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POWER
SYSTEMS

Docket No. STN 50-470F

May 31, 1985
LD-85-028

Hugh L. Thompson, Director
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Additional CESSAR FSAR Clarifications

Dear Mr. Thompson:

Please find the enclosed CESSAR FSAR clarifications. These changes were identified by C-E and/or the NRC, as part of a cleanup effort to close out Amendment 10, and have already been informally provided to the Staff. These changes are not considered to be significant and should require minimal NRC review. A description of each change is provided in the attachment and the affected CESSAR-F pages are enclosed.

If you have any questions or comments concerning these changes, please contact me or Mr. G. A. Davis of my staff at (203) 285-5207.

Very truly truly yours,

COMBUSTION ENGINEERING, INC

A. E. Scherer
Director
Nuclear Licensing

AES:bks
Attachment
cc: P. Moriette

E003
1/1

ATTACHMENT TO LD-85-028

DESCRIPTION OF CESSAR CHANGES

The changes indicated on CESSAR-F pages 4.3-5, 4.3-8, Table 6.3.2-1 and Table 9.3-6 correct typographical errors and do not indicate changes in any methods or analysis results.

The CESSAR text describing the Safety Injection System (SIS) delay time is clarified to avoid any confusion. CESSAR Section 6.3 and Table 8.3.1-4 are revised to provide a consistent description of the delay time for the SIS.

The change to Table 7.5-3 was made because the installed transmitter is now Class 1E.

The change to page 9.1-17 of Section 9.1 clarifies a misleading statement on the dose rate in the spent fuel pool area. This correction is not significant, since monitoring of personnel in the spent fuel pool area ensures that all dose limitations and ALARA considerations are met.

The editorial changes to pages 10.4-2 and B-6u provide additional clarification.

analyses constituted one limiting combination of parameters for full power operation in the first cycle. Other combinations of parameters which will result in acceptable consequences of the safety analysis do exist, e.g., a higher $F^n q$ is acceptable at a reduced power level. Implementation in the technical specification is via an operating limit on the monitored peak linear heat generation rate.

B. The thermal margin to a minimum DNBR of 1.19 (using the C-E-1 CHF correlation as discussed in Section 4.4.2.2. and 4.4.4.1), which is available to accommodate anticipated operational occurrences, is a function of several parameters, including

i) the coolant conditions

ii) the axial power distribution

iii) the axially integrated radial peaking factor, $F^n r$; where $F^n r$ is the rod radial nuclear factor or the rod radial peaking factor and is defined in Section 4.4.2.2.2.1, Paragraph A (referred to as F_r in that section).

The coolant conditions assumed in the safety analyses, an $F^n r$ of 1.55, and the set of axial shapes displayed in Figure 4.4-3 constitute a set of limiting combinations of parameters for full power operation. Other combinations giving acceptable accident analysis consequences are equally acceptable. Implementation of these limits in the technical specification is via an operating limit on allowed minimum monitored DNBR underflow vs. measured incore axial shape index. This operating limit will be established prior to issuance of the operating license and will be based on consideration of many different allowed operating conditions (axial and radial power distributions as well as coolants) at any axial shape index. It will be shown in the following paragraphs that operation within these design limits is achievable.

4.3.2.2.3 Expected Power Distributions

Figures 4.3-3 through 4.3-17 and 4.3-18 through 4.3-23 show typical first cycle planar radial and unrodded core average axial power distributions, respectively. They illustrate conditions expected at full power for various times in the fuel cycle as specified on the figures. It is expected that the normal operation of the reactor will be with limited CEA insertion so that these power distributions represent the expected power distribution during most of the cycle. The three-dimensional peaking factor, $F^n r$, expected during (q) steady-state operation is then just the product of the unrodded planar radial peaking factor ($F^n q$) and the axial peaking factor. The maximum expected value of $F^n q$ is 1.88 during the first cycle and, as can be seen from the above (r) figures, occurs near the beginning-of-cycle for steady-state, base loaded operation with no full-length of part-length CEA insertion. Additionally, Figures 4.3-24 through 4.3-35 show typical planar radial power distributions for later cycles of System 80. Based on the similarity of these radial power distributions to those in the first cycle, it is clear that no significant differences in expected power distributions between cycles should exist. The uncertainty associated with these calculated power distributions is discussed in Section 4.3.3.1.2.2.6.

are calculated with two-dimensional, quarter-core nuclear models. Spatial distributions of materials and flux weighting are explicitly performed for the particular conditions at which the reactivity coefficients are calculated. The adequacy of this method is discussed in Section 4.3.3.1.2.

4.3.2.3.1 Fuel Temperature Coefficient

The fuel temperature coefficient is the change in reactivity per unit change in fuel temperature. A change in fuel temperature affects the reaction rates in both the thermal and epithermal neutron energy regimes. Epithermally, the principal contributor to the change in reaction rate with fuel temperature is the Doppler effect arising from the increase in absorption widths of the resonances with an increase in fuel temperature. nThe ensuing increase in absorption rate with fuel temperature causes a negative fuel temperature coefficient. In the thermal energy regime, a change in reaction rate with fuel temperature arises from the effect of temperature dependent scattering properties of the fuel matrix on the thermal neutron spectrum. In typical PWR fuels containing strong resonance absorbers such as U-238 and Pu-240, the magnitude of the component of the fuel temperature coefficient arising from the Doppler effect is more than a factor of 10 larger than the magnitude of the thermal energy component.

Figure 4.3-45 shows the dependence of the calculated fuel temperature coefficient on the fuel temperature, both at the beginning and the end of the first cycle.

4.3.2.3.2 Moderator Temperature Coefficient

The moderator temperature coefficient relates changes in reactivity to uniform changes in moderator temperature, including the effects of moderator density changes with changes in moderator temperature. Typically, an increase in the moderator temperature causes a decrease in the core moderator density and, therefore, less thermalization, which reduces the core reactivity. However, when soluble boron is present in the moderator, a reduction in moderator density causes a reduction in the content of soluble boron in the core, thus producing a positive contribution to the moderator temperature coefficient. In order to limit the dissolved boron concentration, burnable poison rods (shims) are provided in the form of cylindrical pellets of alumnia with uniformly dispersed boron carbide particles. The number of shims is given in Table 4.3-1 and their distribution in one quadrant of the core is shown in Figure 4.3-1. The distribution is identical for the other three quadrants. The reactivity control provided by the shims is given in Table 4.3-1. This control makes possible a reduction in the dissolved boron concentration to the values given in Table 4.3-1.

The calculated moderator temperature coefficients for various core conditions at beginnings and end of first cycle are given in Table 4.3-4. The moderator temperature coefficients are more negative at end-of-cycle because the soluble boron in the coolant is reduced. The buildup of equilibrium xenon produces a net negative change of -0.4×10^{-4} Wp/°F in the moderator temperature coefficient; this change is due mainly to the accompanying reduction in critical soluble boron. The changing fuel isotopic concentrations and the changing neutron spectrum during the fuel cycle depletion also contribute a small negative component to the moderator temperature coefficient. $\Delta\rho$

b. Components of the Safety Injection System and instrumentation which must operate following a LOCA are designed to operate in the environment of Section 3.11.

c. The Safety Injection System is designed to perform the functions of Section 6.3.1.1 for the entire duration of a LOCA.

d. The Safety Injection System is designed to Seismic Category I requirements.

6.3.1.3 Interface Requirements

Below are detailed the interface requirements that the SIS places on certain aspects of the BOP, listed by categories. In addition, applicable GDC and Regulatory Guides, which C-E utilizes in its design of the SIS, are presented. These GDC and Regulatory Guides are listed only to show what C-E considers to be relevant, and are not imposed as interface requirements, unless specifically called out as such in a particular interface requirement.

Relevant GDC - 1, 2, 3, 4, 13, 18, 20, 21, 22, 23, 35, 36, 37, 54, 57

Relevant Reg. Guides - 1.1, 1.26, 1.28, 1.29, 1.31, 1.36, 1.38, 1.44, 1.46, 1.48, 1.53, 1.64, 1.68, 1.75, 1.79, 1.82

A. Power

1. The Safety Injection System pumps and valves shall be capable of being powered from the plant turbine generator (onsite power source), and/or plant startup power source (offsite power), and the emergency generators (emergency power). 7

2. Power connections shall be through a minimum of two independent buses so that in the event of a LOCA in conjunction with a single failure in the electrical supply, the flow from one high-pressure and one low-pressure safety injection train shall be available for core protection.

3. Each electrical bus of the above shall be connected to one high-pressure safety injection pump and associated valves and one low-pressure safety injection pump and associated valves.

4. Each emergency generator and the automatic sequencers necessary for generator loading shall be designed such that flow is delivered to the RCS ~~core is attained~~ within a maximum of ~~[illegible]~~ 29 seconds after SIAS is generated. The emergency generator interface requirements are described in Section 8.3.1 and shall be complied with.

5. Instrument power supplies shall be provided as stated in Chapter 8.

6. The SIS hot leg injection valves shall be powered such that a single electrical failure cannot cause spurious initiation of hot leg injection flow through either hot leg injection line, nor

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March 31, 1982

6.3.3.3 Small Break Analysis

6.3.3.3.1 Evaluation Model

The calculations reported in this section were performed using the C-E small break evaluation model which is described in Reference 1 and was approved by the NRC in Reference 2. In the C-E model, the CEFLASH-4AS[3] computer program is used to determine the primary system hydraulic parameters during the blowdown phase, and the COMPERC-II[4] computer program is used to determine the system behavior during the reflood phase. Fuel rod temperatures and clad oxidation percentages are calculated using the STRIKIN-II[5] and PARCH[6] computer programs. The interfacing between these programs is discussed in detail in Reference 1.

6.3.3.3.2 Safety Injection System Assumptions

As discussed in Section 6.3.3.3.2.2, the safety injection system (SIS) includes two high pressure pumps, two low pressure pumps and four safety injection tanks. It is conservatively assumed that offsite power is lost upon reactor trip and therefore all safety injection pumps must await diesel startup and load sequencing before they can start. The total time
29 delay assumed is 30 seconds. from the time that the SIAS is generated to the time that SI flow is delivered to the RCS. For breaks in the pump discharge leg, it is also assumed that all safety injection flow delivered to the broken cold leg spills out the break.

An analysis of the possible single failures that can occur within the SIS has shown that the worst single failure for the small break spectrum is the failure of one of the emergency diesels to start[1]. This failure causes a loss of both a high pressure pump and a low pressure pump and results in a minimum of safety injection water being available to cool the core. Therefore, based on the above assumptions, the following safety injection flows are credited for the small break analysis since each high pressure safety injection pump (HPSIP) is piped so that it can feed all four cold leg injection points:

a. for a break in the pump discharge leg, the HPSIP flow credited is 75% of the flow from one HPSIP. The remaining 25% is assumed to spill out the break.

b. for breaks in other locations, the HPSIP flow credited is 100% of one HPSIP.

Since each low pressure safety injection pump (LPSIP) is piped so that it feeds two of the cold leg injection points:

a. for a break in the pump discharge leg, the LPSIP flow credited is 50% of the flow from one LPSIP. The remaining 50% is assumed to spill out the break.

b. for breaks in other locations, the LPSIP flow credited is 100% of one LPSIP.

This trend continues until the core does not uncover at all. For System 80 this occurs for a break size between 0.05 ft^2 and 0.02 ft^2 (and for all smaller breaks).

As the break size decreases, both the later time of initial core uncovery and its shallower depth tend to mitigate the temperature transient. However, the increased duration of uncovery acts in the opposite direction. In progressing from the 0.2 ft^2 break to 0.05 ft^2 break the increased duration dominates and therefore the peak clad temperatures rise. This trend continues until a break size is reached, typified by the 0.05 ft^2 break, where the three parameters are balanced. For breaks smaller than this, the increase in time to initial core uncovery and the shallower depth dominate causing less severe temperature transients. This trend continues until the core does not uncover as typified by the 0.02 ft^2 break. Thus, by analyzing several break sizes over this range, the behavior of PCT versus break size can be adequately determined.

To demonstrate the conservatism associated with the small break ECCS performance results provided herein, the 0.05 ft^2 break was reanalyzed using a more realistic measure of the decay heat generation rate. As required by Appendix K to 10CFR50, the spectrum analysis employed a decay heat generation rate equal to 120% of the standard ANS curve. The reanalysis of the 0.05 ft^2 break used a decay heat generation rate equal to 100% of the ANS curve. This one change reduced the peak clad temperature from 1557°F to 1020°F.

6.3.3.3.7 Instrument Tube Rupture

In addition to the six small breaks discussed above, the rupture of an in-core instrument tube was considered. A break, equal in size to a completely severed instrument tube (0.003 ft^2) was postulated to occur in the reactor vessel bottom head.

Following rupture, the primary system depressurizes until a reactor scram signal and safety injection actuation signal (SIAS) are generated due to low pressurizer pressure at 1600 psia. The assumed loss of offsite power causes the primary coolant pump and the feedwater pumps to coast down. After the 25 ~~30~~ second delay required to ~~start the emergency diesel and the high pressure safety injection pump~~ following SIAS, safety injection flow is initiated to the ~~reactor vessel~~ RCS. At this time an emergency feedwater pump is also started, providing a source of cooling to the steam generators. Due to the assumed failure of one diesel, only one high pressure safety injection pump and one emergency feedwater pump are available. (Four SITs and one low pressure safety injection pump are also available but do not inject due to the high RCS pressure.) The steam generator secondary sides also become isolated at this time.

The primary side depressurization continues accompanied by a rise in secondary side pressure until the secondary side pressure reaches the lowest set point of the steam generator safety relief valves. The primary system pressure continues to fall until it is just slightly greater than the secondary side pressure. At this point, the flow from the one operating HPSIP (66.3 lbm/sec) exceeds the leak flow (26.4 lbm/sec). Therefore the

failure of one of the low pressure pumps to start(2). This results in a minimum amount of safety injection water, available to the core, without affecting the operation of the containment spray system.

Therefore, based on the above assumptions, the following safety injection flows are credited for the large break analysis:

Two high pressure safety injection pumps (HPSIP's) are piped so that each one can feed all four cold leg injection points. Thus:

a. for a break in the pump discharge leg, the safety injection flow credited is 75% of the flow from two HPSIP's since it is assumed that all injection in the broken cold leg is spilled.

b. for breaks in other locations, the safety injection flow credited is 100% of two HPSIP's.

Two low pressure safety injection pumps (LPSIP's) are piped so that each one feeds two cold leg injection points. Thus:

a. for a break in the pump discharge leg, the safety injection flow credited is 50% of the flow from one LPSIP. The bases for this flow is that only one LPSIP is operable (worst single failure) and one of the two injection points for the operable pump is located in the broken loop and thus that flow is spilled.

b. for breaks in other locations, the safety injection flow is 100% of one LPSIP.

Four safety injection tanks (SIT's) are piped so that each SIT feeds a single cold leg injection point. Thus:

a. for a break in the pump discharge leg, the safety injection flow credited is 100% flow from three SIT's since it is assumed that all injection in the cold leg is spilled.

b. for breaks in other locations, the safety injection flow credited is 100% flow from four SIT's.

The rate at which emergency cooling water is delivered to the reactor vessel downcomer for the limiting break is shown in Figure 6.3.3.2-6L. System delivery data for the high and low pressure pumps are presented in Section 6.3.3.3. As shown in Table 6.3.3.2-1, no credit is taken for pump flow until the tanks are empty, resulting in a minimum effective delay of over 55. seconds from the time the SIAS setpoint is reached until pump flow is ~~initiated~~ delivered to the RCS. The actual delay time will not exceed ~~30~~ 29 seconds following a SIAS. In the large break analysis, no operator action has been assumed.

6.3.3.2.3 Core and System Parameters

The significant core and system parameters used in the large break calculations are presented in Table 6.3.3.2-2. The peak linear heat generation

TABLE 6.3.2-1 (Cont'd.) (Sheet 2 of 2)

SAFETY INJECTION SYSTEM
COMPONENTS PARAMETERS

Safety Injection Tanks	
Quantity	4
Safety Classification	2
Code	ASME III, Class 2
Design Pressure, Internal/External	700 psig/100 psig
Design Temperature	200°F
Operating Temperature	140°F
Normal Operating Pressure	610 psig
Minimum Operating Pressure	600 psig
Volume, Total	2400 ft^3
Liquid	
Minimum	1790
Nominal	1858
Maximum	~~1951~~ 1927
Fluid	Borated Water, 4200 ppm Boron Nominal, 6200 ppm max.
Material	Clad - Stainless Steel, type 304, 316, or approved alternate Body - Carbon Steel, type SA-516 Gr.7 or approved alternate

TABLE 7.5-3

POST-ACCIDENT MONITORING INSTRUMENTATION[1]

Parameter[3]	Type of Readout	Number of 1E Channels	Range[~~5~~ 4]	Location	
Pressurizer Pressure	Indicator	2	15-3000 psia	Control Room	8
	Recorder	1	15-3000 psia	Control Room	
Reactor Coolant System Pressure	Indicator	2 (4)	0-4000 psi~~a~~G	Control Room	8
	Recorder	1 (4)	0-4000 psi~~a~~G	Control Room	
Pressurizer Level	Indicator	2	0-100%	Control Room	
	Recorder	1	0-100%	Control Room	
Steam Generator Pressure	Indicator	2/SG	15-1524 psia	Control Room	8
	Recorder	1/SG	15-1524 psia	Control Room	
Steam Generator Level	Indicator	2/SG	0-100%	Control Room	
	Recorder	1/SG	0-100%	Control Room	
Containment Pressure (2)	Indicator	2	-4 to 85 psig	Control Room	
	Recorder	1	-5 to 180 psig	Control Room	
Reactor Coolant Temperature-Hot Leg	Indicator	4	50-750°F	Control Room	
	Recorder	2	50-750°F	Control Room	8
Reactor Coolant Temperature-Cold Leg	Indicator	4	50-750°F	Control Room	
	Recorder	2	50-750°F	Control Room	

Notes:

1. Post-accident monitoring instrumentation is qualified for the appropriate environmental conditions (refer to Section 3.11).

2. The containment pressure instrumentation may be supplied by the Applicant, the specified range and accuracy values above are typical.

3. MCBD's are provided in Section 7.1.

~~4. Non 1E Channel provided per RG 1.97, Rev. 2 Category 3 application where the state of the art will not support requirements for higher qualified instrumentation~~. 8

4. ~~5~~. Post-accident channel accuracy is a time dependent function of post-accident environmental conditions.

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TABLE 8.3.1-4

REQUIRED STANDBY GENERATOR LOADS

The loads designated below for a particular ESFAS signal are those that shall be started to establish the stated conditions within the times specified. If offsite power is unavailable, then the loads designated below for a particular ESFAS signal shall be supplied from the standby generator, and sequenced on to meet the stated conditions. CSAS will also start the Standby Generator and sequencing shall be determined by the Applicant.

	Loss of Offsite Power Concurrent with the below listed ESFAS:	
Equipment	SIAS	EFAS
LPSI Pumps	(1)	N/A
HPSI Pumps	(1)	N/A
Motor Operated Valves (as appropriate for each signal)	(1)	(2)
Emergency Feedwater Pumps	N/A	(2)

NOTES

N/A: Not Actuated

(1) Sequence on such that flow is delivered to the RCS ~~core~~ ~~is attained~~ within a maximum of 29 ~~30~~ seconds after SIAS. 8

(2) Sequence on such that flow to the steam generator is attained within a maximum of 45 seconds after EFAS. See Section 5.1.4.

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The resulting radiation level from the spent fuel is 2.5 mr/hr or less at the surface of the water in the refueling pool (30 mr/hr or less ~~at~~ from the spent fuel in the spent fuel pool) when the shielding of the appropriate fuel handling machine is taken into account.

the reactor for the operator; electronic and visual indication of the refueling machine position over the core; a protective shroud into which the fuel assembly is drawn by the refueling machine; transfer system upenders manual operation by a special tool in the event that the hydraulic system becomes inoperative; and removal of the transfer system components from the refueling pool for servicing without draining the water from the pool.

c. The fuel transfer tube is sufficiently large to provide natural circulation cooling of a fuel assembly in the unlikely event that the transfer carriage should be stopped in the tube. The manual operator for the fuel transfer tube valve extends from the valve to the operating deck. Also, the valve operator has enough flexibility to allow for operation of the valve even with thermal expansion of the fuel transfer tube.

d. Travel stops in both the refueling and spent fuel handling machines restrict withdrawal of the spent fuel assemblies. This results in the maintenance of a minimum water cover of 9 feet over the active portion of the fuel assembly. ~~resulting in a radiation level of 2.5 mr/hr or less at the surface of the water when the additional shielding of the handling tool is taken into account. The depth of water surrounding the fuel transfer canal, transfer tube and spent fuel storage pool is sufficient to limit the maximum continuous radiation levels in working areas to 2.5 mr/hr.~~

9.1.4.3.5 Reactor Vessel Closure Head Handling

The maximum load-carrying capability of the reactor vessel support system was evaluated by an elastic-plastic static analysis of the support columns using the MARC finite element computer program. Each column was modeled with 13 rectangular section beam finite elements. The material properties used were for SA 533 Gr B steel at room temperature (Figures 9.1-18 and 9.1-19). From the stress-strain curves the initial yield point was determined to be 65,000 psi.

The axial load vs. deflection curve (Figure 9.1-20) was obtained by applying a gradually increasing axial load up to the point of static instability. The stiffness of each reactor vessel support column is seen to be about 40 $\times 10^6$ lb per inch, up to a limit load of about 22 $\times 10^6$ lbs.

A nonlinear dynamic elastic-plastic analysis of the reactor vessel support system was performed using the MARC program to evaluate the time-dependent behavior of the columns. The results of a straight drop of the head assembly from 18 feet indicate that the load in the columns will reach the maximum load-carrying capability of the columns and some plastic deformation will occur. The peak axial deflection for a drop from this height was determined to be .56 inches. The results show that the reactor vessel supports would remain intact and continue to support the weight of the vessel following such an event.

TABLE 9.3-6

(Sheet 1 of 4)

CHEMICAL AND VOLUME CONTROL SYSTEM

PROCESS FLOW DATA[5]

CVCS Minimum Purification Operation (One Charging Pump in Operation)

CVCS Location:	1	2	3	4	5	6	7	8	8a	8b	8c	9
Flow, gpm	30	30	30	30	30	30	20	10	2	8	10	30
Press., psig	2235	2235	460	460	60	60	60	60	60	60	60	60
Temp., F	565	300	300	120	120	120	120	120	120	120	120	120

CHANGE TO 3.9

CVCS Minimum Purification Operation (One Charging Pump in Operation)

CVCS Location:	10	10a	10b	10c	11	12	13m	13b	13	13c	13n	13d, e,f,g	13h, j,k,l	14a, b,c,d	14e	14f	14g
Flow, gpm	30	30	30	44	44	44	44	44	44	0	0	0	0	39	0	15.6	15.6
Press., psig	60	60	20	20	20	2485	2485	2455	2485	2400	2400	2400	2400	100	100	100	70
Temp., F	120	120	120	120	120	120	120	385	385	120	120	120	120	180	180	180	180

CVCS Normal Purification Operation (Two Charging Pumps in Operation)

CVCS Location:	1	2	3	4	5	6	7	8	8a	8b	8c	9
Flow, gpm	72	72	72	72	72	72	62	10	2	8	10	72
Press., psig	2235	2235	460	460	60	60	60	60	60	60	60	60
Temp., F	565	310	310	120	120	120	120	120	120	120	120	120

CVCS Normal Purification Operation (Two Charging Pumps in Operation)

CVCS Location:	10	10a	10b	10c	11	12	13m	13b	13	13c	13n	13d, e,f,g	13h, j,k,l	14a, b,c,d	14e	14f	14g
Flow, gpm	72	72	72	86	88	88	61.6	61.6	61.6	26.4	26.4	6.6	6.6	3.9	0	15.6	15.6
Press., psig	60	60	20	20	20	2485	2455	2485	2400	2400	2400	2400	2400	100	100	100	70
Temp., F	120	120	120	120	120	120	120	445	445	120	120	120	120	180	180	180	180

j. Include redundancy in the design so that neither a single component failure nor a single operator error result in excess steam releases.

k. Provide a condenser interlock which will block turbine bypass flow when unit condenser pressure exceeds a preset limit.

The environmental design criteria are listed in the Applicant's SAR.

10.4.4.2 System Description and Operation

10.4.4.2.1 General Description

The turbine bypass system consists of the steam bypass control system, the turbine bypass valves and associated piping and instrumentation. The steam bypass control system is described in Section 7.7.1.1.5.

10.4.4.2.2 Piping and Instrumentation

A typical turbine bypass system consisting of eight turbine bypass valves located in lines branching from each main steam line, down stream of the main steam isolation valves and connecting to the main condenser is shown in Figure 10.3.4-1.

10.4.4.2.3 Turbine Bypass Valves

The turbine bypass valves are air operated valves with a combined capacity of 55% of the total full power steam flow at normal full ~~pwer~~ POWER steam generator pressure (1070 psia). The valves are normally controlled by the steam bypass control system but are capable of remote or local manual operation. When operating automatically the valves modulate full open ~~to~~ or full close in a minimum of 15 seconds and a maximum of 20 seconds. In response to a quick opening signal from the steam bypass control system they are designed to open in 1 second. In response to a closing signal from the steam bypass control system they are designed to close in 5 seconds. The system is capable of controlling at flows as low as 63,000 lb/hr in order to permit operation at hot standby during precore hot functional testing.

10.4.4.2.4 System Operation

The turbine bypass system takes steam from the main steam lines upstream of the turbine stop valves and discharges it directly to the main condenser, bypassing the turbine-generator. During normal operation, the bypass valves are under the control of the steam bypass control system, as discussed in Section 7.7.1.1.5. During cooldown or hot shutdown, the turbine bypass valves may be actuated individually from the main control room to regulate steam generator pressure and reactor coolant temperature change.

10.4.4.3 Safety Evaluation

The valves in the turbine bypass system are designed to fail closed to prevent uncontrolled bypass of steam to the condenser. Should the bypass valves fail to open on command, the secondary safety valves provide main

4. This table evaluates the ICC Detection Instrumentation's conformance to the NUREG-0737, Item II.F.2 documentation requirements. Table 3-2 evaluates conformance to Attachment 1 of Item II.F.2. Table 3-3 evaluates conformance to Appendix B of NUREG-0737. 6

5. The ICC detection instrumentation processing and display consists of two computer systems; the 2 redundant channel safety grade microcomputer based QSPDS, and the single large scale non safety grade minicomputer based CFMS. The ICC inputs are acquired and processed by the safety grade QSPDS 9 and isolated and transmitted to the primary display in the non-safety-grade CFMS. The QSPDS also has the seismically qualified displays for the ICC detection instruments. The software functions for processing are listed in Section 2.2; 6 the functions for display are listed in Section 2.3.

The software for the QSPDS has been designed using the 9 recommendations of the draft standard, IEEE Std. P742/ANS 4.3.2, "Criteria for the Application of Programmable Digital 6 Computer Systems in the Safety Systems of Nuclear Power 9 Generating Stations" as a design guideline. This design procedure verifies and validates that the QSPDS software is properly implemented and integrated with the system hardware 6 to meet the system's functional requirements. This procedure is quality assured by means of the C-E QADP. Although the 9 CFMS is designed as a non-safety class system, a similar procedure is being applied to the CFMS design to assure compatibility with the QSPDS. 6

The QSPDS hardware is designed as a redundant safety grade 9 qualified computer system which is designed to the avail ability goal of 0.99 with the appropriate spare parts and maintenance support. The CFMS is a single highly reliable 6 minicomputer system that is designed to the availability goal of (0.99) with the appropriate spare parts and maintenance 9 support.

6. This requirement is plant specific.

7. Guidelines and procedures for the use of the ICC instruments 6 will be addressed on a plant-specific basis. The C-E Owner's Group has developed generic emergency procedure guidelines addressing ICC.

8. Key operator actions for ICC are contained in emergency 9 procedures and will be addressed on a plant specific basis. ~~The C-E Owner's Group has developed generic emergency procedure guidelines addressing ICC.~~ 6

Amendment No. 9
February 27, 1984