SNUPPS

Standardized Nuclear Unit
Power Plant System

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May 14, 1984

SLNRC 84- 0080 FILE: 0543/0278
SUBJ: SNUPPS Technical Specification
Reactor Systems Branch Issues

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

Docket Nos.: STN 50-482 and STN 50-483

References:
1. NRC letter (B. Youngblood) to Union Electric Company (D. Schnell) and Kansas Gas and Electric Company (G. Koester), dated May 7, 1984, Request for Additional Information
2. ULNRC-792, dated April 9, 1984: Callaway Plant Unit No. 1 Technical Specifications
3. ULNRC-816, dated May 11, 1984: Callaway Plant Unit No. 1 Technical Specifications

Dear Mr. Denton:

Enclosed are SNUPPS responses to the reference NRC request for additional information which is based on the recent Reactor Systems Branch (RSB) review of the Technical Specifications for the Callaway Plant. The responses are also applicable to the draft Technical Specifications for Wolf Creek Generating Station.

In addition to providing the requisite additional information, this letter documents, in the following paragraphs, certain objections to the timeliness and scope of the RSB review.

The SNUPPS Technical Specifications have been under review and development by Union Electric , Kansas Gas and Electric , Bechtel, Westinghouse, and SNUPPS Staff since early 1981. During this time there have been numerous working meetings at which the Technical Specifications were refined until they reached the form in which they presently exist. The first marked-up version of the Callaway Plant Standard Technical Specifications was submitted on June 29, 1982. This was followed by an initial Wolf Creek submittal on August 5, 1982, and numerous other SNUPPS submittals both formal and informal since that time (see SLNRC 83-036 7/8/83; SLNRC 84-029, 2/10/84; SLNRC 84-038, 2/27/84; and SLNRC 84-046, 3/20/84). These submittals were interspersed with meetings

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with the NRC on miscellaneous Technical Specification subjects on 2/3/83, 8/30/83, 9/19/83, 11/14/83, 12/13/83, 1/3/84, 2/1/84, and 3/13/84. These meetings included detailed discussion of Technical Specification issues with many NRC technical review branches including RSB.

The proof and review of Callaway's Technical Specifications was to begin on January 11, 1984 and conclude on or about February 17, 1984. At that time, all the NRC branches were to have completed their review of the Technical Specifications; however, during this time virtually no feedback was received on any of the issues subsequently raised by RSB.

Finally, during the week of April 9, 1984, the SNUPPS Utilities were informed of 67 RSB issues related to the Callaway Technical Specifications. After extensive evaluation of the issues by the SNUPPS Utilities and Staff and Westinghouse and Bechtel, the SNUPPS responses were discussed with the RSB in meetings on April 13, April 26 and May 3, 1984. As a result of these discussions, the final list of 16 questions in the reference letter was issued.

In the review of the RSB concerns, it was recognized that a majority of the issues were the result of the format and content of the Standard Technical Specifications and were not based on any design features specific to the SNUPPS plants. Thus, these issues are generic. Nevertheless, RSB indicated that approval of the Callaway Technical Specifications was contingent on resolution of all the issues. Therefore, to support the Callaway licensing schedule, the SNUPPS Utilities agreed to address the RSB issues, including those which are generic, rather than invoke either NRC staff procedures for handling generic regulatory requirements or NRC staff procedures for control of backfitting new technical requirements.

It should be noted that in several instances where the SNUPPS Utilities proposed changes which impacted Standard Technical Specification form or content, the NRC staff stated that the changes could not be made without formal review by the Committee for the Review of Generic Requirements. The changes proposed by RSB which impact Standard Technical Specification form and content had to undergo no such formal review.

The previously completed proof-and-review process, the reviews documented in Reference 2 and 3, and the enclosed SNUPPS responses to the RSB issues provide acceptable technical bases for approval by the NRC staff of the Callaway Technical Specifications for operation up to and including full power operation.

Very truly yours,

NA Petrick

Nicholas A. Petrick

MHF/nld10a10&11
Enclosure
cc: See Page 3

cc:	D. F. Schnell	UE
	D. T. McPhee	KCPL
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	J. Neisler/B. Little	USNRC/CAL
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1. Reactor Trip System Instrumentation Setpoints, Table 2.2-1 (Section 2.2, page 2-4)

In reviewing the RPS/ESFAS Setpoint Methodology Table, which is used to determine the relationship between the RPS/ESFAS setpoints, the allowable values, and the values of these parameters which are used in the safety analyses, the following discrepancies were observed:

	Setpoint Methodology Table	FSAR Table 15.0.4	FSAR text
(a) PZR pressure-Hi	2445 psig	2410 psig	-
(b) Steam Generator Level Lo-Lo	0% span	7.2% span	8.5%* span

During the April 13, 1984 meeting the applicant stated that:

(a) The Chapter 15 accident analyses assumed the higher PZR pressure trip of 2445 psig. However, the applicant stated that the NRC has not been provided these analyses.

(b) The Steam Generator Level Lo-Lo trip setpoint is incorrect, and the FSAR will be changed to reflect the 0% value.

We require the applicant:

(1) To document its responses as described in (a), and (b) above.

(2) To confirm that there are no other discrepancies between the analyses and those values shown on the setpoint methodology table.

*FSAR Section 15.2.8.2, Feedwater Line Rupture, Steam Generator Water Level Lo-Lo trip setpoint minus 15% = 23.5 - 15 = 8.5%.

Response:

#1. 1)There are two values in FSAR Table 15.0.4 which disagree with the safety analysis values noted in the setpoint study.

(a) Pressurizer Pressure high

The FSAR safety analysis value provided for reactor trip on high pressurizer pressure is 2410 psig. This value was used to perform the FSAR analysis. When it became known that error associated with the Barton transmitters had to be increased, an evaluation was performed by Westinghouse to determine the impact of increasing the analysis value. It was found that pressurizer pressure could be as high as 2445 psig without impacting the analysis. The additional error introduced by the long term drift of the Barton transmitters, which is in excess of the Westinghouse purchase specification requirements, is completely enveloped by the 35 pound increase in the safety analyses value. Based on this evaluation 2445 psig was used in determining the reactor trip setpoint.

The only transient for which high pressurizer pressure is assumed to be available is the Loss of Load event. Westinghouse has performed an evaluation for the acceptability of a higher analytical setpoint to address an identified long term drift concern with Barton transmitters. The results of the evaluation concluded that the design basis for this transient would still be met with insignificant affect on peak pressure and DNBR.

This can be demonstrated for SNUPPS as illustrated in Figures 15.2.1 through 15.2-8 of the FSAR. Of the four cases analyzed, the first two cases (Fig. 15.2-1 through 15.2-4) tripped on OT Delta T and Low-Low steam generator level. The third and fourth cases tripped on high pressurizer pressure. The time to reach a 2460 psia setpoint versus 2425 psia is less than one second. The DNBR for these cases never drop below the initial steady state value. If the high pressure trip were not available, the OT Delta T trip has been shown in various studies to be sufficient to meet the basis for this event. Therefore, Westinghouse has determined that the analytical setpoint for 2445 psig is acceptable.

Upon final resolution of the transmitter problem by Barton, SNUPPS will evaluate the need for an FSAR change.

(b) Steam Generator Level Lo-Lo

The reactor trip safety analysis value for steam generator low-low level is incorrectly reported in the FSAR as 7.2%. The limiting analysis was performed assuming a setpoint of 0% level. The FSAR will be amended in a future revision to indicate 0%.

2)There were no further discrepancies between the safety analysis limits in the FSAR and those used in the Westinghouse Setpoint Methodology.

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2. <u>Reactor Trip System Instrumentation Response Times</u> (Section 3/4.3.1, Table 3.3-2, pages 3/4 3-8)

This table specifies the maximum acceptable response times for the reactor trip system instrumentation. Item 17 of this table indicates the response time for a reactor trip as a result of safety injection signal as not applicable (N.A.).

The staff notes that the reactor trip due to a safety injection signal is relied upon in the steam line break safety analysis. Therefore, the staff believes that the reactor trip system instrumentation response times should be provided in this table for a reactor trip due to safety injection.

The staff requires that the applicant either: (a) propose a response time consistent with the safety analyses, or (b) provide a justification for not specifying such a response time.

Response:

#2 The response time applicable for reactor trip on a safety injection signal is already covered in the ESFAS response time table (table 3.3-5).

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3. Engineered Safety Features Actuation System (ESFAS) Instrumentation
(Section 3/4.3.2, Table 3.3-3, page 3/4 3-14)

This table provides the minimum required number of operable ESFAS channels and the modes in which they are to be operable. Item 1.e specifies that the steam line pressure - Lo channels are to be operable in Modes 1, 2, and 3##. The footnote symbol ## indicates that these channels may be bypassed above P-11.

The staff notes that this is a contradiction since if these ESFAS channels are to be operable in Modes 1 and 2, they may not be bypassed above P-11 in Mode 3.

In the April 13, 1984 meeting the applicant stated that the above footnote is a typographical error and the applicant will propose a Tech Spec change to replace it by another footnote to state the channels may be bypassed below P-11. The staff believes this change should be made and is acceptable.

Response:

#3 To correct a typographical error, the SNUPPS utilities have submitted a change to page 3/4 3-14 (attached) deleting one of the # symbols from 1.e. Since no other ## footnotes appear on this table, a change to page 3/4 3-20 was also submitted (attached) to delete the same.

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TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Component Cooling Water, Turbine Trip, Auxiliary Feedwater-Motor-Driven Pump, Emergency Diesel Generator Operation, Containment Cooling, and Essential Service Water Operation)					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-1	3	2	2	1, 2, 3	15*
d. Pressurizer Pressure - Low	4	2	3	1, 2, 3#	19*
e. Steam Line Pressure-Low	3/steam line	2/steam line any steam line	2/steam line	1, 2, 3#	15*
2. Containment Spray					
a. Manual Initiation	2 pair	1 pair operated simultaneously	2 pair	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays (SSPS)	2	1	2	1, 2, 3, 4	14
c. Containment Pressure-High-3	4	2	3	1, 2, 3	16

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TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

~~##Trip function may be bypassed in this MODE above the P-11 (Pressurizer Pressure Interlock) Setpoint.~~

###Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam line pressure is not blocked.

*The provisions of Specification 3.0.4 are not applicable.

**One in Separation Group 1 and one in Separation Group 4.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 - 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 met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

4. Engineered Safety Features Response Times (Section 3/4.3.2, Table 3.3-5, page 3/4 3-31)

This table specifies the maximum acceptable response times for the engineered safety features.

Item 7 of this table shows no response time requirement for the function of steam line isolation on a signal from the steam line high negative pressure rate. Below P-11 (a reactor coolant pressure of 1970 psig) this high negative rate trip is the only automatic means for steam line isolation following a steam line break outside the containment.

The staff requires the applicant to: either (a) specify a maximum acceptable response time for item 7 above, or (b) justify the the consequences of a steam line break outside the containment without a timely steamline isolation.

Response:

#4 The response time for high negative steam pressure rate should be less than or equal to 7 seconds (Table 3.3-5). Since the accident analysis for steam line breaks outside containment below P-11 are bounded by those steam line breaks which take credit for Steam Line Pressure-Low and Containment Pressure High-High for Steam Line Isolation, the response time of $\leq$ 7 seconds for Steam Line Isolation from Steam Line Pressure Negative Rate-High is adequately conservative.

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TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
5. Containment Pressure--High-3	
a. Containment Spray	≤ 32(1)/20(2)
b. Phase "B" Isolation	≤ 31.5
6. Containment Pressure--High-2	
Steam Line Isolation	≤ 7
7. Steam Line Pressure-Negative Rate-High	
Steam Line Isolation	~~N.A.~~ ≤7
8. Steam Generator Water Level--High-High	
a. Feedwater Isolation	≤ 7
b. Turbine Trip	≤ 2.5
9. Steam Generator Water Level - Low-Low	
a. Start Motor-Driven Auxiliary Feedwater Pumps	≤ 60
b. Start Turbine-Driven Auxiliary Feedwater Pump	≤ 60
10. Loss-of-Offsite Power	
Start Turbine-Driven Auxiliary Feedwater Pump	N.A.
11. Trip of All Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	N.A.

5. <u>Engineered Safety Features Response Times</u> (Section 3/4.3.2, Table 3.3-5, page 3/4 3-31)

This table specifies the maximum acceptable response times for the engineered safety features.

Item 8.b of this table requires a response time of less than or equal to 2.5 seconds for the steam generator level Hi-Hi signal to generate a Turbine Trip. However, in the FSAR analysis of excessive feedwater flow, Table 15.1-1, the Turbine Trip is assumed 2.2 seconds after a steam generator level Hi-Hi signal is generated.

The staff requires the applicant to either: (a) propose appropriate modifications to the Tech Specs, or (b) justify that the difference between the two times is insignificant.

Response:

#5 The response time for a turbine trip following a steam generator high-high level signal is not applicable with respect to the safety analysis. The turbine trip is provided for turbine protection only; if this signal was not generated, steam flow would eventually be terminated by steamline isolation generated by a low steam pressure signal.

Independent of the above consideration, a value of 2.5 seconds will have an insignificant effect on the safety analysis as presented in Section 15.1 of the SNUPPS FSAR. The minimum DNBR for this event is reached shortly after turbine trip (Table 15.1-1).

At the time of minimum DNBR, the DNBR is changing very slowly. This can be seen from the DNBR transient as presented in Figure 15.1-2. If the turbine trip is delayed by an additional .3 seconds, the DNBR will decrease by negligible amount but still be above the limit value of 1.30. The transient results of Figures 15.1-1 and 1-2 would not be visibly changed.

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6. Several Tech Spec Sections

(1) Technical Specification 3/4.4.3, entitled "Pressurizer," limits the pressurizer level to less than or equal to 92% (1657 ft^3) during Modes 1, 2 and 3. There is no Technical Specification low level limit on the pressurizer.

(2) The Tech Specs impose no limits on Steam Generator level in Modes 1, 2 and 3. However, Technical Specifications 3.4.1.2, 3.4.1.3, and 3.4.1.4.1 define an operable Steam Generator as one having $\geq$ 10% of the wide range span during Modes 3, 4, and 5.

(3) Tech Spec 3.1.1.4 specifies the minimum temperature for criticality in Modes 1 and 2 as Tave $\geq$ 551°F. Tech Spec 3.2.5 specifies the maximum Tave relative to DNB as Tave $\leq$ 595°F during Mode 1.

(4) Tech Spec 3.2.5 specifies the minimum pressurizer pressure elative to DNB as p $\geq$ 2220 psia during Mode 1. Tech Spec 3.4.2.1 and 2 specify the maximum possible operating system pressure (without safety valve lifting) to be 2485 psig ± 1% during Modes 1, 2, 3, 4, and 5.

The above four parameters, namely pressurizer level, steam generator level, Tave, and pressurizer pressure are among other input parameters that are used for the accident analysis. Nominal programmed values plus or minus an error allowance are used in the accident analysis, while per the Technical Specifications the plant is allowed to operate within a much wider range. For example, the plant may operate at the following conditions without violating the Technical Specifications:

(a) A pressurizer level of 92% or 0%

(b) A steam generator level of 78% or 23.5% (Although the Tech Specs do not limit the steam generator level to a high or low value, the reactor trip setpoint of steam generator level Hi-Hi is 78%. The Lo-Lo trip setpoint is 23.5%.)

(c) A Tave of between 551°F and 595°F.

(d) A pressurizer pressure of between 2220 psia and 2500 psia.

The staff finds that the Tech Specs may not be consistent with the FSAR analyses. That is, the plant may be operated outside those bounds assumed in the safety analyses. Therefore, it is not clear that the safety analyses are bounding.

The staff requires that the applicant either: (a) propose appropriate Tech Spec changes; or (b) verify that the Tech Specs are consistent with the FSAR accident analysis.

Response

#6 The concern raised by this question appears to be that the plant may be operated within all existing Technical Specification Limiting Conditions for Operations and yet may be in a condition which is not consistent with the analysis performed in the SNUPPS Final Safety Analysis Report (FSAR). SNUPPS has utilized the analyses and evaluations from the FSAR to derive the Callaway Final Draft Technical Specifications in accordance with 10CFR50.36(b). SNUPPS also provided the FSAR safety analyses in accordance with the SRP for Chapter 15.

The SNUPPS Technical Specifications and analyses are no different than those of other recent, Westinghouse reactor licensees. While licensing requirements for near term operating license applicants (NTOLs) and operating reactor licenses (ORs) may be different, the RSB's implied safety significance of this issue applies equally to NTOLs and ORs. SNUPPS therefore contends that the existing Technical Specifications and FSAR analyses are adequate and support licensing for Callaway.

The initial conditions in the accident analyses in Chapter 15 of the SNUPPS FSAR are based on the nominal programmed values of temperature, pressurizer pressure and level and steam generator level. To these nominal values are added appropriate measurement uncertainties which are added in the conservative direction (positive or negative) as appropriate to the accident in question. The Tech Specs are based upon the assumptions and results of the safety analyses which are consistent with the assumptions of nominal initial conditions plus or minus uncertainties.

The system parameters of RCS temperature, pressurizer pressure, pressurizer level, and steam generator level are maintained by automatic control systems. Temperature is controlled by the rod control system, and pressure by the heaters and sprays. The pressurizer and steam generator levels are programmed as a function of power level and use various input parameters to maintain the program level. All of these systems may be operated manually, in which case the operator strives to duplicate the automatic system.

The correct maintenance of these plant variables is consistent with the Tech Specs for two reasons. First, although failures in these systems could cause deviations from the programmed values, these failures would be detected by the resulting plant transient and subsequently fixed. These transients cause and are already bounded by the Condition II events and analyses presented in the SNUPPS FSAR. Second, these programmed values are displayed in the control room for each channel and these parameters are also monitored by the operator in accordance with plant operating procedures. In addition, deviation alarms sound in the control room if any of these four control parameters fall outside the program value. Thus, if a failure in the control system did not cause a transient or if the plant were operating under manual control, the operator would detect the failures as a result of the alarms or according to the surveillance requirements in the procedures and/or correct and maintain them as well.

The operating procedures used by the plant ensure that the plant is operated in accordance with the design of these control systems, whether they are operated automatically or in manual control. All control system parameters are specified in the Precautions, Limitations, and Setpoints (PLS) document. The PLS document parameters are developed to optimize plant operation and are consistent with the assumptions made in the safety analysis as to initial conditions and the programming of all control system variables. Furthermore, those control system variables which are important to the safety analysis are specifically noted in the PLS, usually by footnote.

Note that the maintenance of operating procedures is required by the Tech Specs. Since these operating procedures prevent abnormal combinations of plant parameters and since failures of the control systems are detectable, the Tech Specs as written are consistent with the safety analysis and no further specifications on controlled parameters is required. (The control systems are not required to operate once the accident has started. The safety analysis does not assume operation of the control system during the course of an accident unless operation of the control system makes the results more limiting with respect to the acceptance criteria.) Finally, the control systems, although not safety grade, are highly reliable systems. There is no indication based on a large amount of plant operating experience that the current Specs are not adequate.

It should be noted that the normal operational transients (Condition I events) do not cause these parameters to significantly deviate from their nominal values. This is because these Condition I events are the design basis transients of the control systems. The control system is designed to maintain the plant parameters within the specified programs and to return them to the program values if they deviate. The system response to these transients is provided in the setpoint study provided for each plant. The results presented in the setpoint study show that the control system can indeed function as designed and maintain the plant parameters as desired. Since normal operation does not cause large changes in these parameters, one is forced to conclude that it requires an abnormal transient (Condition II, III, or IV) to generate these adverse or odd initial conditions.

Thus, in order for an accident to occur in which the initial conditions are significantly outside the normal operating range, the plant must first undergo a transient which would generate the abnormal initial condition for the accident. This is, in effect, the consideration of two transients at once, which is outside the scope of the licensing basis accident analyses for the plant. In particular, if the plant were operating under automatic control of these parameters, the accident would have to occur very shortly after the first transient, since the

control system would be acting to restore the parameters to their normal values. The occurrence of such a "smart" accident is highly unlikely, and it is inappropriate to apply ANS classification and acceptance criteria. Since the ANS classification and acceptance criteria are based on the anticipated frequency of occurrence of a postulated accident, this "smart" accident would have a different classification than the accident based on normal conditions and would have more relaxed acceptance criteria.

Independent of the above considerations, there are extreme limits on temperature, pressurizer pressure, pressurizer level and steam generator level provided in the Tech Specs. For normal operation, Tavg is limited to 595°F (maximum), and pressure should not be less than 2220 psia. (Spec 3.2.5). Other limits are provided in terms of the trip setpoints which exist for pressurizer level (high), steam generator level (high and low) and pressurizer pressure (high and low). Both pressure and temperature are monitored with respect to reactor trip by the overpower/overtemperature delta-T trips. A minimum value of Tavg is specified in Tech Spec 3.1.1.4, the minimum temperature for criticality. Should any transient cause one of the trip setpoints to be reached, protective action will then be initiated. Thus the protection system setpoints serve to provide an envelope of initial conditions which prevent operation of the plant in extreme configurations.

CONCLUSIONS

A review of the Technical Specifications and the FSAR Chapter 15 Accident Analysis indicates that no safety issue exists due to the lack of Technical Specifications on the parameters discussed above. This conclusion has been reached as a result of the stated assumptions for the FSAR Chapter 15 Accident Analysis, normal operating practices and control limitations such as interlocks and limiters. For all these parameters, there are adequate annunciators to alert the operator to excessive deviation from the nominal parameters and allow timely restoration.

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7. <u>Relief Valves</u> (Section 3.4.4, page 3/4 4-10)

This section specifies the operability requirements for the pressurizer power-operated relief valves (PORVs). The applicability for this section is in Modes 1, 2, and 3, i.e., PORVs are required operable above a T_{ave} of 350°F.

The staff notes that Section 3.4.9.3, "Overpressure Protection Systems," requires low temperature overpressure protection below a T_{ave} of 368°F. However, the action statement for section 3.4.4 permits blocking off of one or more PORVs. Therefore, the PORVs would not be available to mitigate overpressurization events between 368°F and 350°F T_{ave} and while the RHRS is not lined up to the RCS. The staff requires that the applicant resolve this inconsistency.

The applicant is required to propose appropriate Tech Spec changes to resolve the above inconsistency.

Response:

#7 To make Specification 3.4.4 more clearly consistent with Specification 3.4.9.3, SNUPPS has submitted the attached changes.

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REACTOR COOLANT SYSTEM

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3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.*

ACTION:

See following page

~~a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~b. With one or more block valve(s) inoperable, within 1 hour: restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~c. The provisions of Specification 3.0.4 are not applicable.~~

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION a. in Specification 3.4.4.

* With all RCS cold leg temperatures above 368°F

ACTION:

a. With one or more PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With both PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore each of the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour: 1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and 2) apply the ACTION of b or c above, as appropriate for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable.

8. <u>Overpressure Protection System</u> (Figure 3.4-4, page 3/4 4-36)

Figure 3.4-4 specifies the maximum allowed PORV setpoint for the cold overpressure mitigation system.

In order to comply with the Appendix G limits throughout the life of the plant, this curve must be updated periodically to account for the irradiation of the pressure vessel.

To ensure that the above setpoint curve is updated when required, Figure 3.4-4 should be added to Surveillance Requirement 4.4.9.1.2 on page 3/4 4-29. Therefore, the staff requires that the applicant propose the appropriate Tech Spec modifications.

Response:

#8 To ensure that Figure 3.4-4 is updated periodically to account for the irradiation of the pressure vessel, the SNUPPS utilities have submitted the attached Specification with indicated changes.

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, ~~and~~ 3.4-3, and 3.4-4.

9. Accumulators (Section 3/4.5.1, page 3/4 5-1)

This section specifies the operability conditions for the cold leg accumulators (CLAs) during Modes 1, 2, and 3. Isolation valve positions, contained water volumes, boron concentration and nitrogen cover pressure in these tanks are specified. However, the contained water temperature is not specified. The CLAs water temperature is one of the assumptions used in the large break LOCA analysis.

With the absence of a temperature specification in the above section the plant may be operated with nonconservative CLA temperatures while not violating the existing Tech Specs. It is not clear if the CLA temperatures should be clearly specified, consistent with the Chapter 15 Analyses.

The staff requires that the applicant either: (a) propose appropriate Technical Specifications to limit the CLAs to within a certain band which is supported by the safety analysis, or (b) provide a technical justification to support the adequacy of the existing Tech Spec section.

Response:

#9 As noted in WCAP-8339, page 4-14, for conservatism the initial containment temperature assumed in an Appendix K ECCS analysis is the minimum value expected during normal operation. In the SNUPPS large break ECCS analysis, the accumulator water temperature is set at the value for initial containment temperature (90°F). Inasmuch as the accumulator tanks are not insulated and are located in containment, there is no reason for believing the water temperature in these tanks could be more than a few degrees different from the ambient temperature. Even if water at the T/S minimum temperature of 37°F were added to an accumulator tank to raise the water level from the low alarm setpoint to the high alarm setpoint value, the overall water temperature in that one particular tank decreases by only about 5°F. This impact would be reduced when spread over all four accumulators present within the SNUPPS plant. Overall, differences between ambient temperature and accumulator water temperature will be small and primarily the result of small day-to-day variations in plant operation.

Moveover, changes in postulated accumulator water temperature within a band around 90°F have a minor impact on calculated ECCS performance while gas pressure and water volume have a dramatic impact. The density of water changes very little as temperature fluctuates, so the magnitude and duration of the flow of accumulator water entering the cold leg are almost independent of the water temperature. For as long as accumulator water flow persists, all steam flowing through the intact loops will be condensed; thus there are no reflooding condensation effects from varying accumulator water temperature.

Neither the water flow rate nor the effectiveness of accumulator injection is affected significantly by credible changes in postulated accumulator water temperature. Westinghouse, for overall consistency in the ECCS computation, utilized the same value for initial containment temperature and accumulator water temperature. This position represents the appropriate computational basis in light of the Appendix K containment pressure stipulation and indicates that no Technical Specification limit on accumulator water temperature is necessary or desirable.

10. ECCS Subsystems - $T_{avg} \geq 350°F$ (Section 3/4.5.2, page 3/4 5-4)

This section specifies the surveillance requirements to ensure the operability of the ECCS. Item (a) of this section lists the ECCS valves that are to be locked in position (either open or closed) with power to the valve operators removed. Item (b.2) specifies a 31 day requirement to verify that any manual valve that is not locked, sealed, or otherwise secured in position, is in its correct position. This section failed to specify the surveillance requirements for those crucial manual valves that are locked in position in order to verify that they are in their correct position.

In the Callaway safety evaluation report (SER), NUREG-0830, October 1981, the staff identified two crucial manual valves: (1) Valve V011, a 24" valve located on the suction line from the RWST to all the ECCS pump inlets. This valve is locked in the open position. However, if it is inadvertently left in the closed position the entire ECCS is rendered inoperable. (2) Valve 8717, an 8" valve located on the return line from the RHR pumps' discharge and leads back to the RWST. This valve is locked in the closed position. If this valve is inadvertently left open the RHRS flow following a large break LOCA would be less than assumed in the safety analysis because of the flow diversion to the RWST.

The staff requires that the applicant either: (a) propose appropriate surveillance requirements to ensure that neither of the two manual valves is left in an unsafe position, or (b) provide justification for not proposing such a surveillance.

Response:

#10 The SNUPPS design and the procedures employed by the operating utilities provide adequate assurance that the ECCS Pumps will not be rendered inoperable by the undetected closure of V-011, nor will the undetected opening of V8717 render the RHR Pump inoperable in the ECCS alignment by causing excess bypass flow.

System operating procedures for the EM System require independent verification that these valves are initially locked in their required position. Both the initial positioning and independent verification are documented by sign-off in valve checklist (s). The surveillance procedure for the EJ System requires the repositioning and locking of valve 8717 to the closed position and independent verification. Again the positioning and the verification are documented on the valve checklist.

In addition to the standard administrative controls described above for important valves, the SNUPPS design has incorporated a position indicating system fo these valves which:

a) Will cause a component level status light for the respective valves and a system level status light to be illuminated on the Main Control Board if the valve leaves its required position e.g. fully open for V-011 and fully closed for V8717.

b) This condition generates an audible alarm on the Main Control board.

mjd/2a23

Response:

#11 The RHR minimum flow rate has been revised to 3800 gpm. Additionally the surveillance requirements for the RHR system have been revised as on the attached page.

11. ECCS Sybsystems - $T_{avg} \geq 350°F$ (Section 3/4 5.2, page 3/4 5-6)

Item (h.3) of this section specifies the surveillance requirement for the RHR pump flow balance test during shutdown. The surveillance, as specified, is satisfied if one RHR pump flow is measured to be equal to or greater than 2848 gpm.

Since the safety injection flow assumed in the safety analyses is entered as a function of RCS pressure, it is not clear that the minimum flow specified in the surveillance requirement above is indicative of the RHR pump capacities.

The staff requires that the applicant either: (a) propose appropriate surveillance requirements to ensure the RHR pump capacities, or (b) provide justification for the adequacy of the existing surveillance requirements.

SURVEILLANCE REQUIREMENTS (Continued)

h. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics and verifying that:

1) For centrifugal charging pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 346 gpm, and

b) The total pump flow rate is less than or equal to 550 gpm.

2) For Safety Injection pump lines, with a single pump running:

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 462 gpm, and

b) The total pump flow rate is less than or equal to ~~650~~ 655 gpm.

i. By performing a flow test, during shutdown, following completion of modifications to the RHR subsystems that alter the subsystem flow characteristics and verifying that:

1) For RHR pump lines, with a single pump running:

a) The sum of the injection line flow rates is greater than or equal to 3800 gpm.

b) The total pump flow rate is less than or equal to 5500 gpm.

12. Turbine Cycle (Section 3/4.7.1, page 3/4 7-4)

This section specifies the minimum acceptable operating conditions for the auxiliary feedwater system. The surveillance requirements for this section require that each motor driven pump be tested to verify that at least a discharge pressure of at least 1535 psig on recirculation flow is development.

The staff is concerned that information obtained from the above surveillance test may not provide assurance that the motor driven pump is capable of delivering a flow that is at least equivalent to what is assumed in the safety analyses.

The staff requires the applicant to provide in the bases a justification for why a 1535 psig discharge pressure is indicative of adequate pump flow.

Response:

#12 Since the SNUPPS design incorporates a flow restricting orifice in the recirculation flow path, it is not necessary to specify both a discharge pressure and flow to ensure the pump will deliver the necessary flow to the steam generators. The attached insert for Bases 3/4.7.1.2 will be submitted for clarification.

mjd/2a24

SAFETY VALVES (Continued)

109 = Power Range Neutron Flux-High Trip Setpoint for four loop operation,

* = Maximum percent of RATED THERMAL POWER permissible by P-8 Setpoint for three loop operation. This value left blank pending NRC approval of three loop operation,

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

Y = Maximum relieving capacity of any one safety valve in lbs/hour.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

SEE INSERT

~~Each electric motor-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 575 gpm at a pressure of 1221 psig to the entrance of the steam generators.~~ The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 1145 gpm at a pressure of 1221 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the RHR System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 4 hours with steam discharge to the atmosphere concurrent with total loss-of-offsite power and then a cooldown to 350°F at 50°F per hour. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 SPECIFIC ACTIVITY

The limitations on Secondary Coolant System specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of a steam line rupture. This dose also includes the effects of a coincident 1 gpm reactor to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

Insert for B 3/4.7.1.2

Change second paragraph first sentence to read:

Testing of each electric motor-driven auxiliary feedwater pump on a fixed orifice recirculation flow and ensuring a discharge pressure of greater than or equal to 1535 psig verifies the capability of each pump to deliver a total feedwater flow of 575 gpm at a pressure of 1221 psig to the entrance of the steam generators.

mjd/2a25

13. <u>Safety Valves, Bases</u> (Section 3/4.4.2, page B 3/4 4-2)

The bases 3/4.4.2 indicate that the pressurizer code safety valves are sized based on the reactor trip on the first reactor trip signal. The bases should be modified to indicate that the pressurizer code safety valves are sized based on the reactor trip on the second safety grade reactor trip signal per SRP Section 5.2.2.

During the April 13, 1983 meeting, the applicant agreed to verify the design basis of the code safety valves and resolve this issue.

Response:

#13 The sizing of the pressurizer safety valves, as stated in the SNUPPS overpressure Protection Report, is based upon a loss of load event assuming no reactor trip or power-operated relief valves.

The Loss of Load Analysis in FSAR Section 15.2 is the limiting transient with respect to overpressure. This analysis verifies that the safety valve capacity (already sized) is sufficient to maintain reactor coolant system pressure below 110% of design pressure (2750 psia). In this analysis, no credit is taken for a reactor trip from the second reactor trip setpoint signal which is either overtemperature delta T or high pressurizer pressure.

See the attached revised technical specifications.

mjd/2a26

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip ~~until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load),~~ and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12-hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

14. Bases, Pressure/Temperature Limits (Section 3/4.4.9, page B 3/4 4-6)

Bases 3/4.4.9, entitled "Pressure/Temperature Limits," which start on page B 3/4 4-6, discuss how the heatup and cooldown rate limits are calculated to keep the plant in conformance with Appendix G.

The staff notes that the details of how the PORV setpoint curve (Figure 3.4-4) is calculated are not provided.

During the April 13 meeting, the applicant agreed to include details that show how the PORV curve is calculated so that the Appendix G limits will not be exceeded. This is acceptable. Provided the applicant proposes Technical Specification modifications to satisfy the above staff position, we conclude that this item is resolved.

Response:

#14 The attached change to the bases explains the calculation of the PORV setpoint curve (fig. 3.4-4).

mjd/2a27

COLD OVERPRESSURE (Continued)

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for, 1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; 2) a 50°F heat transport effect made possible by the geometrical relationship of the RHR suction line and the RCS wide range temperature indicator used for COMS; 3) instrument uncertainties; and 4) single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, technical specifications require lockout of both safety injection pumps and all but one centrifugal charging pump while in MODES 4, 5 and 6 with the reactor vessel head installed and disallow start of a RCP if secondary temperature is more than 50°F above primary temperature. Exception to these mode requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only one centifugal charging pump OPERABLE and no safety injection pumps OPERABLE is allowed for up to 4 hours. As shown by analysis, LOCA's occuring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single RHR pump with no credit for accumulator injection. Given the short time duration that the condition of having only one centrifugal changing pump OPERABLE is allowed and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging and safety injection pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic safety injection actuation signals except Containment Pressure - High are blocked. In normal conditions a single failure of the ESF actuation circuitry will result in the starting of at most one train of safety injection (one centrifugal charging pump, and one safety injection pump). For temperatures above 325°F an overpressure event occuring as a result of starting two pumps can be successfully mitigated by operation of both PORV's without exceeding Appendix G limits. Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains of safety injection during this 4 hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

mjd/1al

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HEATUP (Continued)

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

COLD OVERPRESSURE

The OPERABILITY of two PORVs, two RHR suction relief valves or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 368°F. Either PORV or RHR suction relief valve has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water-solid RCS.

RHR RCS suction isolation valves 8701A and B are interlocked with an "A" train wide range pressure transmitter and valves 8702A and B are interlocked with a "B" train wide range pressure transmitter. Removing power from valves 8701B and 8702A, prevents a single failure from inadvertently isolating both RHR suction relief valves while maintaining RHR isolation capability for both RHR flow paths.

In addition to opening RCS vents to meet the requirement of 3.4.9.3.c, it is acceptable to remove a pressurizer code safety valve, open a PORV block valve and remove power from the valve operator in conjunction with disassembly of a PORV and removal of its internals, or otherwise open the RCS.

COLD OVERPRESSURE (Continued)

Although COMS is required to be OPERABLE when RCS temperature is less than 368°F, operation with all centrifugal charging pumps and both safety injection pumps OPERABLE is acceptable when RCS temperature is greater than 350°F. Should an inadvertent safety injection occur above 350°F a single PORV has sufficient capacity to relieve the combined flow rate of all pumps. Above 350°F one RCP and all pressurizer safety valves are required to be OPERABLE. Operation of a RCP eliminates the possibility of a 50°F difference existing between indicated and actual RCS temperature as a result of heat transport effects. Considering intrument uncertainties only, an indicated RCS temperature of 350°F is sufficiently high to allow full RCS pressurization in accordance with Appendix G limitations. Should an overpressure event occur in these conditions the pressurizer safety valves provide acceptable and redundant overpressure protection.

The Maximum Allowed PORV setpoint for the Cold Overpressure Mitigation System will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR 50, Appendix H and in accordance with the schedule in Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

15. <u>Several Technical Specification Sections</u>

The different sections in the Technical Specifications state limits on minimum and/or maximum values of process variables, e.g., temperature, pressure, flow rates, levels, and volumes.

The staff is concerned that (a) The process variable limits discussed above are not, in all cases, used in the safety analyses. This concern is discussed under item 15 above and (b) that the analyses assumptions do not always start from the specified values in the Tech Specs after adding an instrumentation error and uncertainty allowance. For example, Section 3.2.5 specifies the maximum T_{ave} to be $\leq$ 595°F during Mode 1. Additionally the Callaway FSAR states that the temperature error used in the safety analyses is $\pm$6.5°F. Therefore, the staff believes that if the T_{ave} limit in the Tech Specs is $\leq$595°F, then the safety analyses where the higher temperature is more limiting should assume a T_{ave} value of 595° $\pm$ 6.5 or 601.5°F as an initial condition.

The staff requires that the applicant: (1) provide justification for not assuming in the safety analyses steady state conditions that are consistent with the limits specified in the Tech Specs after adding a conservative uncertainty margin, and (2) provide a discussion in the bases for choosing the uncertainty margin. The applicant should make a distinction between the value of the parameter as measured and limited by the Tech Specs and the value of the parameter as assumed in the safety analyses.

Response:

#15 As described in the response to RSB question 6, the Accident Analysis assumes event initiation from nominal conditions with allowances for uncertainties such as measurement error and control dead band. These nominal conditions are maintained by automatic control systems such that deviation from the nominal operating points are limited to within the allowance bands. For the reasons discussed in the response to question 6 it is not necessary to add to the Technical Specifications restrictions on all process variables used in the Safety Analysis. Where the Technical Specifications do contain restrictions on process variables the specified limiting values are typically actual values, that is either design values or those used in the analysis, without additional allowances for measurement uncertainty. Where it is necessary to consider measurement uncertainty, the Technical Specifications specifically address (with the exception of RCS TAVG and Pressurizer Pressure) the manner in which uncertainties are considered. In the case of RCS TAVG and Pressurizer Pressure the attached Technical Specification change is hereby proposed to clarify this situation.

All values in the Technical Specifications other than those whose uncertainties are specifically specified whether analytical, design, etc. may be treated as indicated values without regard for instrument uncertainties. This is acceptable because of the relatively small magnitude of typical measurement uncertainties (one to two percent of calibrated span) when compared to the conservatisms included in the plant design and safety analysis. These measurement uncertainties are maintained small by conformance to the Operating Quality Assurance Program which includes requirements for controls of Measuring & Test Equipment, Documents, Design, Test and Inspection, and Procedures. Small deviations in tank levels or pressures, pump flow or discharge pressure, etc. resulting from measurement uncertainty are negligible considering the conservatisms upon which the "limiting" values are based.

By the methods described above the operator can compare indicated values (unless allowances for measurement uncertainties are specified) to those values in the Technical Specifications to ensure compliance, thereby eliminating the use of intermediate documents to account for measurement uncertainties.

JHR/mjd/2a7

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS Four Loops in Operation	 Three Loops in Operation
Indicated Reactor Coolant System T_{avg}	$\leq$ ~~595°F~~ 592.5°F #	**
Indicated Pressurizer Pressure	$\geq$ ~~2220 psia*~~ $\geq$ 2220 psig ##	**

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

**These values left blank pending NRC approval of three loop operation.

This value assumes 2.5°F instrument error.

This value assumes 15 psig instrument error.

DRAFT

16. Relief Valves (Section 3/4.4.9, page 3/4 4-10)

It is the staff's understanding that your steam generator tube rupture analysis presented in Chapter 15 of your FSAR relied on the availability and operability of the PORV in order to limit offsite doses to within 10 CFR 100 guideline values. Similarly, your cooldown evaluation in FSAR section 5.4.7 performed to show compliance BTP RSB 5-1 relied on the availability and operability of the PORV to provide the necessary depressurization function. Your proposed technical specifications however, appear to be inconsistent with your FSAR assumptions in that they allow the PORV to be taken out of service for an indefinite period of time. Please demonstrate how you comply with the requirements of 10 CFR 50.36 regarding how your technical specifications for the PORV were derived from the FSAR safety analyses. Specifically, we believe it is necessary to show that the steam generator tube rupture criteria and the RSB 5-1 criteria can be met assuming an inoperable PORV consistent with your proposed technical specification, or you should otherwise demonstrate that your technical specification is consistent with the FSAR analyses.

Response:

#16 In order to incorporate the RSB concern in meeting the assumptions made in the FSAR for a design basis steam generator tube rupture (SGTR) accident (in which the RCS and damaged SG pressures are capable of being equalized within 30 minutes), and the cooldown evaluation in FSAR section 5.4.7, the attached revision to the SNUPPS Technical Specification is submitted.

JHR/mjd/2a28

DRAFT

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.#

ACTION:

See following page

~~a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~b. With one or more block valve(s) inoperable, within 1 hour: restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

~~c. The provisions of Specification 3.0.4 are not applicable.~~

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION a. in Specification 3.4.4.

With all RCS cold leg temperatures above 368°F

ACTION:

a. With one or more PORV(s) inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With both PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore each of the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

d. With one or more block valve(s) inoperable, within 1 hour: 1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and 2) apply the ACTION of b or c above, as appropriate for the isolated PORV(s).

e. The provisions of Specification 3.0.4 are not applicable.