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November 21, 2000

U.S. Nuclear Regulatory Commission
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ULNRC-04346

Gentlemen:

DOCKET NUMBER 50-483
UNION ELECTRIC COMPANY
CALLAWAY PLANT
REVISION TO TECHNICAL SPECIFICATION 3.3.2,
"ESFAS INSTRUMENTATION"

Union Electric Company herewith transmits an application for amendment to Facility Operating License No. NPF-30 for the Callaway Plant.

This amendment application would add Surveillance Requirement (SR) 3.3.2.10 to Functions 6.f and 6.h of Technical Specification (TS) Table 3.3.2-1. SR 3.3.2.10 requires that ESF RESPONSE TIMES be verified to be within limits every 18 months on a STAGGERED TEST BASIS. Function 6.f covers the start of the turbine-driven auxiliary feedwater (AFW) pump on a loss of offsite power. Function 6.h covers the AFW pump suction transfer from the condensate storage tank (CST) to the essential service water (ESW) system on low suction pressure.

In accordance with NRC Administrative Letter 98-10, administrative controls were implemented soon after the discovery of both of these issues involving insufficient Technical Specifications. In both cases, only a portion of the circuitry had not been response time tested. The response times of those portions of the affected functions were either determined from a review of plant computer data or were measured during supplemental testing.

As required by SR 3.3.2.10, the turbine-driven AFW pump spin-up time is measured every 18 months on a STAGGERED TEST BASIS. This test measures the time required for the turbine-driven AFW pump to start, including required valve strokes, and reach the required discharge pressure after slave relay K641 energizes when a steam generator water level low-low signal is initiated. The response time of the undervoltage relay that detects the loss of offsite power is measured in accordance with SR 3.3.5.4. The only portion of Function 6.f that was not previously response time tested is the BOP-ESFAS separation group 2 actuation circuitry. A review of plant computer data taken during Refuel 10 (October-November 1999) ESFAS testing verified that the turbine-driven AFW

A001

pump start signal occurred in less than 2 seconds after the main feeder breaker opened upon a simulated loss of offsite power. After adding this value to the measured response times for the rest of the equipment required for Function 6.f, a total response time less than the required limit of 60 seconds was obtained. This review was completed before the Completion Time of LCO 3.3.2 Required Action R.1 expired.

The portion of Function 6.h that was not previously response time tested extends from the pressure transmitters that detect low suction pressure from the CST through the BOP-ESFAS separation group 1 and 4 actuation bistables. The response time of this portion of Function 6.h was measured during supplemental testing performed within 24 hours of discovery. After adding this value to the measured response times for the rest of the equipment required for Function 6.h, a total response time less than the required limit of 60 seconds was obtained.

The Callaway Plant Onsite Review Committee and the Nuclear Safety Review Board have reviewed this amendment application. Attachments 1 through 4 provide the Description and Assessment, Markup of Technical Specifications, Retyped Technical Specifications, and Draft FSAR Changes, respectively, in support of this amendment request. There are no Bases Changes associated with this amendment application. Attachment 4 FSAR mark-ups are provided for information only. It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10CFR50.92. Pursuant to 10CFR51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Approval of this amendment application is requested by March 1, 2001. The amendment will be fully implemented prior to entering MODE 3 from MODE 4 during startup from Refuel 11.

If you have any questions on this amendment application, please contact us.

Very truly yours,

David Shafer

for Alan C. Passwater
Manager-Corporate Nuclear Services

Attachments:

1 - Description and Assessment
2 - Markup of Technical Specifications
3 - Retyped Technical Specifications
4 - Draft FSAR Changes

STATE OF MISSOURI)
) S S
CITY OF ST. LOUIS)

David Shafer, of lawful age, being first duly sworn upon oath says that he is Supervising Engineer, Corporate Nuclear Services, Regulatory Operations for Union Electric Company; that he has read the foregoing document and knows the content thereof; that he has executed the same for and on behalf of said company with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By David Shafer
David Shafer
Supervising Engineer,
Regulatory Operations,
Corporate Nuclear Services

SUBSCRIBED and sworn to before me this 21st day of November, 2000.

Debby Anzalone

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ATTACHMENT ONE

DESCRIPTION AND ASSESSMENT

DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

1.1 This amendment application would revise Technical Specification (TS) 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," by adding Surveillance Requirement (SR) 3.3.2.10 to Functions 6.f and 6.h of TS Table 3.3.2-1.

1.2 Final Safety Analysis Report (FSAR) Section

Changes to the FSAR will also be required after this amendment application is approved. Changes to FSAR Chapter 15 are required to reflect auxiliary feedwater (AFW) assumptions related to the analysis of the small break loss of coolant accident (LOCA). Changes to FSAR Table 16.3-2 are required to add the 60 second response time limit that will establish the acceptance criteria against which SR 3.3.2.10 will be based for Functions 6.f and 6.h in TS Table 3.3.2-1.

2.0 DESCRIPTION

This amendment application would add SR 3.3.2.10 to Functions 6.f and 6.h of TS Table 3.3.2-1. SR 3.3.2.10 requires that ESF RESPONSE TIMES be verified to be within limits every 18 months on a STAGGERED TEST BASIS. Function 6.f covers the start of the turbine-driven AFW pump on a loss of offsite power. Function 6.h covers the AFW pump suction transfer from the condensate storage tank (CST) to the essential service water (ESW) system on low suction pressure.

Attachment 2 provides the TS markups.

3.0 BACKGROUND

SR 3.3.2.10 requires a verification that ESF RESPONSE TIMES are within their limits every 18 months on a STAGGERED TEST BASIS, as defined in the TS. As discussed in the SR 3.3.2.10 Bases, FSAR Table 16.3-2 establishes the acceptance criteria time limits for the response time tests. These limits are less

than or equal to the maximum values assumed in the accident analyses. However, the SR 3.3.2.10 Bases also states: "No credit was taken in the safety analyses for those channels with response times listed as N.A. No response time testing requirements apply where N.A. is listed in Reference 9." Reference 9 is FSAR Table 16.3-2, which lists the associated response times as N.A. for items 10 (Loss of Offsite Power - Start Turbine-Driven Auxiliary Feedwater Pump) and 12 (Auxiliary Feedwater Pump Suction Pressure Low - Transfer to Essential Service Water). As a result, the current Callaway Response Time Testing Program does not verify the response times for the start of the turbine-driven AFW pump on a loss of offsite power signal or the transfer of the AFW pump suction from the CST to ESW.

Prior to initial licensing of Callaway in 1984, Westinghouse confirmed that neither the start of the turbine-driven AFW pump on a loss of offsite power signal nor the transfer of the AFW pump suction from the CST to ESW was assumed in any Westinghouse analyses. Since Westinghouse had performed all the licensing basis FSAR Chapter 15 accident analyses, it was appropriate to list "N.A." for the response time of these functions in the original licensing basis version of TS Table 3.3-5, which was subsequently relocated to the FSAR as Table 16.3-2 in OL Amendment 104 (November 22, 1995) in accordance with NRC Generic Letter 93-08, "Relocation of Technical Specification Tables of Instrument Response Time Limits."

Small Break LOCA Analysis

The original licensing basis Callaway small break LOCA analysis was performed by Westinghouse utilizing the WFLASH Evaluation Model. That Evaluation Model did not credit the turbine-driven AFW pump; only the motor-driven AFW pumps were modeled.

The current Callaway small break LOCA analysis of record was performed by Westinghouse utilizing the NOTRUMP Evaluation Model. This analysis was last performed in 1987 in support of the Cycle 3 reload that first used VANTAGE 5 fuel assemblies at Callaway. In the NOTRUMP Evaluation Model the turbine-driven AFW pump is modeled, instead of the motor-driven AFW pumps, to remove potential asymmetric effects (i.e., flow from the turbine-driven AFW pump is delivered to all four steam generators). The current Callaway small break LOCA analysis models AFW delivery from the turbine-driven AFW pump and credits its actuation on a loss of offsite power signal. A delay of not more than 60 seconds has been modeled in the analysis to account for the time between the AFW actuation signal and the time at which the turbine-driven AFW pump is at full flow. The response time of less than or equal to 60 seconds accounts for the

delays associated with the delivery of flow from the turbine-driven AFW pump for mitigation of a small break LOCA. A loss of power, assumed to occur coincident with the reactor trip signal, is sensed and processed by the circuitry for loss of power DG start covered by TS LCO 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation." The 4.16 kV ESF bus undervoltage relays provide inputs to the load shedder and emergency load sequencer (LSELS) cabinets, which in turn provide a signal to BOP-ESFAS cabinets via LSELS output relays. The BOP-ESFAS separation group 2 cabinet starts the turbine-driven AFW pump. The response time includes all signal actuation delays as well as valve stroke and turbine-driven AFW pump spin-up times.

In 1987, neither Westinghouse nor AmerenUE recognized the impact on TS Table 3.3-5 (currently FSAR Table 16.3-2), item 10, as a result of the modeling of the turbine-driven AFW pump flow in the revised Callaway small break LOCA analysis.

Non-LOCA Transient Analyses

Furthermore, every non-LOCA transient analysis in FSAR Chapter 15 that credits AFW pump start within 60 seconds after the associated ESF actuation signal is generated (e.g., inadvertent opening of a steam generator relief or safety valve, steam system piping failure, loss of non-emergency AC power, loss of normal feedwater, feedwater system pipe break, and steam generator tube failure) must assume that the safety grade suction source, provided by the ESW system, is credited. The preferred suction source, the CST, is not designed to withstand a design basis tornado or safe shutdown earthquake. The rationale for not crediting the CST is consistent with the Callaway licensing basis. The license amendment and amendment request cited in Reference 2 provide further background discussion.

The response time of less than or equal to 60 seconds accounts for the delays associated with the AFW pump suction transfer from the CST to ESW on low suction pressure. The response time includes all signal actuation delays, DG starting and sequencer loading delays, valve stroke times, and ESW pump spin-up time.

As such the Response Time Testing Program, that ensures the Chapter 15 analyses are met, must be revised. The response times included in FSAR Table 16.3-2, items 10 and 12, must be revised from N.A. to $\leq$ 60 seconds.

4.0 TECHNICAL ANALYSIS

Design Basis Function

The AFW system automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The system has two motor-driven pumps and a turbine-driven pump. The AFW pumps normally take suction through a common suction line from the preferred source of water, the CST. A low pressure signal in the AFW pump suction line protects the AFW pumps against a loss of the normal supply of water from the CST. Three pressure switches are located in the AFW pump suction line from the CST. A low pressure signal sensed by any two of the three switches coincident with an auxiliary feedwater actuation signal will automatically realign the AFW pump suctions to the safety-related ESW system. The CST capacity allows the plant to remove decay heat from the primary system during a 4-hour station blackout event, as discussed in FSAR Appendix 8.3A; however, the CST is not the safety-related source of water to the AFW pumps. The CST is the preferred suction supply to the AFW pumps due to the quality of the water. However, the CST is a non-seismic structure that can not be credited in FSAR Chapter 15 analyses. The safety-related source of water to the AFW pumps is provided by the ESW system.

Loss of power to either ESF bus will start the turbine-driven AFW pump to ensure the steam generators contain sufficient inventory to serve as the heat sink for reactor decay heat and sensible heat removal. A loss of offsite power (LOP) is indicated by the undervoltage relays detecting a loss of voltage on each EFS bus, as discussed in the Bases for TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," Function 6.f, "Auxiliary Feedwater - Loss of Offsite Power," and in TS 3.3.5, "Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation." Upon satisfying the two-out-of-four loss of voltage logic in the LSELS cabinets, LSELS output relays feed the redundant separation groups 1 and 4 BOP-ESFAS cabinets. The separation groups 1 and 4 BOP-ESFAS cabinets perform several functions, but on a LOP signal these cabinets perform only two functions, i.e., the steam generator blowdown and sample lines are isolated and electrically isolated outputs are provided to the separation group 2 BOP-ESFAS cabinet. The separation group 2 BOP-ESFAS cabinet is only associated with the turbine-driven AFW pump and is considered to be part of that end device, i.e., the turbine-driven AFW pump, in the TS. On a LOP signal, the turbine steam supply valves (ABHV0005 and ABHV0006) and the turbine trip and throttle valve (FCHV0312) are opened for the start of the turbine-driven AFW pump.

Impact on Nuclear Safety

Adding SR 3.3.2.10 to Functions 6.f and 6.h of TS Table 3.3.2-1 will ensure that accident analysis assumptions regarding the response time of equipment credited in accident mitigation are verified on a periodic basis.

Probabilistic Safety Assessment (PSA) Evaluation

There is no impact on the Callaway PSA since that study is concerned mainly with equipment functionality, not response time. The functional performance of the AFW and ESW systems is demonstrated by the Surveillance Requirements of TS 3.7.5, "Auxiliary Feedwater (AFW) System," and TS 3.7.8, "Essential Service Water (ESW) System." Functional capabilities of the instrumentation are demonstrated by the Surveillance Requirements of TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," in Table 3.3.2-1 for Functions 6.f and 6.h as well as the Surveillance Requirements of TS 3.3.5, " Loss of Power (LOP) Diesel Generator (DG) Start Instrumentation," which feeds a signal to the BOP-ESFAS by relay actuation for starting the turbine-driven AFW pump. Equipment response time is inherent in the PSA assumption that the equipment will perform its intended function. Adding SR 3.3.2.10 to Functions 6.f and 6.h of TS Table 3.3.2-1 will ensure the equipment performs its intended function.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

The proposed change involves modifying the TS to impose more stringent surveillance testing requirements. The proposed change does not involve a significant hazards consideration because operation of Callaway Plant in accordance with this change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since there are no hardware changes. The Reactor Trip System (RTS) and Engineered Safety Feature Actuation System (ESFAS) instrumentation will be unaffected. These protection systems will continue to function in a manner consistent with the plant design basis. All design, material, and construction standards that were applicable prior to the

request are maintained.

The proposed change imposes more stringent surveillance testing requirements to ensure safety-related structures, systems, and components are tested in a manner consistent with the safety analysis and licensing basis.

The proposed change will not affect the probability of any event initiators. There will be no degradation in the performance of, or an increase in the number of challenges imposed on, safety-related equipment assumed to function during an accident situation. There will be no change to normal plant operating parameters or accident mitigation performance.

The proposed change will not alter any assumptions or change any mitigation actions in the radiological consequence evaluations in the FSAR.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

There are no hardware changes nor are there any changes in the method by which any safety-related plant system performs its safety function. This change will not affect the normal method of plant operation or change any operating parameters. No performance requirements will be affected; however, the proposed change does impose additional surveillance testing requirements. These additional requirements are consistent with assumptions made in the safety analysis and licensing basis.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of this change. There will be no adverse effect or challenges imposed on any safety-related system as a result of this change.

This change does not alter the design or performance of the 7300 Process Protection System, Nuclear Instrumentation System, or Solid State Protection System used in the plant protection systems.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

(3) Involve a significant reduction in a margin or safety.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, departure from nucleate boiling ratio (DNBR) limits, heat flux hot channel factor (F_Q), nuclear enthalpy rise hot channel factor ($F\Delta H$), loss of coolant accident peak cladding temperature (LOCA PCT), peak local power density, or any other margin of safety. The radiological dose consequence acceptance criteria listed in the Standard Review Plan will continue to be met.

The imposition of more stringent surveillance testing requirements increases the margin of safety by ensuring that the affected safety analysis assumptions on equipment response time are verified on a periodic frequency.

Therefore, the proposed change does not involve a significant reduction in any margin of safety.

Conclusion:

Based on the preceding information, it has been determined that the proposed change meets the requirements of 10CFR50.92(c) and does not involve a significant hazards consideration.

5.2 Regulatory Safety Analysis

Applicable Regulatory Requirements/Criteria

The regulatory bases and guidance documents associated with the systems discussed in this amendment application include:

GDC-13 requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

GDC-20 requires that the protection system(s) shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control

systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

GDC-21 requires that the protection system(s) shall be designed for high functional reliability and testability.

GDC-22 through GDC-25 and GDC-29 require various design attributes for the protection system(s), including independence, safe failure modes, separation from control systems, requirements for reactivity control malfunctions, and protection against anticipated operational occurrences.

GDC-34 requires that a system to remove residual heat shall be provided. When the plant is above the RHR cut-in conditions, the secondary side of the plant including the AFW system removes residual heat.

GDC-44 requires a cooling water system(s) be provided to transfer heat to an ultimate heat sink. GDC-45 and GDC-46 require the cooling water system(s) be designed to permit inspection and testing. These criteria apply to the ESW system at Callaway.

Regulatory Guide 1.22 discusses an acceptable method of satisfying GDC-20 and GDC-21 regarding the periodic testing of protection system actuation functions. These periodic tests should duplicate, as closely as practicable, the performance that is required of the actuation devices in the event of an accident.

10CFR50.55a(h) requires that the Callaway protection systems, including the ESFAS Functions 6.f and 6.h, meet IEEE 279-1971. Sections 4.9 - 4.11 of IEEE 279-1971 discuss testing provisions for protection systems. Regulatory Guide 1.118, Revision 2, discusses acceptable methods for testing protection systems, including Section 6.3.4 of IEEE 338-1977 for response time testing.

Analysis

There have been no changes to the ESFAS instrumentation or AFW/ESW system design such that any of the above regulatory requirements and criteria would not be met. This amendment application imposes additional surveillance testing requirements on the ESFAS Functions 6.f and 6.h consistent with the above requirements.

Conclusion

The evaluation performed by Union Electric Company concludes that Callaway Plant continues to comply with the above regulatory requirements.

6.0 ENVIRONMENTAL EVALUATION

Union Electric Company has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10CFR20, or would change an inspection or surveillance requirement. Union Electric Company has evaluated the proposed change and has determined that the change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10CFR51.22 (c)(9). Therefore, pursuant to 10CFR51.22 (b), an environmental assessment of the proposed change is not required.

7.0 REFERENCES

1. ULNRC-04308 dated September 8, 2000
2. OL Amendment No. 55 dated July 27, 1990 and ULNRC-02167 dated March 6, 1990

ATTACHMENT TWO

MARKUP OF TECHNICAL SPECIFICATIONS

Table 3.3.2-1 (page 7 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
6. Auxiliary Feedwater					
d. SG Water Level Low-Low (continued)					
(3) Vessel ΔT Equivalent including delay timers - Trip Time Delay					
(a) Vessel ΔT (Power-1)	1,2	4	M	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ Vessel ΔT Equivalent to 13.9% RTP[k]
(b) Vessel ΔT (Power-2)	1,2	4	M	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ Vessel ΔT Equivalent to 23.9% RTP[l]
(4) Containment Pressure - Environmental Allowance Modifier	1, 2, 3	4	N	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 2.0 psig
e. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
f. Loss of Offsite Power	1,2,3	2 trains	R	SR 3.3.2.7 SR 3.3.2.10	NA
g. Trip of all Main Feedwater Pumps	1,2[n]	2 per pump	J	SR 3.3.2.8	NA

(continued)

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.
(k) With a time delay ≤ 240 seconds.
(l) With a time delay ≤ 130 seconds.
(n) Trip function may be blocked just before shutdown of the last operating main feedwater pump and restored just after the first main feedwater pump is put into service following performance of its startup trip test.

Table 3.3.2-1 (page 8 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
6. Auxiliary Feedwater (continued)					
h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	3	O	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.10 → SR 3.3.2.12	≥ 20.64 psia
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.13	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 35.2%
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4	1,2,3	2 per train, 2 trains	F	SR 3.3.2.11	NA
b. Pressurizer Pressure, P-11	1,2,3	3	L	SR 3.3.2.5 SR 3.3.2.9	≤ 1981 psig
9. Automatic Pressurizer PORV Actuation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.14	NA
b. Pressurizer Pressure – High	1,2,3	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≤2350 psig

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.

ATTACHMENT THREE

RETYPED TECHNICAL SPECIFICATIONS

Table 3.3.2-1 (page 7 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
6. Auxiliary Feedwater					
d. SG Water Level Low-Low (continued)					
(3) Vessel ΔT Equivalent including delay timers - Trip Time Delay					
(a) Vessel ΔT (Power-1)	1,2	4	M	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ Vessel ΔT Equivalent to 13.9% RTP[k]
(b) Vessel ΔT (Power-2)	1,2	4	M	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ Vessel ΔT Equivalent to 23.9% RTP[l]
(4) Containment Pressure - Environmental Allowance Modifier	1, 2, 3	4	N	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 2.0 psig
e. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
f. Loss of Offsite Power	1,2,3	2 trains	R	SR 3.3.2.7 SR 3.3.2.10	NA
g. Trip of all Main Feedwater Pumps	1,2[n]	2 per pump	J	SR 3.3.2.8	NA

(continued)

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.
(k) With a time delay ≤ 240 seconds.
(l) With a time delay ≤ 130 seconds.
(n) Trip function may be blocked just before shutdown of the last operating main feedwater pump and restored just after the first main feedwater pump is put into service following performance of its startup trip test.

Table 3.3.2-1 (page 8 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE[a]
6. Auxiliary Feedwater (continued)					
h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	3	O	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.10 SR 3.3.2.12	≥ 20.64 psia
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.13	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 35.2%
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4	1,2,3	2 per train, 2 trains	F	SR 3.3.2.11	NA
b. Pressurizer Pressure, P-11	1,2,3	3	L	SR 3.3.2.5 SR 3.3.2.9	≤ 1981 psig
9. Automatic Pressurizer PORV Actuation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.14	NA
b. Pressurizer Pressure – High	1,2,3	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≤2350 psig

(a) The Allowable Value defines the limiting safety system setting. See the Bases for the Trip Setpoints.

ATTACHMENT FOUR

DRAFT FSAR CHANGES

TABLE 15.0-6 (Sheet 5)

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment/Alarms	ESF Equipment
Loss of coolant accidents resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary	Low pressurizer pressure (1), SIS, OTΔT, manual	SIS on Hi-1 containment pressure or low pressurizer pressure (1), AFAS and FWIS on SIS (modeled only in SBLOCA analysis) (1)	Essential service water system, component cooling water system, steam generator safety and/or relief valves, feedwater isolation valves (modeled for SBLOCA only)	Emergency core cooling system, TD auxiliary feedwater pump (modeled for SBLOCA only), containment heat removal system, emergency power system

on loss of offsite power (1)

Notes:

(1) Trip function credited in the safety analysis. Some analyses credit different trip functions depending on different analysis cases presented.

(2) Turbine trip on high-high steam generator water level not directly credited in the Section 15.1.2 analysis.

(3) No automatic protection action required.

(4) Trip function specifically excluded from potentially available functions to drive analysis in a given direction.

FYI - No change (see next page)

further reduce containment pressure. Approximately 13 hours after initiation of the LOCA, the ECCS is realigned to supply water to the RCS hot legs in order to control the boric acid concentration in the reactor vessel.

Description of Small-Break LOCA Transient

Ruptures of small cross sections will cause expulsion of the coolant at a rate which can be accommodated by the charging pumps. These pumps would maintain an operational water level in the pressurizer permitting the operator to execute an orderly shutdown. The coolant, which would be released to the containment, contains the fission products existing at equilibrium.

The maximum break size for which the normal makeup system can maintain the pressurizer level is obtained by comparing the calculated flow from the RCS through the postulated break against the charging pump makeup flow at normal RCS pressure, i.e., 2,250 psia. A makeup flow rate from one charging pump is adequate to sustain a pressurizer level corresponding to a pressure of 2,250 psia for a break through a 0.375-inch-diameter hole. This break results in a loss of approximately 17.25 lb/sec.

Should a larger break occur, depressurization of the RCS causes fluid to flow into the loops from the pressurizer resulting in a pressure and level decrease in the pressurizer. Reactor trip occurs when the low pressurizer pressure trip setpoint is reached. During the earlier part of the small break transient, the effect of the break flow is not strong enough to overcome the flow maintained by the reactor coolant pumps through the core as they are coasting down following reactor trip. Therefore, upward flow through the core is maintained. The ECCS is actuated when the appropriate setpoint is reached. The consequences of the accident are limited in two ways:

a. Reactor trip and borated water injection complement void formation in the core and cause a rapid reduction of nuclear power to a residual level corresponding to the delayed fission and fission product decay heat.

b. Injection of borated water ensures sufficient flooding of the core to prevent excessive clad temperatures.

Before the break occurs the plant is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. During blowdown, heat from fission product decay, hot internals, and the vessel continues to be transferred to the RCS. The heat transfer between the RCS and the secondary system may be in either direction, depending on the relative temperatures. In the case of continued heat addition to the secondary, secondary system pressure increases and steam relief

via the atmospheric relief and/or safety valves may occur. Makeup to the secondary side is automatically provided by the turbine-driven auxiliary feedwater pump~~s~~. The safety injection signal isolates normal feedwater flow by closing the main feedwater isolation valves. ~~and initiates auxiliary feedwater flow by starting the auxiliary feedwater pumps.~~ The secondary flow aids in the reduction of Reactor Coolant System pressure. Only the turbine-driven AFW pump is modeled in the analysis to remove potential asymmetric effects. A loss of offsite power signal initiates the turbine-driven AFW pump.
When the RCS depressurizes to approximately 600 psia, the cold leg accumulators begin to inject borated water into the reactor coolant loops. The vessel mixture level starts to move up to cover the fuel before the accumulator injection for most breaks. For all breaks, the accumulator injection provides enough water supply to bring the mixture level up to the upper plenum region where it is maintained. Due to the loss of offsite power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor trip during the accident and the effects of pump coastdown are included in the blowdown analyses.

15.6.5.3 Analysis of Effects and Consequences

15.6.5.3.1 Method of Analysis

The requirements of an acceptable ECCS evaluation model are presented in Appendix K of 10 CFR 50 (Ref. 5). The requirements of Appendix K regarding specific model features were met by selecting models which provide a significant overall conservatism in the analysis. The assumptions made pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA occurs and include such items as the core peaking factors, the containment pressure, and the performance of the ECCS. Decay heat generated throughout the transient is also conservatively calculated, as required by Appendix K of 10 CFR 50.

Large-Break LOCA Evaluation Model

The analysis of a large-break LOCA transient is divided into three phases: 1) blowdown, 2) refill, and 3) reflood. There are three distinct transients analyzed in each phase: 1) thermal-hydraulic transient in the RCS, 2) the pressure and temperature transient within the containment, and 3) the fuel and clad temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

The description of the various aspects of the LOCA analysis methodology is given in References 6, 11, 12, 17, 18, 22 and 23. These documents describe the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Acceptance Criteria. The SATAN-VI, WREFLOOD, COCO, and LOCTA-IV codes which are used

TABLE 16.3-2 (Continued)
ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
9. Steam Generator Water Level-Low-Low	
a. Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60[8]
b. Start Turbine-Driven Auxiliary Feedwater Pump	≤ 60[8]
c. Feedwater Isolation	≤ 2[5][8]
10. Loss-of-Offsite Power	
Start Turbine-Driven Auxiliary Feedwater Pump	~~N.A.~~ ≤ 60[9]
11. Trip of All Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	N.A.
12. Auxiliary Feedwater Pump Suction Pressure-Low	
Transfer to Essential Service Water	~~N.A.~~ ≤ 60[1]
13. RWST Level-Low-Low Coincident with Safety Injection	
Automatic Switchover to Containment Sump	≤ 60
14. Loss of Power	
a. 4 kV Bus Undervoltage-Loss of Voltage	≤ 14
b. 4 kV Bus Undervoltage-Grid Degraded Voltage	≤ 144
15. Phase "A" Isolation	
a. Control Room Isolation	N.A.
b. Containment Purge Isolation	≤ 2[5]

TABLE 16.3-2 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

TABLE NOTATIONS

(1) Signal actuation, Diesel generator starting, and sequencer loading delays included. Valve stroke times and spin-up times for pumps and fans included, as applicable.

(2) Diesel generator starting delay not included. Offsite power available.

(3) Diesel generator starting and sequence loading delay included. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(4) Diesel generator starting and sequence loading delays not included. Offsite power available. RHR pumps not included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(5) Does not include valve closure time.

(6) Includes time for diesel to reach full speed.

(7) Diesel generator starting and sequence loading delays included. Sequential transfer of charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included. Response time assumes only opening of RWST valves.

(8) Does not include Trip Time Delays. Response times noted above include the transmitters, 7300 process protection cabinets, solid state protection cabinets, and actuation devices only. This reflects the response times necessary for THERMAL POWER in excess of 22.41% RATED THERMAL POWER.

(9) Includes signal actuation, valve stroke, and pump spin-up times.