MARKED-UP TECHNICAL SPECIFICATIONS PAGES
(NUREG 1399)

Attachment 3 to TXX-92116
3/4 3-1 through 3/4 3-12

NO CHANGE
ON THIS PAGE

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific Reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

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[a], 4[a], 5[a]	9
2. Power Range, Neutron Flux					
a. High Setpoint	4	2	3	1, 2	2
b. Low Setpoint	4	2	3	1[c], 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[c], 2	3
6. Source Range, Neutron Flux					
a. Reactor Trip and Indication					
a. 1) Startup	2	1	2	2[b]	4
b. 2) Shutdown	2	1	2	3, 4, 5	5
b. Boron Dilution Flux Doubling	2	1	2	3[h], 4, 5	5
7. Overtemperature N-16	4	2	3	1, 2	12
8. Overpower N-16	4	2	3	1, 2	12
9. Pressurizer Pressure--Low	4	2	3	1[d]	6[e]
10. Pressurizer Pressure- High	4	2	3	1, 2	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1[d]	6
12. Reactor Coolant Flow--Low					
a. Single Loop	3/loop	2/loop in any loop	2/loop	1[f]	6
b. Two Loops	3/loop	2/loop in any two loops	2/loop	1[g]	6
13. Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen. in any stm. gen.	3/stm. gen.	1, 2	6[e]
14. Undervoltage--Reactor Coolant Pumps	4-1/bus	2	3	1[d]	6
15. Underfrequency--Reactor Coolant Pumps	4-1/bus	2	3	1[d]	6
16. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1[i]	6
b. Turbine Stop Valve Closure	4	4	4	1[i]	10
17. Safety Injection Input from ESFAS	2	1	2	1, 2	8

NO CHANGE ON THIS PAGE

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2^{b}	7
b. Low Power Reactor Trips Block, P-7					
1) P-10 Input	4	2	3	1,2	7
2) P-13 Input	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-9	4	2	3	1^{i}	7
e. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
19. Reactor Trip Breakers	2	1	2	1, 2	8, 11
	2	1	2	3^{a}, 4^{a}, 5^{a}	9
20. Automatic Trip and Interlock Logic	2	1	2	1, 2	8
	2	1	2	3^{a}, 4^{a}, 5^{a}	9

NO CHANGE ON THIS PAGE

TABLE 3.3-1 (Continued)

TABLE NOTATIONS

[a]Only if the reactor trip breakers happen to be in the closed position and the Control Rod Drive System is capable of rod withdrawal.

[b]Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

[c]Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

[d]Above the P-7 (At Power) Setpoint

[e]The applicable MODES and ACTION statements for these channels noted in Table 3.3-2 are more restrictive and therefore, applicable.

[f]Above the P-8 (3-loop flow permissive) Setpoint.

[g]Above the P-7 and below the P-8 Setpoints.

[h]The boron dilution flux doubling signals may be blocked during reactor startup.

[i]Above the P-9 (Reactor trip on Turbine trip Interlock) Setpoint.

[h]Deleted

ACTION STATEMENTS

ACTION 1 - 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 be in HOT STANDBY within the next 6 hours.

ACTION 2 - 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 6 hours,

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range Neutron Flux Trip Setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours per Specification 4.2.4.2.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint,

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

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 within the next hour open
and
the reactor trip breakers, suspend all operations involving positive reactivity changes. and verify either valve 1CS-8455 or valves 1CS-8560, FCV-111B, 1CS-8439, 1CS-8441, and 1CS-8453 are closed and secured in position, and verify this position at least once per 14 days thereafter. With no channels OPERABLE complete all the above actions within 4 hours. and verify the positions of the above valves at least once per 14 days thereafter.

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 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

ACTION 7 - 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.

NO CHANGE
ON THIS PAGE

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 8 - 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 or maintenance, provided the other channel is OPERABLE.

ACTION 9 - 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 10 - 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 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status, during which time ACTION 8 applies.

ACTION 12 - 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 6 hours, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing per Specifications 4.3.1.1 or 4.2.5.4.

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(14)	N.A.	1, 2, 3[a], 4[a], 5[a]
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1[c], 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1[c], 2
6. Source Range, Neutron Flux	S	R(4, 13)	S/U(1), Q(9)	N.A. ~~R(12)~~	N.A.	2[b], 3, 4, 5
7. Overtemperature N-16	S	D(2, 4) M(3, 4) Q(4, 6) R(4, 5)	Q	N.A.	N.A.	1, 2
8. Overpower N-16	S	D(2, 4) R(4, 5)	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R	Q(8)	N.A.	N.A.	1[d]
10. Pressurizer Pressure--High	S	R	Q	N.A.	N.A.	1, 2

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. Pressurizer Water Level--High	S	R	Q	N.A.	N.A.	1^d
12. Reactor Coolant Flow--Low	S	R	Q	N.A.	N.A.	1^d
13. Steam Generator Water Level--Low-Low	S	R	Q(8)	N.A.	N.A.	1, 2
14. Undervoltage - Reactor Coolant Pumps	N.A.	R	N.A.	Q(10)	N.A.	1^d
15. Underfrequency - Reactor Coolant Pumps	N.A.	R	N.A.	Q	N.A.	1^d
16. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1^e
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1^e
17. Safety Injection Input from ESFAS	N.A.	N.A.	N.A.	R	N.A.	1, 2
18. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2^b

NO CHANGE
ON THIS PAGE

NO CHANGE ON THIS PAGE

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
18. Reactor Trip System Interlocks (Continued)						
b. Low Power Reactor Trips Block, P-7						
1) Power Range Neutron Flux P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
2) Turbine First Stage Pressure P-13	N.A.	R	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
d. Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1
d. Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, 11)	N.A.	1, 2, 3[a], 4[a], 5[a]
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7)	1, 2, 3[a], 4[a], 5[a]
21. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(15), R(16)	N.A.	1, 2, 3[a], 4[a], 5[a]

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

[a]Only if the reactor trip breakers happen to be in the closed position and the Control Rod Drive System is capable of rod withdrawal.

[b]Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

[c]Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

[d]Above the P-7 (At Power) Setpoint.

[e]Above the P-9 (Reactor trip on Turbine trip Interlock) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power and N-16 power indication above 15% of RATED THERMAL POWER. Adjust excore channel and/or N-16 channel gains consistent with calorimetric power if absolute difference of the respective channel is greater than 2%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1 or 2.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. For the purpose of these surveillance requirements, "M" is defined as at least once per 31 EFPD. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1 or 2.

(4) Neutron and N-16 detectors may be excluded from CHANNEL CALIBRATION.

(5) Detector plateau curves shall be obtained and evaluated. For the Intermediate Range Neutron Flux, Power Range Neutron Flux and N-16 channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 1 or 2.

(6) Incore - Excore Calibration, above 75% of RATED THERMAL POWER. For the purpose of these surveillance requirements "Q" is defined as at least once per 92 EFPD. The provisions of Specification 4.0.4 are not applicable for entry into MODE 1 or 2.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) The MODES specified for these channels in Table 4.3-2 are more restrictive and therefore applicable.

(9) Quarterly surveillance in MODES 3[a], 4[a], and 5[a] shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Quarterly surveillance shall include verification of the Boron Dilution Alarm Setpoint of less than or equal to an increase of twice the count rate within a 10-minute period. e

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the reactor trip breakers.

(12) At least once per 18 months during shutdown, verify that on a simulated Boron Dilution Flux Doubling test signal the normal CVCS discharge valves close and the centrifugal charging pumps suction valves from the RWST open.

(13) With the high voltage setting varied as recommended by the manufacturer, an initial discriminator bias curve shall be measured for each detector. Subsequent discriminator bias curves shall be obtained, evaluated and compared to the initial curves.

(14) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(15) Local manual shunt trip prior to placing breaker in service.

(16) Automatic undervoltage trip.

(12) Deleted

WPT-14385, "NOTIFICATION OF NONCONSERVATIVE
BORON DILUTION ANALYSIS INPUT ASSUMPTIONS"
FEBRUARY 12, 1992

Enclosure 1 to TXX-92116

Westinghouse Electric Corporation
Energy Systems
Box 355
Pittsburgh Pennsylvania 15230-0355

Mr. W. J. Cahill, Jr., Executive Vice President
Nuclear Engineering & Operations
TU Electric Company
P. O. Box 1002
Glen Rose, Texas 76043

February 21, 1992

S.O. No. TBX/TCX-4708

(No Response Required)

Attention: S. M. Maier

TU ELECTRIC COMPANY
COMANCHE PEAK STEAM ELECTRIC STATION
UNIT NUMBER 1 & 2
NOTIFICATION OF NONCONSERVATIVE BORON DILUTION ANALYSIS INPUT ASSUMPTIONS

References:
1. CPSES-9204091, "CPSES Units 1 (sic) ICRR vs. Boron Concentration Data for Modes 3, 4, and 5 Boron Dilution Event Analysis," February 4, 1992
2. Standard Review Plan, NUREG-0800, Section 15.4.6, Rev. 1 - July 1981
3. NRC Generic Letter 85-05, "Inadvertent Boron Dilution Events," January 31, 1985
4. NS-TMA-2273, "Boron Dilution Concerns at Cold and Hot Shutdown," July 8, 1980

Dear Mr. Cahill:

Introduction

Based on recent conversations between Westinghouse Transient Analysis and Texas Utilities Electric (TUE), various nonconservatisms have been identified related to the input assumptions/boundary conditions (ICRR data and flux-multiplication setpoint) in the analyses of the licensing-basis boron dilution event. The Modes 3, 4 and 5 licensing-basis boron dilution event analyses for the Comanche Peak plant utilizes inverse-count-rate-ratio (ICRR) data and a flux-doubling (2ϕ) alarm to determine both the time of the 2ϕ alarm and the time from the alarm to loss of plant shutdown margin following an inadvertent dilution event.

Based on information received from TUE, the Comanche Peak licensing-basis boron dilution event analyses (FSAR Section 15.4.6) are no longer bounding. In addition, the safety analysis value for the flux-doubling signal/alarm in the plant Technical Specifications may no longer be valid with the current analysis methods.

Summary of the Issue

The ICRR data used in the Comanche Peak analyses of the inadvertent boron dilution event is based on plant data which was the most limiting of all the data available in the late 1970s. This data, along with a nominal 2ϕ flux-multiplication setpoint, provided the basis for the detection mechanism following an inadvertent boron dilution event. The flux-doubling signal actuates the protective function of the boron dilution mitigation system (BDMS) to isolate the dilution source and initiate a reboration of the RCS. TUE has provided Westinghouse with Comanche Peak Unit 1 plant-specific ICRR data (Reference 1) that is not bounded by the most-limiting known data from the 1970s. Furthermore, the methodology is no longer conservative with respect to the 2ϕ setpoint which includes no instrumentation uncertainties that ought to be applied to produce an equivalent "trip setpoint" as presented in the Technical Specifications.

The effect on the Comanche Peak licensing basis is that the inadvertent boron dilution analyses for Modes 3, 4 and 5 are no longer bounding since Comanche Peak Unit 1 has demonstrated a worse characteristic for the ICRR data than was used in the analysis to determine the times of the flux-doubling signal. Preliminary analyses using the Comanche Peak plant-specific ICRR data and a revised, more-conservative flux-multiplication setpoint do not yield acceptable results.

Justification for Continued Operation (JCO)

The purpose of the FSAR Section 15.4.6 analyses results is to show that the acceptance criterion delineated in the Standard Review Plan (Reference 2) is satisfied. The acceptance criterion is that the reactor should not be diluted to the point of total loss of plant shutdown margin. With respect to the generic implications of this issue, it should be noted that it has already been determined that this is not a safety problem. The specific concern is that of a return to criticality in Mode 3, 4 or 5 and the resulting potential challenge to the integrity of the RCS due to a pressure increase. Thus, safety analyses have traditionally been performed to show that plant shutdown margin is not lost and that criticality does not occur. The NRC has reiterated the position (Reference 3) that the criteria in the Standard Review Plan (Reference 2) remain valid. Based on safety analysis performed by the NRC in which a return to criticality was modeled, the noted generic letter (Reference 3) was written to address Generic Issue 22, *Inadvertent Boron Dilution Events*. This generic letter states that power excursions are possible due to a boron dilution event but that the excursion should be self limiting. The NRC analysis which supports the information in the generic letter indicate that the self-limiting boron dilution power excursions are not expected to exceed the overpressurization criterion delineated in Reference 2.

While the specifics of the current concern of the BDMS at TUE differ from those which eventually resulted in the generic letter (Reference 3), the underlying concern (plant criticality) and the conclusions delineated by the generic letter still apply to TUE's concern today. The use of the BDMS at Comanche Peak serves to provide a means to detect a dilution event and automatically initiate corrective action in most instances. While not specifically documented, there are possible situations in which a flux-doubling signal would not detect an inadvertent dilution; e.g., some small dilution flowrate in which the flux increases so slowly that the BDMS microprocessor could not produce a 2ϕ signal, and operator awareness is required for preventing criticality. However, it should be noted that even if criticality does occur, the BDMS will have actuated automatic reboration and the plant will return to a subcritical condition event with no operator action. The Comanche Peak BDMS provi‹ : a diverse mitigation feature for the boron dilution event as compared to plants not equipped with such an automatic system.

Available Compensating Actions

Since the Comanche Peak BDMS may not detect a dilution event and automatically initiate corrective action in all instances to prevent criticality, enhanced awareness by the operator is recommended during activities where a potential dilution situation could exist.

The high-flux-at-shutdown alarm (HFSA) provides an indication of the amount of neutrons being emitted into containment at any given time. By lowering the setpoint on the HFSA instrumentation, an early warning of an unexpected flux increase could be available to the operators when in the shutdown modes of plant operation. The operators need to be cognizant of the fact that electronic noise affects the HFSA instrumentation and spurious alarms at a reduced setpoint (from the normal 5ϕ setting) could occur.

Increased surveillance may be considered on those valves in the CVCS which are along the potential dilution paths. Awareness of the sources and possible paths of an inadvertent boron dilution would greatly enhance the operator's chances of early mitigation of the event. Likewise, the locking out of the valves along possible dilution paths when planned RCS dilution is not occurring would greatly decrease the chances of an inadvertent dilution.

Administrative controls may be implemented which place limits on the assumed dilution flowrate and RCS boron concentration such that there is sufficient time for operator action following an inadvertent dilution event. Originally developed in 1980 in response to an NRC concern for a boron dilution event while operating the plant on RHR (Modes 4 and 5), the operating procedures (Reference 4) have been implemented in instances where plant-specific and mode-specific analyses have not been performed and documented in the FSAR. Recently revised, the procedures allow greater Mode 4 and Mode 5 operational flexibility. The key improvements in the revised procedures over the original ones (Reference 4) are summarized below.

1. The revised curves are more likely to remain unchanged as a result of the fuel cycle reload verification process due to

   a. removal of the dependency on a boron worth assumption, and
   b. removal of the dependency on a specific range of RCS or critical boron concentrations through the introduction of the ratio of the two.

2. The revised curves are specified as a function of RHR flowrate and, therefore, support greater Modes 4 and 5 operational flexibility.

3. The revised curves are based on a Mode-5 RCS volume assumption corresponding to the vessel drained down to the hot-leg midloop level, also supporting greater cold shutdown operational flexibility.

4. The procedure is based on quantities typically known and readily available to the plant operator: RCS boron concentration and critical boron concentration.

It should be noted that these procedures are designed to allow at least 15 minutes for operator action from the beginning of the dilution event, not from the time of an assumed alarm, until loss of shutdown margin. In Mode 3, the volume available for dilution is much larger than in Mode 4 or 5; therefore, it is expected that 15 minutes is also available from the beginning of the dilution for the operator to detect the event.

The Reference-4 letter and operating procedures are found in Attachment 1; the revised procedures, recommended for use by TUE, are found in Attachment 2.

Any or all of these actions, as a supplement to the automatic BDMS, lead towards a reduction in the probability of a boron dilution event and/or a reduction in the probability that the reactor would reach criticality should an inadvertent dilution event occur.

TUE should also be aware that Westinghouse has communicated the boron dilution analysis concerns to the WOG-RRG in a telecon on February 19, 1992. Westinghouse will continue to have discussions with the WOG on this subject.

If there are any questions please contact Melita Osborne on 412-374-4481.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

C. Benton for

R. H. Owoc

J. L. Vota, Manager
Comanche Peak Projects

WPT-14385
ET-NSL-OPL-II-92-095

cc:	W. J. Cahill, Jr.	1L, 1A
	CCS	1L, 1A, 1AR
	S. C. Wood	1L, 1A
	VETIP Coordinator	1L, 1A
	T. A. Hope	1L, 1A
	W. G. Guldemond	1L, 1A
	L. Terry	1L, 1A
	S. M. Maier	1L, 1A
	D. Woodlan	1L, 1A
	D. Throckmorton	1L, 1A

Attachment 1

Letter NS-TMA-2273 - Original (1980) Operating Procedures

Westinghouse Electric Corporation Power Systems

PWR Systems Division

Box 355
Pittsburgh Pennsylvania 15230

July 8, 1980

NS-TMA-2273

Mr. Victor Stello
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, MD 20014

SUBJECT: Boron Dilution Concerns at Cold and Hot Shutdown

Dear Mr. Stello:

On June 27, 1980, Ed Jordan of your staff was notified by Westinghouse of an Unreviewed Safety Question under 10CFR50.59. This notification concerned the potential for an inadvertent boron dilution event while shutdown and operating on the Residual Heat Removal System. Attachment 1 is the text of the written notification supplied to our customers on July 8, 1980 which outlines potential Westinghouse concerns and the basis for recommended interim actions which address these concerns. These interim actions are somewhat modified from those previously reported. If there are any questions regarding the attached, please contact D. W. Call at 412/373-5074.

Very truly yours,

T. M. Anderson, Manager
Nuclear Safety Department

Attachment

cc: E. Jordan
R. Woods

~~8007210410~~ 7pp.

FEB 21 '92 15:24

ATTACHMENT 1

On June 27, 1980, you were notified of certain Westinghouse concerns and recommended actions regarding the potential for an inadvertent boron dilution event at cold or hot shutdown conditions while on the Residual Heat Removal System. This notification was in accord with Westinghouse determination that these concerns constitute an Unreviewed Safety Question under 10CFR Part 50.59. The NRC Office of Inspection and Enforcement was also notified on June 27, 1980 that these concerns have generic applicability to Westinghouse-supplied nuclear power plants. Further clarification was made to the NRC Office of Inspection and Enforcement on June 30, 1980 that Westinghouse concerns are not applicable while the plant is greater than 5% shutdown.

This letter is intended to formally document these concerns and to provide additional relevant information. This letter also modifies the earlier recommended actions by a more detailed specification of applicable plant operating conditions.

Inadvertent boron dilution at shutdown has been generally regarded as an event which can be identified and terminated by operator action prior to a return to critical. Automatic protection has not been a standard feature for Westinghouse plants. Westinghouse has recently been conducting a general investigation of this potential event relative to the licensing requirements imposed on newer plants not yet in operation. This investigation is not yet complete. However, it has been determined that under certain shutdown conditions and with certain assumed dilution rates, adequate time for operator action to prevent a return to critical may not be available.

The current Westinghouse evaluations are based on plant conditions as noted below:

1. The Reactor Coolant System effective volume is limited to the vessel and the active portions of the hot and cold legs when on RHR, i.e., steam generator volumes are not included.

2. The plant is borated to a shutdown margin greater than or equal to 1% $\Delta k/k$.

3. Uniform mixing of clean and borated RCS water is not assumed, i.e., mixing of the clean, injected water and the affected loop is assumed but instantaneous, uniform mixing with the vessel, hot legs, and cold leg volumes upstream of the charging lines is not assumed. Thus a "dilution front" moves through the cold legs, downcomer, and lower plenum to the core volume as a single volume front. This results in subsequent decreases in shutdown margin due to dilution fronts moving through the active core region with a time constant equal to the loop transit time when on RHR (five to seven minutes).

If a return to critical occurs as a result of an inadvertent dilution, the following potential concerns have been identified:

1. A rapid, uncontrolled power excursion into the low and intermediate power ranges occurs, resulting in a power/flow mismatch due to the low flow (approximately 1 - 2% of nominal) provided by the RHR pumps.

2. The potential exists for significant system overpressurization. Pressure increases above the RHR cut off head (approximately 600 psig) further accentuate the effects of a power/flow mismatch when all RCS (RHR) flow is lost. An investigation of the adequacy of existing cold overpressurization protection systems is necessary in order to assess the full impact of this potential problem.

3. The potential exists for limited fuel damage. This is not currently a significant concern. Preliminary evaluation indicates that the potential for exceeding DNB limits is low due to the cold initial operating conditions. Further investigation of this problem is underway.

The recommended interim actions to prevent or mitigate an inadvertent boron dilution at shutdown conditions are detailed in Appendix A. If no cocked control rods are required, as specified in Figure A-1, the plant operator has fifteen minutes from the initiation of dilution event to terminate the event before a return to critical occurs. It is the Westinghouse position that a fifteen minute time interval from the initiation of the dilution to the time shutdown margin is lost is sufficient time for operator action. If cocked control rods are required, the source range reactor trip provides positive indication for immediate operator action to terminate dilution.

It is expected that the operator has available the following information for determination that a dilution event is in progress:

1. Source Range Neutron Flux with,

   a. High Flux at Shutdown Alarm set at half a decade above background.

   b. Use of the audible count rate indication to distinguish significant changes in flux, i.e., a doubling of the count rate.

   c. Periodic, i.e., frequent surveillance of the Source Range meters performed by the operator.

2. Status indication of the Chemical and Volume Control System and Reactor Makeup Water System with,

a. Indication of boric acid and blended (total) flow rate, or

b. Indication of boric acid and clean makeup flow rate,

c. CVCS valve position status lights, and

d. Reactor Makeup Water Pump "running" status light.

The operator action necessary upon determination that a dilution event is in progress (by High Flux at Shutdown Alarm, Source Range Reactor Trip, "P-6 Available" indication, high indicated or audible count rates, or make up flow deviation alarms) is:

1. Immediately open the charging/SI pump suction valves from the RWST (that open on receipt of an "S" signal). (For 312 plants these are LCV-115-B, D. For 412 plants these are LCV-112-D, E.)

2. Immediately close the charging/SI pump suction valves from the VCT (that close on receipt of an "S" signal). (For 312 plants these are LCV-115-C, E. For 412 plants these are LCV-112-B, C.)

3. For two-loop plants, immediately open the charging suction valves from the RWST. (For 212 plants these are LCV-113-B and LCV-112-C.) Also immediately close the charging suction valves from the VCT. (For 212 plants these are LCV-113-A and LCV-112-B.)

Through the use of Appendix A and the above noted operator action requirements, Westinghouse is attempting to minimize the operational burden placed on the plant to prevent or mitigate an inadvertent dilution event while maintaining adequate safety margin. Our investigation of this event is continuing. A detailed analytical model of the system response to a dilution event at shutdown conditions is being developed and the potential for system overpressurization and fuel failure will subsequently be assessed. The Westinghouse investigation is expected to be completed by September 15, 1980. We will keep you informed as to the results of our efforts.

APPENDIX A

Figure A-1, attached, provides the shutdown margin requirements as a function of Reactor Coolant System boron concentration and maximum possible dilution flow rate. Prior to use of this figure, the plant must determine the maximum dilution flow rate of all charging pumps not rendered inoperable once the plant is placed on RHR. To cover all modes, it should be assumed that the flow rate is based on pump runout unless there are flow limiting devices in the system (orifices, piping resistances, etc.). The Reactor Makeup Water pump capacity may be limiting in the determination of the maximum possible dilution flow rate.

Figure A-1 notes areas of acceptable operation of three different dilution flow rates as a function of RCS boron concentration and borated shutdown margin (K_{eff}). For a given dilution flow rate, if the RCS boron concentration and shutdown margin result in a point placed to the left of the flow rate line, no control rod bank withdrawal is necessary. If the results place the plant to the right of the line, then either the shutdown margin must be increased such that the plant is moded to the area of acceptable operation, or 1% $\Delta k/k$ in control rods must be withdrawn to provide additional shutdown margin. The tripping of the withdrawn rods provides positive operator indication that a dilution event is in progress and additional time for operator termination of the event. In all cases, a shutdown margin of 5% $\Delta k/k$ ($K_{eff} \leq 0.95$) is considered sufficient for continued operation without a requirement for control rod bank withdrawal.

Figure A-1 is based on best estimate calculations for the "all rods in" configuration. It is recommended that the Westinghouse Nuclear Design Report for your plant be used as a reference in determining the RCS boron concentration with the appropriate conservatism to be used in the figure. The Westinghouse Nuclear Fuel Division is available to provide assistance in meeting the constraints imposed by the Figure A-1 requirements.

Use of Figure A-1 is applicable any time there is boration/dilution capability from the normal boric acid blending system. The above procedure is not required if boration and/or makeup during cold and hot shutdown is performed utilizing water from the RWST. This requires that the normal dilution/boration path is isolated from the charging path. Two means of lockout to isolate the charging path are available:

1. Lock out Reactor Makeup Water Supply.

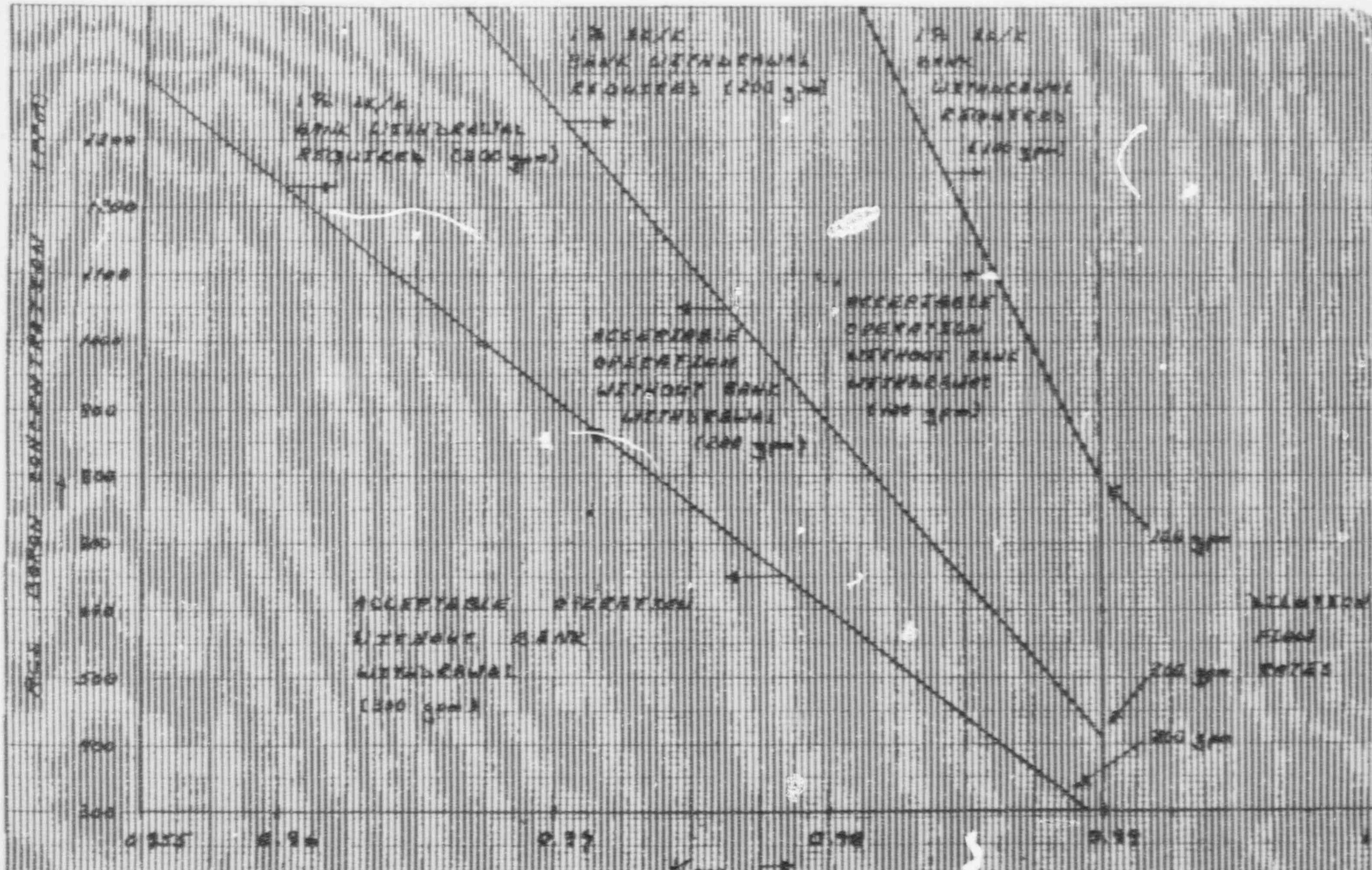
   This is accomplished by valve 8338 for 212 plants, valve 8457 for 312 plants, and valve 8455 for 412 plants.

OR:

2. Lock out valves between the boric acid blender and the VCT.

   These are FCV-111B, FCV-110B, 8339, 8355, and 8361 for 212 plants; FCV-114A, FCV-113B, 8454, 8441, and 8439 for 312 plants; FCV-111B, FCV-110B, 8453, 8441, 8439 for 412 plants.

This recommendation precludes the occurrence of an inadvertent dilution while borating or making up water from the RWST under these conditions.



SHUTDOWN MARGIN REQUIREMENTS AS A FUNCTION OF RCS BORON CONCENTRATION AND DILUTION FLOW RATE

Attachment 2

Revised (1989) Operating Procedures

Revised Interim Procedure

I. Procedure Parameters

1. Residual Heat Removal System (RHRS) flow rate, gpm.
   Range: 1000 to 6000

2. Maximum predicted dilution flow rate (q), gpm.
   Range: 100 to 300

3. Dilution Factor (DLF), [illegible] [illegible]bc/Cbi
   Range: 0.6 to 1.0

4. RCS critical boron conce[illegible] [illegible] all rods inserted minus the most reactive RCCA stu[illegible] [illegible] the core (Cbc), ppm.

5. Required RCS boron concentration to ensure 15 minutes are available from the beginning of a dilution event until loss of shutdown margin for a given RHR flow rate and dilution flow rate (Cbi), ppm.

II. Application Guidelines

1. Determine current or minimum intended RHRS flow rate.

2. Determine maximum predicted dilution flow rate. Use the closest curve corresponding to a value which is greater than or equal to this.

3. Calculate RCS critical boron concentration (Cbc).

4. Use revised figure to find DLF = DLF(q,RHR flow).

5. Calculate Cbi = Cbc/DLF.

6. Ensure RCS boron concentration is ≥ Cbi or 1% delta-k/k in control rods is withdrawn (tripping of withdrawn rods provides positive operator indication of a boron dilution event in progress and provides additional time for operator action). An alternative approach is to further limit the maximum possible dilution flow rate.

III. Limits of Applicability

1. Applicable for Mode 4 (hot shutdown) and Mode 5 (cold shutdown) operation.

2. If the calculated value of Cbi is less than the RCS boron concentration required to meet the Technical Specification minimum shutdown margin requirement, borate to the latter.

3. The RHRS flow rates used for determining Cbi should be the lowest projected flowrates during Mode 4 and 5 operation.

4. The maximum predicted dilution flow rate while operating on RHRS may be limited by the maximum capacity Reactor Makeup Water System to deliver unborated water to the charging pump suction, or it could be limited by the maximum delivery rate of all charging pumps not rendered inoperable. In any case, the assumed flow rate should be based on pump runout unless there are flow limiting devices in the system.

5. Use of the curves ensures availability of time for operator mitigation of an inadvertent boron dilution assuming no cocked rods. Use of the curves is not necessary if boration and/or makeup during hot and cold shutdown is performed utilizing water from the Refueling Water Storage Tank. This requires that the normal dilution/boration path is isolated from the charging path which precludes the occurrence of an inadvertent boron dilution.

6. The minimum RHR flow rate considered is 1000 gpm. The curves must not be extrapolated below this minimum value.

7. The maximum RHR flow rate considered is 6000 gpm. For any flow rate higher than this, use of the curves should be based on 6000 gpm.

8. These curves are derived for a minimum Mode 4 RCS volume assuming the vessel is filled and vented and a minimum Mode 5 RCS volume assuming the water level drained to the mid-plane of the hot leg.

D. E. EISENHUT LETTER OF JANUARY 31, 1985,
GENERIC LETTER 85-05

Enclosure 2 to TXX-92116

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 31, 1985

TO ALL PRESSURIZED WATER REACTOR LICENSEES

FEB 11 1985

Gentlemen:

B. R. CLEMENTS

SUBJECT: INADVERTENT BORON DILUTION EVENTS (Generic Letter 85-05)

The purpose of this letter is to inform each licensee of operating pressurized water reactors of the staff position resulting from the evaluation of Generic Issue 22, "Inadvertent Boron Dilution Events" regarding the need for upgrading the instrumentation for detection of boron dilution events in operating reactors.

A boron dilution event is considered as an anticipated operational occurrence which may occur at moderate frequency. The staff has performed analyses of unmitigated boron dilution events for a typical plant for each pressurized water reactor (PWR) ver r. The staff determined that while power excursions during boron dilution events are possible if the operator does not take any action and sufficient volume of dilution water is available, the excursion should be self-limiting. The staff analyses indicate that these type of boron dilution transients should not exceed the staff's acceptance criteria. However, our analyses also show that a few plants may experience slight overpressurization in excess of the 110% overpressure limit in the Residual Heat Removal system if the event occurs during a particular mode of operation.

In addition, the staff recognizes that many operating plants do not have distinct, positive alarms to alert the operators to boron dilution events but rely on other devices such as audible count rate meters. Other problems include lack of alarm redundancy and lack of technical specifications which would prevent operators from taking alarming devices out of service. The staff also does not consider it prudent to credit operat rs with the ability to recognize a boron dilution event and take the proper mitigative action within specified time limits in the absence of positive boron dilution alarms.

Considering all of the above factors and possible consequences of boron dilution events, the staff has concluded that the criteria in Section 15.4.6 of the Standard Review Plan are adequate and should continue to be applied to plants currently undergoing licensing review. However, the consequences are not severe enough to jeopardize the health and safety of the public and do not warrant backfitting requirements for boron dilution events at operating reactors. The staff will continue to review the analyses of the Boron Dilution Event in reload applications to assure that reasonable confidence is provided that operators can be expected to take the right corrective action using the installed systems.

In summary, while the NRC will not require operating plant backfits for boron dilution events at this time, the staff would regard an unmitigated boron dilution event as a serious breakdown in the licensee's ability to control its plant, and strongly urges each licensee to assure itself that adequate protection against boron dilution events exists in its plants.

~~8502010300~~ 3pp.

This generic letter is provided for information only, and does not involve any reporting requirements. Therefore, no clearance from the Office of Management and Budget is required.

Darrell G. Eisenhut, Director
Division of Licensing

Enclosure:
List of Generic Letters

LIST OF RECENTLY ISSUED GENERIC LETTERS

GENERIC LETTER NO.	SUBJECT	DATE
84-15	Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability	7/2/84
84-16	Adequacy of On-Shift Operating Experience for Applicants	6/27/84
84-17	Annual Meeting to Discuss Recent Developments Regarding Operator Training, Qualifications and Examinations	7/3/84
84-18	Filing of Applications for Licenses and Amendments	7/6/84
84-19	Availability of Supplement 1 to NUREG-0933 "A Prioritization of Generic Safety Issues"	8/6/84
84-20	Scheduling Guidance for Licensee Submittals of Reloads that Involve Unreviewed Safety Questions	8/20/84
84-21	Long Term Low Power Operation in PWR's	10/16/84
84-22	Not used	
84-23	Reactor Vessel Water Level Instrumentation in BWRs	10/26/84
84-24	Clarification of Compliance to 10 CFR 50.49 Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants	12/27/84
85-01	Fire Protection Policy Steering Committee Report	1/9/85
85-03	Clarification of Equivalent Control Capacity For Standby Liquid Control Systems	1/28/85
85-04	Operator Licensing Examinations	1/29/85
85-05	Inadvertent Boron Dilution Events	1/31/85

NUREG 0797 CPSES SER AND SUPPLEMENTS

Enclosure 3 to TXX-92116
Pages SER, 15-4 through 15-6 and
SSER 23, 15-1

(5) startup of an inactive reactor coolant pump at an incorrect temperature

None of these transients are limiting; the most severe in terms of departure from nucleate boiling ratio and system pressure are the excessive load increase events. Only slight changes in primary system pressure were calculated, and the departure from nucleate boiling ratio did not fall below 1.4. The staff finds these results acceptable because they do not violate the appropriate limits.

15.2.2 Decreased Cooling Transients

The applicant has analyzed the following events which produced decreased primary system cooling:

(1) loss of external electrical load

(2) turbine trip

(3) inadvertent closure of main steam isolation valves

(4) loss of condenser vacuum and other events resulting in turbine trip

(5) loss of nonemergency ac power to the station auxiliaries

(6) loss of normal feedwater flow

(7) partial loss of forced reactor coolant flow

None of these transients are limiting; the most severe in terms of primary system overpressurization is the turbine trip transient, which results in a peak RCS pressure of approximately 2550 psia. Because this peak pressure is much lower than 110% of the RCS design pressure, the staff finds these results acceptable.

15.2.3 Increased Core Reactivity Transients

15.2.3.1 Boron Dilution Events

The principal means of causing an inadvertent boron dilution are the opening of the primary water makeup control valve and failure of the blend system, either by the controller or mechanical failure. The chemical volume and control system (CVCS) is designed to limit, even under various postulated failure modes, the dilution rate to values which, will allow sufficient time for automatic or operator response (depending on the mode of operation) to terminate the dilution before the shutdown margin is exhausted. This dilution rate is indicated by instrumentation. The applicant has analyzed the boron dilution event for all modes of operation.

Dilution During Refueling

Uncontrolled boron dilution cannot occur during the refueling mode because all sources of unborated water are isolated in this mode.

Dilution During Cold Shutdown

In this mode, the dilution rate is limited to a possible maximum of 10 ppm/min. This corresponds to the unborated primary water flow rate of 150 gpm. This restriction is administratively carried out and is to be implemented in the plant's Technical Specifications.

With the above dilution rate and a shutdown margin of 1% ΔK/K, a safety-grade microprocessor connected to the source range monitors will sound an alarm in the control room, close the normal CVCS discharge valves, and open the centrifugal charging pumps suction valves from the RWST, when the neutron count rate doubles over any 10 min-period. This microprocessor continuously monitors then neutron count rate and compares it to the neutron count rate 10 min earlier. When the count rate is doubled, the system initials the valve position changes and sounds an alarm. These actions by the microprocessor will prevent return to criticality.

Dilution During Hot-Shutdown and Hot-Standby

The differences in assumptions between this mode of operation and the preceding case are

(1) The shutdown margin is 1.6% ΔK/K for this case while it is 1% ΔK/K for the preceding case.

(2) The active RCS volume is 5000 ft^3 for this case while it is 3500 ft^3 for the preceding case.

Therefore, the dilution rate is limited for this case to 20 ppm/min. This corresponds to 425 gpm of unborated water. The microprocessor will provide the same protection for this case as for the preceding case.

Dilution During Startup and Power Operation

In these modes of operation, the reactor is protected by various trips and rod insertion limit alarms such that enough time is afforded the operator to terminate the dilution event and return the reactor under control.

15.2.3.2. Uncontrolled Rod Cluster Control Assembly (Rod) Bank Withdrawal From Zero Power Conditions

The applicant's discussion of the consequences of an uncontrolled rod cluster control assembly bank withdrawal at zero power have been analyzed. Such a transient can be caused by a failure of the reactor control or rod control systems. The analysis assumes a conservatively small (in absolute magnitude) negative Doppler coefficient and a positive moderator coefficient. Further, hot zero power initial conditions with the reactor just critical are chosen because they are known to maximize the calculated consequences. The reactivity insertion rate is assumed to be equivalent to the simultaneous withdrawal of the two highest worth banks at maximum speed (45 in./min).

Reactor trip is assumed to occur on the low setting of the power range neutron flux channel at 35% of full power (a 10% uncertainty has been added to the setpoint value). The maximum heat flux is less than the full power value, and the average fuel temperature increases to a value lower than the nominal full power value.

Evaluation Findings

The staff has reviewed the reactivity worths and reactivity coefficients used in the analysis and concludes that conservative values have been used. The staff has reviewed the calculated consequences of this design transient and concludes that they are acceptable.

Therefore, the staff finds that the requirements of GDC 20, which requires that protection be automatically initiated, and GDC 25, which requires that a single failure of the protection system does not result in violation of specified fuel design limits, have been satisfied.

15.2.3.3 Rod Cluster Control Assembly Malfunct' ns

The applicant has analyzed rod cluster control assembly misalignment incidents including a dropped full-length assembly, a dropped full-length bank, a misaligned full-length assembly and the withdrawal of a single assembly while operating at power. Misaligned rods are detectable by: (1) asymmetric power distributions sensed by excore nuclear instrumentation or core exit thermocouples, (2) rod deviation alarms, and (3) rod position indicators. A deviation of a rod from its bank by about 15 in. or twice the resolution of the rod position indicator will not cause power distributions to exceed design limits. When one or rod position channels are out of service, additional surveillance will be required to ensure rod alignment.

In the event of a dropped assembly, the automatic controller may return the reactor to full power. Analysis indicates that departure from nucleate boiling will not occur during this event. For the case of dropped groups, the reactor is tripped by the power-range negative-neutron-flux-rate trip, and the reactor is shut down without violating fuel damage limits.

For cases where a bank is inserted to its insertion limit with a single rod in the bank stuck in the fully withdrawn position, analysis indicates that departure from nucleate boiling will not occur. The staff has reviewed the calculated estimates of the expected reactivity and power distribution changes that accompany postulated misalignments of representative assemblies. The staff has concluded that the values used in this analysis conservatively bound the expected values, including calculational uncertainties.

The inadvertent withdrawal of a single assembly requires multiple failures in the control system, multiple operator errors, or deliberate operator actions, combined with a single failure of the control system. As a result, the single assembly withdrawal is classified as an infrequent occurrence. The resulting transient is similar to that which results from a bank withdrawal, but the increased peaking factor may cause departure from nucleate boiling to occur in the region surrounding the withdrawn assembly. Fewer than 5% of the rods in the core experience departure from nucleate boiling for such a transient.

15 ACCIDENT ANALYSIS

15.2 Moderate Frequency Transients

15.2.1 Increased Cooling Transients

The applicant has analyzed the following events which produced increased primary system cooling: (1) decrease in feedwater temperature, (2) increase in feedwater flow, (3) excessive load increase, and (4) opening of system generator relief or safety valve. None of these transients are limiting. Since the Technical Specifications for the Comanche Peak Steam Electric Station (CPSES) require all four reactor coolant loops to be operational in Modes 1 and 2, the applicant stated that there is no need to analyze the event of startup of an inactive reactor coolant pump at an incorrect temperature. Therefore, the staff concludes that the analytical results are acceptable because the specified acceptable fuel design limits are not violated.

15.2.3 Increased Core Reactivity Transients

15.2.3.1 Boron Dilution Events

The principal means of causing an inadvertent boron dilution are the opening of the primary water makeup control valve and failure of the blend system. The chemical volume and control system (CVCS) is designed to limit the dilution rate to values which will allow sufficient time for automatic or operator response to terminate the dilution before the shutdown margin is exhausted. Amendment 74 to the FSAR makes changes to reflect the plant as-built conditions such as a dilution flow rate of 167 gallons per minute (gpm) for all modes of operation and 4169 ft^3 minimum reactor coolant system volume for dilution during hot shutdown and hot standby.

The staff concludes that the applicant's analyses of inadvertent boron dilution for all modes of operation demonstrate that sufficient time exists for automatic or operator action to terminate the dilution before shutdown margin is exhausted.

15.3 Infrequent Transients and Postulated Accidents

15.3.7 Reactor Coolant Pump Locked Rotor Accident

The locked rotor accident was analyzed by postulating an instantaneous seizure of one reactor coolant system pump rotor. The reactor flow would decrease rapidly and the reactor would shut down as a result of a low-flow signal. The applicant reanalyzed this accident considering certain modified core values. The results showed a maximum cladding temperature of 1795°F and a peak coolant system pressure of 2648 psia, which indicates that the coolant system pressure boundary still maintains its integrity. The staff, therefore, concludes that the reanalysis of locked rotor accident is acceptable.

PWR BORON DILUTION EVENT ANALYSES CONDUCTED BY
INEL DATED APRIL 13, 1984

Enclosure 4 to TXX-92116