SNUPPS

Standardized Nuclear Unit Power Plant System

5 Choke Cherry Road
Rockville, Maryland 20850
(301) 869-8010

Nicholas A. Petrick
Executive Director

May 16, 1984

SLNRC 84-0082 FILE: 0543/278
SUBJ: SNUPPS Technical Specifications
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 (B.J. Youngblood) letter to SNUPPS (D.F. Schnell) and G.L. Koester) dated 5/14/84, Request for Additional Information.
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3. NRC (B.J. Youngblood) letter to SNUPPS (D.F. Schnell) and G.L Koester) dated 5/7/84, Request for Additional Information.

Dear Mr. Denton:

Attached hereto are SNUPPS responses to the questions forwarded by reference 1. These responses were discussed with Reactor Systems Branch (RSB) on May 2 and May 15, 1984.

In addition, the response to question 13 of reference 1 is also being resubmitted to correct typographical errors in the original.

Very truly yours,

Nicholas A. Petrick for

JHR/mjd/2a30
Attachments

B001
1/1

cc:	D. F. Schnell	UE
	D. T. McPhee	KCPL
	G. L. Koester	KGE
	J. Neisler/B. Little	USNRC/CAL
	W. Schum/A. Smith	USNRC/WC
	B. L. Forney	USNRC/RIII
	E. H. Johnson	USNRC/RIV
	V. Stello	USNRC

bcc:	F. D. Crawford	KCPL
	D. W. Capone	UE
	M. L. Johnson	KGE
	J. A. Bailey	KGE/WC
	M. A. Hannah	KGE
	D. E. Shafer	UE
	A. C. Passwater	UE
	S. E. Miltenberger	UE/CAL
	F. T. Rhodes	KGE/WC
	W. Robinson	UE/CAL
	J. A. Zell	KGE/WC
	J. H. Smith	B
	P. A. Ward	B
	L. R. Benson	W
	R. Jansen	W
	J. O. Cermak	Staff
	F. Schwoerer	Staff

Attachment 1

Response to RSB

Questions 17 and 18

17. Reactor Trip System Instrumentation (Section 3/4.3.1, Table 3.3-1 page 3/4 3-2)

This table specifies the minimum number of reactor trip system instrumentation channels operable and the applicable modes of their operation.

Item 6.c on this table specifies that only one Source Range Monitor (SRM) channel is required to be operable during Modes 3, 4 and 5. During these modes the SRM does not provide a reactor trip function. However, it provides a boron dilution mitigation function by sensing the increase in neutron flux, and upon flux doubling it initiates the valve motion that isolates the unborated water sources and aligns the borated water source (RWST) to the charging pumps suction. One operable SRM represents a single point of vulnerability for the boron dilution mitigation system (BDMS). During the FSAR review stage, the applicant committed to install BDMS identical in design to that of the Comanche Peak Steam Electric Station (CPSES). The applicant also committed to meet the same design criteria and standards as those for the CPSES system. The CPSES BDMS, as reviewed and approved by the staff, is a fully redundant system which relies on two SRMs powered by two different class 1E buses. Allowing only a single channel of source range monitor for Boron Dilution mitigation is a deviation from the FSAR analysis assumptions as reviewed approved by the staff and should be corrected.

Response:

#17 The attached Technical Specification has been revised as shown to satisfy the RSB concern (see item 6 in table 3.1-1 and ACTION 5).

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2	1
	2	1	2	3*, 4*, 5*	10
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2#
b. Low Setpoint	4	2	3	1###, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1###, 2	3
6. Source Range, Neutron Flux					
a. Startup	2	1	2	2##	4
b. Shutdown	2	1	2	3, 4, 5	5
7. Overtemperature ΔT					
a. Four Loop Operation	4	2	3	1, 2	6#
b. Three Loop Operation	**	**	**	**	**
8. Overpower ΔT					
a. Four Loop Operation	4	2	3	1, 2	6#
b. Three Loop Operation	**	**	**	**	**
9. Pressurizer Pressure-Low	4	2	3	1	6#
10. Pressurizer Pressure-High	4	2	3	1, 2	6#

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers, suspend all operations involving positive reactivity changes and verify Valves BG-V178 and BG-V601 are closed and secured in position within the next hour.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER 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 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 1 hour.

18. Special Test Exceptions, Reactor Coolant Loops (Section 3/4.10.4, page 3/4 10-4)

This technical specification permits plant operation without any reactor coolant pumps operating up to 10% thermal power on fission heat for startup or physics tests. The staff is unaware of any safety analysis that demonstrates that transients or accidents initiated from this operating condition would be acceptable. Both the steady state and transient reactor coolant system temperature profiles, margin to saturation, core DNBR, and other related thermal-hydraulic and fuel performance parameters, particularly thermal-hydraulic stability, should be assessed. The acceptability of the reactor protective system setpoints during various transients and accidents initiated from this condition must also be justified.

Response

#18 The Westinghouse safety evaluations written for plants which performed post TMI natural circulation tests were based on analysis in which protective action did not occur until 10% power. Thus, it is permissable to have the pumps off and be less than 10% power for these tests. The plants for which Westinghouse submitted these evaluations were Sequoyah, North Anna, Salem, and Diablo Canyon. After several of these tests were performed the NRC agreed that these evaluations were shown to be generic and were no longer necessary to ensure safe operation during the test. Therefore it is the SNUPPS position that this specification is satisfactory as written.

The natural circulation test will be performed at 3 percent power as stated in the FSAR test abstract (section 14.2.12.3.41).

JHR/mjd/1b30

Attachment 2

Corrected Response to RSB

Question 13

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 and/or Turbine Trip 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 a turbine trip signal. Credit is taken for 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.

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