Proposed

Technical Specification

Changes

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	9#
6. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)*	2/Bus	2/Bus	1/Bus	1, 2, 3	12
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)					
(1) Undervoltage Device #1*	2/Bus	2/Bus	1/Bus	1, 2, 3	12
(2) Undervoltage Device #2*	2/Bus	2/Bus	1/Bus	1, 2, 3	12
c. 480 V Emergency Bus Undervoltage (Degraded Voltage)*	2/Bus	2/Bus	1/Bus	1, 2, 3	12
7. AUXILIARY FEEDWATER AUTOMATIC START Steam Generator (SG) Level Instruments	4/SG	2/SG[1/]	2/SG	1, 2, 3	11

DELETE

1/ 2/SG for either steam generator will start one train of AFW.

* This specification will be effective prior to Cycle 7 restart.

ADD NEXT PAGE

ADD TO TABLE 3.3-3

7. AUXILIARY FEEDWATER (AFAS)					
a. Manual (Trip Buttons)	4/SG	2/SG	4/SG	1, 2, 3	11
b. Automatic Actuation Logic	4/SG	2/SG	3/SG	1, 2, 3	8
c. SG Level (1A/1B) - Low	4/SG	2/SG	3/SG	1, 2, 3	13#, 14
8. AUXILIARY FEEDWATER ISOLATION					
a. SG 1A - SG 1B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	13#, 14
b. Feedwater Header SG 1A - SG 1B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	13#, 14

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 10 - 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 requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Sepcification 4.3.2.1.1.

ACTION 11 - Instrument operability requirements are contained in the Reactor Protection System requirements for Reactor Trip on Steam Generator Level. If an Automatic Start channel is inoperable, operation may continue provided that the affected pump is verified to be OPERABLE per Specification 4.7.1.2.a within 8 hours and at least once per 7 days thereafter; and the Automatic Start channel shall be restored to OPERABLE status within 30 days or the reactor shall be in at least HOT SHUTDOWN within the next 12 hours.

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ACTION 12 - 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.

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ACTION 11 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6m. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided that one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
6. LOSS OF POWER		
a. (1) 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)*	2900 ± 29 volts with a 1 ± .5 second time delay	2900 ± 29 volts with a 1 ± .5 second time delay
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)		
(1) Undervoltage Device #1*	3675 ± 36 volts with a 7 ± 1 minute time delay	3675 ± 36 volts with a 7 ± 1 minute time delay
(2) Undervoltage Device #2*	3592 ± 36 volts with a 18 ± 2 second time delay	3592 ± 36 volts with a 18 ± 2 second time delay
c. 480 volts Emergency Bus Undervoltage (Degraded Voltage)*	429 + 5-0 volts with a 7 ± 1 second time delay	429 + 5 -0 volts with a 7 ± 1 second time delay
7. AUXILIARY FEEDWATER	≥ 30% level	≥ 30% level

DELETE

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* This specification will be effective prior to Cycle 7 restart.

ADD TO TABLE 3.3-4

7. AUXILIARY FEEDWATER (AFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. SG 1A&1B Level Low	≥ 29.0%	≥ 28.5%
8. AUXILIARY FEEDWATER ISOLATION		
a. Steam Generator ΔP-High	≤ 275 psid	≤ 281 psid
b. Feedwater Header High ΔP	≤ 150.0 psid	≤ 157.5 psid

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Containment Fan Coolers	Not Applicable
Feedwater Isolation	Not Applicable
Containment Isolation	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIS	
Containment Isolation	Not Applicable
Shield Building Ventilation System	Not Applicable
d. RAS	
Containment Sump Recirculation	Not Applicable
e. MSIS	
Main Steam Isolation	Not Applicable
Feedwater Isolation	Not Applicable
ADD	
f. AFAS	
Auxiliary Feedwater Actuation	Not Applicable
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 30.0*/19.5**
b. Containment Isolation ***	$\leq$ 30.5*/20.5**
c. Containment Fan Coolers	$\leq$ 30.0*/17.0**
d. Feedwater Isolation	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ 30.0*/19.5**
b. Containment Isolation***	≤ 30.5*/20.5**
c. Shield Building Ventilation System	≤ 30.0*/14.0**
d. Containment Fan Coolers	≤ 30.0*/17.0**
e. Feedwater Isolation	≤ 60.0
4. Containment Pressure--High-High	
a. Containment Spray	≤ 30.0*/18.5**
5. Containment Radiation-High	
a. Containment Isolation***	≤ 30.5*/20.5**
b. Shield Building Ventilation System	≤ 30.0*/14.0**
6. Steam Generator Pressure-Low	
a. Main Steam Isolation	≤ 6.9
b. Feedwater Isolation	≤ 60.0
7. Refueling Water Storage Tank-Low	
a. Containment Sump Recirculation	≤ 91.5
ADD 8. Steam Generator Level -Low	
a. Auxiliary Feedwater	≥205.**≤600* (New Value)

TABLE NOTATION

*Diesel generator starting and sequence loading delays included.

**Diesel generator starting and sequence loading delays not included. Offsite power available.

***Not applicable to containment isolation valve I-HV-18-1.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. LOSS OF POWER				
a. 4.16 kv Emergency Bus Under-voltage (Loss of Voltage)*	S	R	M	1, 2, 3
b. 4.16 kv Emergency Bus Under-voltage (Degraded Voltage)				
(1) Undervoltage Device #1*	S	R	M	1, 2, 3
(2) Undervoltage Device #2*	S	R	M	1, 2, 3
b. 480 V Emergency Bus Under-voltage (Degraded Voltage)*	S	R	M	1, 2, 3
7. AUXILIARY FEEDWATER				
a. Auto Start	--------------(See Surveillance 4.7.1.2.b)-----------			
b. Steam Generator	------------------(See RPS Table 4.3-1)--------------			

DELETE

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* This specification will be effective prior to Cycle 7 restart.

ADD TO TABLE 4.3-2

7. AUXILIARY FEEDWATER (AFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1,2,3
b. SG Level (A/B) - Low	S	R	M	1,2,3
c. Automatic Actuation Logic	N.A.	N.A.	M	1,2,3
8. AUXILIARY FEEDWATER ISOLATION				
a. SG Level (A/B) - Low and SG Differential Pressure (BtoA/AtoB) - High	N.A.	R	M	1,2,3
b. SG Level (A/B) - Low and Feedwater Header Differential Pressure (BtoA/AtoB) - High	N.A.	R	M	1,2,3

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Manager of Power Resources Nuclear, the Vice President of Power Resources and to the Chairman of the Company Nuclear Review Board.

f. Review of those REPORTABLE OCCURRENCES requires 24 hour notification to the Commission.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Company Nuclear Review Board.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Company Nuclear Review Board.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Company Nuclear Review Board.

k. Review of every unplanned onsite release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President Nuclear Energy and to the Company Nuclear Review Board.

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL and RADWASTE TREATMENT SYSTEMS.

m. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last FRG meeting.

Add

AUTHORITY

6.5.1.7 The Facility Review Group shall:

a. Recommend to the Plant Manager written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President of Power Resources and the Company Nuclear Review Board of disagreement between the FRG and the Plant Manager; however, the Plant Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

ST. LUCIE UNIT NO. I
AUTOMATIC AUXILIARY FEEDWATER ACTUATION SYSTEM
SAFETY EVALUATION/NO SIGNIFICANT HAZARDS CONSIDERATIONS DETERMINATION

This is a request to revise Technical Specification 3/4.3.2, Engineered Safety Feature Actuation System (ESFAS) Instrumentation, of the Technical Specifications for St. Lucie Unit 1.

1.0 Description of Technical Specification Change

The proposed change contains those technical specifications needed to support the installation of the safety grade automatic auxiliary feedwater actuation system (AFAS), which will be installed to satisfy a requirement in NUREG-0737, Item II.E.1.2. The proposed change revises Table 3.3-3 (ESFAS Instrumentation), Table 3.3-4 (ESFAS Instrumentation Trip Values), Table 3.3-5 (Engineered Safety Features Response Times), and Table 4.3-2 (ESFAS Instrumentation Surveillance Requirements), and lists an additional responsibility for the Facility Review Group (FRG) in Technical Specification 6.5.1.6.

The change to Table 3.3-3 revises item 7, Auxiliary Feedwater, to include manual trip buttons, automatic actuation logic, and steam generator level as inputs to the AFAS. Item 8, Auxiliary Feedwater Isolation, was added to account for the systems ability to isolate a faulted steam generator based on high differential pressure between the steam generators and/or the feedwater headers. ACTION STATEMENTS were added to describe required action when the AFAS does not satisfy the channel requirements listed in Table 3.3-3.

The change to Table 3.3-4 revises item 7 and adds Item 8 to include all the inputs listed above with their applicable trip values and allowable values. Note that the change to the AFAS steam generator low level setpoint is based solely on the addition of new level transmitters with less instrument uncertainty than those presently installed. The input to safety analyses is unaffected.

The change to Table 3.3-5 adds item 1.f to include a manual AFAS initiating signal and revises item 8, Steam Generator Level, to reflect the new Auxiliary Feedwater (AFW) system response time based solely on new equipment instrument uncertainty as stated by the vendor ($\pm$25 sec).

The change to Table 4.3-2 revises item 7 and adds item 8 to include the inputs to the AFAS, as listed above, and lists the associated surveillance requirements for each input.

The change to Technical Specification 6.5.1.6 adds paragraph 6.5.1.6.m to the list of responsibilities of the Facility Review Group (FRG). It requires the FRG to review and document the judgement concerning prolonged operation in an abnormal configuration as allowed in ACTION STATEMENT 13 of Table 3.3-3.

1.1 Description of Change to the AFW System

The present AFW system includes automatic actuation of auxiliary feedwater on low steam generator level, after the expiration of a pre-set time delay. This system represents FPL's short-term commitment to NUREG-0737, Item II.E.1.2 The new AFAS which is to be installed prior to

start-up of Cycle 7, also includes automatic actuation of the AFW system on low steam generator level and an adjustable time delay. However, the AFAS also includes logic that will automatically isolate a faulted steam generator on low SG level coincident with high steam generator and/or high feedwater header differential pressure signals. Provisions are incorporated in the AFAS so that the actuation signal can be manually initiated.

The new AFAS will be as described and approved in the NRC's Safety Evaluation of the Auxiliary Feedwater System, Florida Power & Light Company, St. Lucie Plant, Unit No. 1, transmitted via NRC letter dated September 14, 1982 (R.A. Clark to R.E. Uhrig), except that a time delay has been added which will delay the actuation of the Auxiliary Feedwater (AFW) System for some pre-selected period of time (205 to 600 seconds) after receiving an initiation signal (low steam generator level setpoint).

If, however, the initiation signal is removed (due to increasing steam generator level above the bistable reset point) before the expiration of the time delay, the time delay is reset to zero. This stops the process and actuation of the AFW system will not occur unless the initiation signal is received again and the time delay expires before the initiation signal is removed.

The primary function of the time delay is to reduce challenges on the AFW system under the condition of reactor trip with offsite power and main feedwater available. The time delay also provides more favorable results for the Steam Line Break analysis by delaying the actuation of the AFW system.

2.0 Safety Analysis

The installation of the safety grade automatic auxiliary feedwater actuation system (AFAS) is being done, in part, to satisfy a requirement in NUREG-0737, Item II.E.1.2. An NRC safety evaluation of the AFAS (NRC letter to FPL, R. A. Clarke to R.E. Uhrig, dated September 14, 1982) stated that the electrical, instrumentation, and control aspects of the AFAS were reviewed by the NRC and that the NRC concluded that the St. Lucie, Unit No. 1 auxiliary feedwater automatic initiation system complies with the staff's long term safety grade requirements, and therefore, is acceptable. The safety evaluation recommended the addition of technical specifications addressing the surveillance testing of the AFAS automatic actuation logic. This proposed technical specification change package complies with those recommendations.

Automatic control of the Auxiliary Feedwater Actuation System will be modified to incorporate a time delay as described in Section 1.1, Description of Change to the AFW System. Two classes of events, increased heat removal and decreased heat removal, were re-evaluated to determine if they were affected by this change. The specific FSAR events re-evaluated, because the addition of the AFAS has the most impact of them, were the Loss of Feedwater event and the Steam Line Break post-trip Return-to-Criticality event. The existing analyses for the Loss of Feedwater event and the Steam Line Break post-trip Return-to-Criticality event are still valid, as described below.

2.1 Steam Line Break Post-Trip Return-to Criticality

The Steam Line Break post-trip Return-to-Criticality Analysis can be affected by changes in the assumptions for the auxiliary feedwater flow rate and the time and duration of delivery. More flow, delivered sooner, could adversely affect the results. The effects of the AFAS on the Steam

Line Break post-trip Return-to Criticality analysis was evaluated by Combustion Engineering, the designer of the AFAS by comparing the Cycle 5 analysis results with those obtained after changing only those assumptions impacted by AFAS. These assumptions are listed in the table below. A comparison between the new values and the reference values shows that the reference analysis is conservative, in that it assumes the same auxiliary feedwater flow, the same AFAS actuation setpoint, and the same minimum delay time. The only difference is that the new AFAS has the ability to automatically isolate the ruptured steam generator on high differential pressure ensuring that no additional water is admitted to the ruptured steam generator. This feature of the AFAS makes the Cycle 5 Steam Line Break analysis conservative.

Parameter	Ref. Analysis Without AFAS	New Analysis With AFAS	Comment
S/G Water Level AFW Actuation Setpoint	62.5% (Normal Water Level)	62.5% (Normal Water Level)	No Difference
Minimum AFAS Delay Time	180 seconds	180 seconds	No Difference
Maximum AFW Flow Rate	1280 gpm	1280 gpm	No Difference
Maximum High SG P AFW Isolation Setpoint	N/A	530 psid*	For HZP case-No impact. For Full Power Case Isolates AFW flow to ruptured S/G prior to AFAS actuation.

* Note that the 530 psid isolation setpoint represents the Technical Specification value plus instrument uncertainties in the conservative direction.

The high SG P AFW isolation setpoint was evaluated to determine if a hysteretic value was required to make the effects of re-initiating AFW flow to the ruptured steam generator acceptable. For this analysis the ruptured steam generator will be isolated when the SG P reaches 530 psid, and remains isolated until the feedwater header P falls below 200 psid. Since the feedwater header P does not fall below 200 psid until well after the time of peak reactivity, re-initiation of AFW is not of concern; therefore, a hysteretic value for the SG P AFW isolation setpoint is not required.

Exxon Nuclear Co. has performed their safety evaluation based on the current AFAS and has found the Cycle 7 analysis to be bounding.

Feedwater line breaks were not analyzed for excessive heat removal because the steam line break is the limiting cooldown event as stated in the FSAR.

2.2 Loss of Feedwater/Feedwater Line Breaks

Loss of Feedwater events, including Feedwater Line Breaks, were previously evaluated as a result of post TMI requirements (NUREG-0737,

Item II.E.1.1). This evaluation showed that the Loss of Feedwater event was limiting for the AFW system. This evaluation was submitted to the NRC and approved in the NRC's SE, transmitted via NRC letter dated September 14, 1982 (R. A. Clark to R. E. Uhrig). This evaluation is presently reflected in Section 10.5 of the FSAR. A re-evaluation of these events confirmed that the loss of main feedwater with loss off offsite power plus an active failure of the A or B battery and a high energy line break in the AFW system was the most limiting with regards to the AFW system requirements.

An evaluation of this event, plus the Loss of Feedwater analysis shown in Section 15.2.8 in the Final Safety Analysis Report (FSAR) demonstrates that, after the start of the events, at least ten minutes are available to verify auto-start or, if necessary to manually initiate auxiliary feedwater flow before steam generator dryout occurs. The new AFAS will actuate the AFW system automatically well within the time frame demonstrated as acceptable in the FSAR. Based on the above, the existing Loss of Feedwater Analysis in the FSAR remains bounding and no additional analysis is required.

3.0 Significant Hazards Consideration

On September 14, 1982, NRC approved the changes to the Auxiliary Feedwater (AFW) System at St. Lucie Unit 1, which were to be made to satisfy NUREG-0737, Section II.E.1.2. Because St. Lucie Unit 2 was being licensed at that time with the same auxiliary feedwater actuation system (AFAS) as what was approved for St. Lucie Unit 1, NRC allowed postponement of the implementation of AFAS on St. Lucie Unit 1 until the Fall 1985 refueling outage, in order to obtain operational experience with the system on Unit 2.

St. Lucie Unit 1 will be implementing the new AFAS during the upcoming refueling outage, scheduled to begin on October 20, 1985. The only difference between the system approved by NRC and the system being installed is the addition of a time delay. The addition of the time delay on the new system performs the same function as the currently installed time delay, i.e., it delays actuation of the system for a pre-selected period of time after receiving an actuation signal (low steam generator level setpoint).

The proposed changes to the technical specifications are necessary to meet NUREG-0737 technical specification recommendations in that additional surveillance requirements are now specified for the AFW system. This is similar to Example (ii) of the examples of amendments that are considered not likely to involve significant hazards considerations, contained in the Commission's guidance for determination of significant hazards considerations, in that the new surveillance requirements are changes that constitute additional limitations, restrictions, or controls not presently included in the technical specifications: for example, a more stringent surveillance requirement.

The proposed change to the steam generator low level setpoint for AFW system initiation (Table 3.3-4 Item 7) is based solely on the addition of new level transmitters with less instrument uncertainty than those presently installed. The input to the safety analyses is unaffected.

The proposed change to the Steam Generator Level-Low response time for AFW System initiation (Table 3.3-5 Item 8) is also based solely on the new equipment instrument uncertainty as stated by the vendor.

The proposed change to Technical Specification 6.5.1.6 adds paragraph 6.5.1.6.m under the Facility Review Group responsibilities to be consistent with the addition of new ACTION STATEMENT 13 of Table 3.3-3.

These last three changes are similar to Example (i) of the examples of amendments that are considered not likely to involve significant hazards considerations, in that they are administrative in nature to achieve consistency throughout the technical specifications and to change the technical specifications to be consistent with the installed instrumentation.

Based on the safety evaluation and discussions above, it has been determined that the proposed changes do not involve significant hazards considerations.