of LPI above 500 psig, which is the setpoint for the RCS pressure low-low initiation function. The change in LCO 3.7.1.2.a will allow a block or trip of EFP-1 when the RCS pressure is below 500 psig and the LPI pumps are running. Combined, these TS changes result in the LPI and EFW systems not operating at the same time. Therefore, the staff evaluation of these plant modifications and associated TS changes focuses on (1) the effect on RCS thermal hydraulic behavior to identify any accident scenario which requires simultaneous operation of the LPI and EFW systems for mitigation of the consequence so as to meet the acceptance criteria, and (2) the acceptability of deviating from established requirements for automatic initiation and operation of the EFW and LPI systems.

Thermal-Hydraulic Evaluation

Bypassing the actuation of the LPI system during accident conditions in which the RCS pressure is higher than 500 psig is acceptable because RCS pressure remains higher than the LPI pump shutoff head such that the LPI would play no role in the mitigation of the consequences of such accidents.

The LPI system operation is needed for mitigation of the consequences of large-break LOCAs and larger-sized small-break LOCAs that depressurize the RCS below 500 psig. The licensee's approved analysis of a large-break LOCA shows that the RCS depressurizes very rapidly and the reactor coolant is lost through the break at a rate greater than the HPI system capacity. The LPI is required to maintain the RCS inventory for decay heat removal. During the rapid depressurization, voiding occurs in the hot leg piping, resulting in a decoupling of the primary and secondary systems. This eliminates the ability to remove heat through the steam generators. Therefore, the secondary side heat sink is not an important factor with regard to preserving the peak cladding temperature (PCT) within the acceptance criterion and with regard to long-term cooling. Mitigation of the consequences is accomplished by the injection of the emergency core cooling system (ECCS) water. Therefore, for a large-break LOCA which requires the LPI operation for consequence mitigation, the EFW system has minimal effect on the performance of safety function and operation of the EFW pumps is not required.

Analysis of small-break LOCAs is documented in Topical Report BAW-1976A, "Small Break Loss of Coolant Accident Analysis for B&W 177 FA Lowered Loop Plants in Response to NUREG-0737, Item II.K.3.31." Depending on break sizes and consequent thermal hydraulic behavior, the small-break LOCAs are separated into four categories. The first three categories are smaller-sized breaks (less than 0.1 sq. ft.) where the RCS depressurization is slow and the RCS pressure does not fall below the secondary pressure. Since the RCS pressure is well above the shutoff head of the LPI pumps, bypassing the LPI actuation has no effect. For a larger-sized (greater than 0.1 sq. ft.) small-break LOCA (Category 4), the secondary heat sink is important only during the early stage of the event when the heat removal relies largely upon the heat transfer through the steam generators. As the accident progresses and the RCS depressurizes to an equilibrium condition with the secondary system, the steam generator heat transfer ceases. As depressurization continues, the RCS pressure falls below the secondary pressure and reverse heat transfer in the steam generator begins. Decay heat removal is then accomplished by safety injection and steam venting after the loop seal is cleared. Since the break is large enough to allow the RCS to depressurize to below the shutoff head of the LPI system, activation of LPI assures the core to be covered with a two-phase mixture during the long term and ensures adequate core cooling. Therefore, by the time the LPI system is actuated with RCS pressure below 500 psig, the steam generators as well as the EFW system have minimal effect. The small-break LOCA analysis in BAW-1976A indicated that the peak cladding temperature is less than 1100°F, well below the acceptance criterion of 2200°F. Therefore, there is ample margin to accommodate the very small effect from trip of the EFW system.

The accident scenarios which require the EFW system for decay heat removal are a loss of RCS flow, a loss of main feedwater, main feedline break, a steam line break, a small-break LOCA, and a steam generator tube rupture. The first three scenarios are RCS pressurization events where the LPI plays no role at all. For a steam line break or a smaller-sized small-break LOCA, the RCS pressure will decrease but remain above the LPI pump shutoff head, and therefore the LPI system has no effect. For a steam generator tube rupture event, the RCS will be depressurized to below the LPI shutoff head. However, the emergency procedure calls for a lockout of the safety injection actuation which will permit actuation of EFP-1.

To summarize, the staff could not identify an accident scenario which would require simultaneous operation of the LPI and EFW systems for mitigation of accident consequences. For large-break and larger-sized small-break LOCAs, which require LPI operation, the EFW system has no significant impact. In other accident scenarios which require the EFW system for decay heat removal, the LPI system operation is not required either because the RCS pressure is above the LPI shutoff head or the LPI is required to be in lockout. Therefore, the staff concludes that simultaneous operation of the LPI and EFW systems is not necessary and the relevant TS changes are acceptable.

Basis for Deviating from the Established Requirements for Automatic Initiation and Operation of LPI and EFW

The staff has reviewed the block/trip modifications and has concluded that this change adds complexity to the EFW and LPI actuation circuitry. The staff considers that this additional complexity produces some small reduction in the overall reliability of these systems. The overall reliability of the automatic initiation of the EFW and LPI systems is still expected to be high.

General Design Criterion (GDC) 20 specifies that the protection system shall be designed to initiate automatically the operation of appropriate systems and to sense accident conditions and to initiate the operation of systems and components important to safety. GDC 21 requires the protection function to be single-failure proof. IEEE Standard 279, to which the licensee has stated conformance in the CR-3 FSAR, requires that protection systems be so designed that, once initiated, a protective action shall go to completion. The TMI Action Plan in NUREG-0737, Section II.E.1.2, "Auxiliary Feedwater System Automatic Initiation and Flow Indication," requires that the EFW system design shall provide for the automatic initiation of the EFW system and that a single failure will not result in the loss of the EFW system function. The intent of the GDC and the IEEE Standard is to ensure that the systems and components important to safety can be automatically initiated and reliably operated and that no single failure would defeat their important safety function. The NUREG-0737 requirement was based on the consideration that an EFW system is needed for decay heat removal during accident conditions. Our evaluation has determined that the LPI system operation is required only at conditions with the RCS pressure below the LPI shutoff head, and that EFW system operation is not required at these conditions when the LPI is actuated. Therefore, delay in actuation of LPI until its function is important to safety, and block or trip of the EFW system when its function is not important to safety, do not contradict the intent of these requirements. On this basis, the staff concludes that the proposed TS changes and associated plant modifications to eliminate simultaneous operation of the LPI and EFW systems are acceptable.

Safety Injection System Response Time

The proposed TS change to increase the HPI and LPI response times from 25 to 35 seconds is consistent with the safety injection delay time assumed in the safety analyses. The large-break LOCA analysis performed for all B&W 177 fuel assembly plants (Topical Report BAW-10103A, Rev. 3) assumed that the HPI and LPI flows are available following a 35-second delay after reaching the actuation setpoints. The small-break LOCA analysis documented in BAW-1976A also assumed an HPI delay time of 35 seconds. For a steam line break, the HPI will inject cold borated ECCS water into the primary system. Since the steam line break analysis did not take credit for boration to reduce the reactivity increase from decrease of moderator temperature, and since the additional cooling from the cold ECCS water will aggravate the overcooling during the transient, an increase in the HPI response time to 35 seconds remains bounded by the current steam line break analysis.

Based on the above, the staff concludes that the proposed response time of 35 seconds is either consistent with the value used in the safety analysis or has no adverse effect, and is therefore acceptable.

SUMMARY

The staff has reviewed the proposed TS changes and associated plant modifications to prevent simultaneous operation of the LPI and EFW systems. Based on the evaluation of various accident scenarios which require either the LPI or the EFW system operation for consequence mitigation, the staff finds no accident conditions that would be adversely affected by the proposed TS changes and modifications. The staff has also determined that the changes and modifications do not contradict the intent of those rules and regulations related to the protection systems. On this basis, the staff concludes that the proposed TS changes and plant modifications are acceptable.

The staff also finds that the TS change to increase the safety injection response time from 25 to 35 seconds is consistent with the LOCA analysis assumption and is acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 4, 1990

Principal Contributors:
G. Hsii
J. Stewart