B 04

NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-305

FILE NUMBER

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

TO:	FROM:	DATE OF DOCUMENT
A. SCHWENCER	WISCONSIN PUBLIC SVC. CORP GREEN BAY, WISCONSIN E.W. JAMES	7/12/77
		DATE RECEIVED 7/14/77

	PROP	INPUT FORM	NUMBER OF COPIES RECEIVED
☑ LETTER ☑ NOTORIZED ☑ ORIGINAL ☑ UNCLASSIFIED ☐ COPY			3 SIGNED

DESCRIPTION

LTR. NOTARIZED 7/12/77....
RESPONSE TO 3/25/77 REQUEST CONCERNING A PROPOSED AMDT # 27 TO THE TECH. SPECS WHICH INCORPORATE THE ROD BOW PENALTIES. W/ATTACHMENTS

(2P & 17P)

DO NOT REMOVE

ACKNOWLEDGED

ENCLOSURE

PLANT NAME: KEWAUNEE
SAB

FOR ACTION/INFORMATION

SAFETY			ENVIRONMENTAL	
	ASSIGNED AD:		ASSIGNED AD:	V. MOORE (LTR)
✓	BRANCH CHIEF: (5)	SCHWENCER	BRANCH CHIEF:	
✓	PROJECT MANAGER:	NEIGHBORS	PROJECT MANAGER:	
✓	LICENSING ASSISTANT:	SHEPPARD	LICENSING ASSISTANT:	
			B. HARLESS	

INTERNAL DISTRIBUTION

✓	REG FILES		SYSTEMS SAFETY		PLANT SYSTEMS		SITE SAFETY &
✓	NRC PDR		HEINEMAN		TEDESCO		ENVIRON ANALYSIS
✓	I & E (4)		SCHROEDER		BENAROYA		DENTON & MULLER
✓	OELD				LAINAS		CRUTCHFIELD
✓	GOSSICK & STAFF		ENGINEERING		IPPOLITO		
	HANAUER		KNIGHT		F. ROSA		ENVIRO TECH.
	MIPC		BOSNAK				ERNST
	CASE		SIHWELL		OPERATING REACTORS		BALLARD
	BOYD		PAWLICKI	✓	STELLO		YOUNGBLOOD
				✓	EISENHUT		
	PROJECT MANAGEMENT		REACTOR SAFETY	✓	SHAO		SITE TECH.
	SKOVHOLT		ROSS	✓	BAER		
	P. COLLINS		NOVAK	✓	BUTLER		GAMMILL (2)
	HOUSTON		ROSZTOCZY	✓	GRIMES		
	MELTZ	✓	CHECK				SITE ANALYSIS
	HELTEMES						VOLLMER
	SK		AT&I				BUNCH
			SALTZMAN			✓	J. COLLINS
			RUTBERG				KREGER

EXTERNAL DISTRIBUTION

			CONTROL NUMBER
✓	LPDR: Kewaunee W/S.		Ap 2
✓	TIC	NSIC	
	NAT LAB		771950444
	REG IV (J. HANCHETT)		
✓	16 CYS ACRS SENT CATEGORY B		

NRC FORM 195 (2-76)

WISCONSIN PUBLIC SERVICE CORPORATION

Public Service

P.O. Box 1200, Green Bay, Wisconsin 54305

July 12, 1977

RECEIVED JUL 14 1977 U.S. NUCLEAR REGULATORY COMMISSION

Division of Operating Reactors
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

ATTN: Mr. A. Schwencer, Chief
Operating Reactors Branch #1

Gentlemen:

REF: Docket 50-305
Operating License DPR-43
Rod Bow

Regulatory File Cy.

On August 18, 1976, we transmitted a letter in regard to fuel rod bow to the Division of Operating Reactors. In that letter we addressed an assumed 6% penalty in $F^N_{\Delta H}$ which at that time was considered appropriate by NCR for fuel with 15,000 to 24,000 MWD/MTU burnup. At the time of that letter, the maximum burnup fuel at the Kewaunee Plant was within that range. As stated in our letter, we agreed to operate with a reduced $F^N_{\Delta H}$ until such time as that reduction was considered unnecessary.

On March 25, 1977, your office informed us by letter that the revised Rod Bow Penalties were considered appropriate and requested that we submit Technical Specifications incorporating the revised penalties and whatever credits that are justifiable. We stated in our previous letter the manner by which we were providing for the Rod Bow Penalty with a present maximum value of 6%, and have been operating in accordance with the direction of the March 25, 1977, letter since August 18, 1976. In response to your request, please find attached forty copies (40) of Proposed Amendment No. 27 to the Kewaunee Nuclear Power Plant Technical Specifications which incorporate the Rod Bow Penalties.

As addressed in our August 18, 1976, letter and in the proposed revised basis of the Technical Specification, we are proposing to take credit for excess Reactor Coolant System flow and a lower maximum T_{inlet}. The resultant Rod Bow Penalty would be 2% on high burnup fuel, >24,000 MWD/MTU. The design flow rate for the Reactor Coolant Pumps is 89,000 gpm each. The Startup Test Report submitted to the AEC in late 1974 addresses measurements of Reactor Coolant System flow. As stated in the Proposed Technical Specification Bases, we are taking credit for 50% of the excess flow. All protective signals related to flow are based upon full flow, not the design values; therefore, existing safety limits need not be adjusted. We also desire to take credit for operation with a lower T_{inlet} than originally designed. To implement this credit, we have proposed to reduce the maximum allowed T_{inlet} value of Specification 3.10K to 536.5F.

771950444

We also request that the Technical Specifications which related specifically to Core 2 be deleted. Those Specifications include 2.10.m and the associated bases. During licensing of Core 2, the K_1 value of the overtemperature safety limit was reduced from 1.11 to 1.08 as addressed in page 4 of the NRC Safety Evaluation of Amendment No. 10 dated April 5, 1976. The reduced K_1 value was an additional conservatism added for Core 2B because at the time that core was not fully evaluated by Westinghouse. Since Core 3, which was fully designed and evaluated by Westinghouse, now is in operation, the value of K_1 on page TS 2.3-1 should be changed back to 1.11.

In addition to the rod bow and Cycle II related Technical Specifications, we are also submitting a request for minor changes to Tables TS 3.14-1, which was found in error, and TS 4.1-1 deleting a Cycle I requirement. Table TS 3.14-1 was found to have several typing and location errors along with one erroneously listed shock suppressor. Table TS 4.1-1 required additional Nuclear Power Range calibrations during the early stages of Cycle I, which is no longer applicable.

Very truly yours,

E. W. James
Senior Vice President
Power Supply & Engineering

EWJ:sna
Attach.

Subscribed and Sworn to
Before Me This 12 Day
of July, 1977

Notary Public, State of Wisconsin

My Commission Expires
September 28, 1980

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection.

Objective

To prevent the principal process variables from exceeding a safety limit.

Specification

a. Reactor trip settings shall be as follows:

1. Nuclear Flux

A.	Source Range (high set point) - within span of source range instrumentation	
B.	Intermediate range (high set point)	$\leq$ 25% of rated power
C.	Power range (low set point)	$\leq$ 25% of rated power
D.	Power range (high set point)	$\leq$109% of rated power
E.	Power range fast flux rate trip (positive)	15% Δq/ 5 sec
F.	Power range fast flux rate trip (negative)	10% Δq/ 5 sec

2. Pressurizer

A.	High pressurizer pressure	$\leq$ 2385 psig
B.	Low pressurizer pressure	$\geq$ 1875 psig
C.	High pressurizer water level	$\leq$ 90% of full scale

3. Reactor Coolant Temperature

A. Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2(T-T')\left[\frac{1+\tau_1 s}{1+\tau_2 s}\right] + K_3(P-P') - f(\Delta I)\right]$

where

ΔT_o = Indicated ΔT at rated power, ^oF

T = Average temperature, ^oF

T' = 567.3 ^oF

P = Pressurizer pressure, psig

P' = 2235 psig

K_1 = 1.11 27

K_2 = 0.0090

K_3 = 0.000566

Proposed Amendment No. 27
7/8/77

3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the limits on core fission power distributions and to the limits on control rod operations.

Objective

To ensure 1) core subcriticality after reactor trip, 2) acceptable core power distribution during power operation in order to maintain fuel integrity in normal operation transients associated with faults of moderate frequency, supplemented by automatic protection and by administrative procedures, and to maintain the design basis initial conditions for limiting faults, and 3) limited potential reactivity insertions caused by hypothetical control rod ejection.

Specification

a. Shutdown Reactivity

When the reactor is subcritical prior to reactor startup, the hot shutdown margin shall be at least that shown in Figure TS 3.10-1. Shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon, boron, or part length rod position.

b. Power Distribution Limits

1. At all times, except during low power physics tests, the hot channel factors defined in the basis must meet the following limits:

$F_Q(Z) \leq (2.25/P) \times K(Z)$ for $P > .5$ 16

$F_Q(Z) \leq (4.50) \times K(Z)$ for $P \leq .5$

$F^N_H \leq 1.55 \quad 1 + 0.2 \quad (1-P)$ For 0 to 24,000 MWD/MTU burnup fuel 27

$F^N_H \leq 1.52 \quad 1 + 0.2 \quad (1-P)$ For greater than 24,000 MWD/MTU burnup fuel

3.10.k During steady state 100% power operation T inlet shall be maintained below 536.5°F. 27

3.10.1 During steady state 100% power operation reactor coolant system pressure shall be maintained above 2200 psig. 10

27

direct control over F^N_H and (c) an error in the predictions for radial power
shape, which may be detected during startup physics tests can be compensated for
in F_Q by tighter axial control, but compensation for F^N_H is less readily available. 9
When a measurement of F^N_H is taken, experimental error must be allowed for and 4%
is the appropriate allowance.

The F^N_H limits of specification 3.10.b.1 include consideration of fuel rod bow
effects. Since the effects of rod bow are dependent on fuel burnup. an additional
penalty is incorporated in a decrease in the F^N_H limit of 2% for 0 - 15000 MWD/MTU
fuel burnup, 4% for 1500 - 24000 MWD/MTU fuel burnup, and 6% for greater than
24000 MWD/MTU fuel burnup. These penalties are counter-balanced by credits for
increased Reactor Coolant flow and lower Core inlet temperature. The Reactor
Coolant System flow has been determined to exceed design by greater than 8%.
Since the flow channel protective trips are set on a percentage of full flow, 27
significant margin to DNB is provided. One half of the additional flow is taken
as a DNB credit to offset 2% of the F^N_H penalty. The assumed T_{inlet} for DNB
analysis was 540^{o}F while the normal T_{inlet} at 100% power is approximately 532^{o}F.
The reduction of maximum allowed T_{inlet} at 100% power to 536.5^{o}F as addressed in
specification 3.10k provides an additional 2% credit to offset the rod bow penalty.
The combination of the penalties and offsets results in a required 2% reduction of
allowed F^N_H for high burnup fuel, 24000 MWD/MTU.

Measurements of the hot channel factors are required as part of startup physics
tests, at least each full power month of operation, and whenever abnormal power
distribution conditions require a reduction of core power to a level based on 9
measured hot channel factors. The incore map taken following initial loading
provides confirmation of the basic nuclear design bases including proper fuel
loading patterns. The periodic monthly incore mapping provides additional

assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal operation, it is not necessary to measure these quantities. Instead 9
it has been determined that, provided certain conditions are observed, the hot channel factor limits will be met; these conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position.
2. Control rod banks are sequenced with overlapping banks as shown in Figure TS 3.10-4.
3. The control bank insertion limits are not violated.
4. Axial power distribution control specifications which are given in terms of flux difference control and control bank insertion limits are observed. 9
   Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation in F^N_H allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above conditions 1 through 4 are observed. these hot channel factors limits are met.

In specification 3.10.b.1 F_Q is arbitrarily limited for P _ 0.5 (except for low power physics tests).

The specifications for axial power distribution control referred to above are designed to minimize the effects of xenon redistribution on the axial power distribution during load-follow maneuvers.

Conformance with specification 3.10.b.6 through 3.10.b.9 ensures the F_Q upper bound envelope of 2.25 times Figure TS 3.10-2 is not exceeded and xenon distributions are not developed which at a later time would cause greater local power peaking, even though the current flux difference is within the limits specified. 16

The target (or reference) value of flux difference is determined as follows. At any time that equilibrium xenon conditions have been established, the indicated flux difference is noted with part length rods withdrawn from the core and with the full length rod control rod bank more than 190 steps withdrawn (i.e., normal full power operating position appropriate for the time in life, usually withdrawn farther as burnup proceeds). This value, divided by the fraction of full power at which the core was operating is the full power value of the target flux difference. Values for all other core power levels are obtained by multiplying the full power value by the fractional power. Since the indicated equilibrium value was noted, no allowances for excore detector error are necessary and indicated deviation of $\pm$ 5% I are permitted from the indicated reference value. During periods where extensive load following is required, it may be impractical to establish the required core conditions for measuring the target flux difference every month. For this reason, the specification provides two methods for updating the target flux difference. Figure TS 3.10-6 shows a typical construction of the target flux difference band at BOL and Figure TS 3.10-5 shows the typical variation of the full power value with burnup.

Strict control of the flux difference (and rod position) is not as necessary during part power operation. This is because xenon distribution control at part power is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at part power. Strict control of the flux difference is not possible during certain physics tests or during required, periodic, excore calibrations which require larger flux differences than permitted. Therefore, the specifications on power distribution control are not applied during physics tests or excore calibrations; this is acceptable due to the low probability of a significant accident occurring during these operations. 9

In some instances of rapid plant power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10% increment in peaking factor for flux difference in the range +14 to -14% (+11% to -11% indicated) increasing by ± 1% for each 2% decrease in rated power. Therefore, while the deviation exists the power level is limited to 90% or lower depending on the indicated flux difference without additional core monitoring. If, for any reason, flux difference is not controlled within the ± 5% band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50% is required to protect against potentially more severe consequences of some accidents unless incore monitoring is initiated. Only when the target band is violated do the limits under specification 3.10.b.8.a apply. 16

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, without part length rods, by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Condition II events the core is protected from overpower and a minimum DNBR of 1.30 by an automatic protection system. Compliance with the sepcifications is assumed as a precondition for Condition II transients, however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

The radial power distribution within the core must satisfy the design values assumed for calculation of power capability. Radial power distributions are measured as part of the startup physics testing and are periodically measured at a monthly or greater frequency. These measurements are taken to assure that the radial power distribution with any quarter core radial power asymmetry conditions are consistent with the assumptions used in power capability analyses.

The quadrant tilt power deviation alarm is used to indicate a sudden or unexpected change from the radial power distribution mentioned above. The two percent tilt alarm setpoint represents a minimum practical value consistent with instrumentation errors and operating procedures. This symmetry level is sufficient to detect significant misalignment of control rods. Misalignment of control rods is considered to be the most likely cause of radial power asymmetry. The requirement for verifying rod position once each shift is imposed to preclude rod misalignment which would cause a tilt condition less than the 2% alarm level. This monitoring is required by Technical Specifications, Section 4.1.

The two hour time interval in this specification is considered ample to identify a dropped or misaligned rod. In the event that the tilt condition cannot be eliminated within the two hour time allowance, additional time would be needed to investigate the cause of the tilt condition. The measurements would include a full core physics map utilizing the movable detector system. For a tilt condition $\leq$ 1.09 an additional 22 hours time interval is authorized to accomplish these measurements. However, to assure that the peak core power is maintained below limiting values, a reduction of reactor power of two percent for each one percent of indicated tilt is required. Physics measurements have indicated that the core radial power peaking would not exceed a two-to-one relationship with the indicated tilt from the excore nuclear detector system for the worst rod misalignment. In the event a tilt condition of $\leq$ 1.09 cannot be eliminated after 24 hours, the reactor power level will be reduced to the range required for flux mapping and turbine synchronization. 9

If tilt ratio greater than 1.09 occurs which is not due to a misaligned rod, the reactor shall be brought to a low power condition for investigation by flux mapping. However, if the tilt condition can be identified as due to rod misalignment, operation can continue at a reduced power (2% for each one percent the tilt ratio exceeds 1.0) for the 12 hour period necessary to correct the rod misalignment.

Trip shutdown reactivity is provided consistent with plant safety analysis assumptions. To maintain the required trip reactivity, the rod insertion limits of Figure TS 3.10-4 must be observed. In addition, for hot shutdown conditions, 9 the shutdown margin of Figure TS 3.10-1 must be provided for protection against the steambreak accident.

Rod insertion limits are used to assure adequate trip reactivity, to assure meeting power distribution limits, and to limit the consequences of a hypothetical rod ejection accident. The available control rod reactivity or excess beyond needs, decreases with decreasing boron concentration, because the negative reactivity required to reduce the core power level from full power to zero power is largest when the boron concentration is low.

The intent of the test to measure control rod worth and shutdown margin (Specification 3.10.d.3) is to measure the worth of all rods less the worth 9 of the worst case of an assumed stuck rod; that is, the most reactive rod. The measurement would be anticipated as part of the initial startup program and infrequently over the life of the plant, to be associated primarily with determinations of special interest, such as end-of-life cooldown or startup of fuel cycles which deviate from normal equilibrium conditions in terms of fuel loading patterns and anticipated control bank worths. These measurements will augment the normal fuel cycle design calculations and place the knowledge of shutdown capability on a firm experimental as well as analytical basis.

Operation with abnormal rod configuration during low power and zero power testing is permitted because of the brief period of the test and because special precautions are taken during the test.

The rod position indicator channel is sufficiently accurate to detect a rod $\pm$7-1/2 inches away from its demand position. If the rod position indicator channel is not operable, the operator will be fully aware of the inoperability of the channel, and special surveillance of core power tilt indications, using established procedures and relying on excore nuclear detectors, and/or movable 9
incore detectors, will be used to verify power distribution symmetry.

One inoperable control rod is acceptable provided the potential consequences of accidents are not worse than the cases analyzed in the safety analysis report. A 30 day period is provided for the re-analysis of all accidents sensitive to the changed initial condition.

The required drop time to dashpot entry is consistent with safety analysis.

The DNR related accident analysis assumed as initial conditions that the T_{inlet} was $4^{O}F$ above nominal design or T_{avg} was $4^{O}F$ above nominal design. The Reactor 10
Coolant System pressure was assumed to be 30 psi below nominal design.

System Name	Snubber I.D. Number	Approximate Location & Elevation	Accessible or Inaccessible (A or I)	Difficult to Remove (X)	High Radiation Area at Shutdown (X)	
(MS)	SS-H146	10'-8" W of col. K 2'-5 1/2" S of col. 6 EL 634'-9 1/2"	A	X		
Residual Heat Removal (RHR)	RHR-H10H	20'-9" N of col. 6 21'-0" E of col. K EL. 601'-0"	I		X	
(RHR)	RHR-H12B	9'-7 1/2" N of col. 6 22'-2" E of col. 9S EL. 626'	I		X	27
(RHR)	R-RHR-H14	36'-1" N of col. W 18'-6 5/8" E of col. N EL. 607'-6"	I			27
(RHR)	R-RHR-H15	36'-0 3/8" N of col. W. 18'-6 5/8" E of col. N EL. 607'-6"	I			27
(RHR)	RHR-H16A	5'-2 1/2" N of col. 6 12'-8 1/2" E of col. K EL. 617'-9"	I	X	X	
(RHR)	R-RHR-H18	27'-2 7/16" N of col. E 23'-4 7/16" E of col. N EL. 611'-0"	I		X	27
(RHR)	RHR-H49	20'-0" N of col. 6 22'-1" E of col. 9S EL. 601'-6"	I		X	27
Safety Injection (SI)	SI-H35	2'-0" N of col. 6 15'-11 3/8" E of col. K EL. 606'-9"	I	X	X	
(SI)	SI-H6D	3'-0" S of col, 6 1'-6" W of col. NW EL 629'-11 3/4"	I		X	27

Proposed Amendment No. 27
7/12/77

System Name	Snubber I.D. Number	Approximate Location & Elevation	Accessible or Inaccessible (A or I)	Difficult to Remove (X)	High Radiation Area at Shutdown (X)
(SI)	2180	Inside Containment EL. 620'-5" NE Quadrant	I	X	
(SI)	2243	Inside Containment EL. 614'-0" NE Quadrant	I	X	
(SI)	2295	Inside Containment NE Quadrant EL. 620'-5"	I	X	
(SI)	2513	Inside Containment NE Quadrant EL. 613'	I		
(SI)	RSI-H2A	46'-11 1/2" N of col. W. 16'-15 3/8" E of col. N EL. 607'-0"	I		
(SI)	RSI-15A	1'-6" N of col. W 22'-2" W of col. N EL. 602'-2"	I		
(SI)	RSI-H38	31'-2 3/4" N of col. E 3'-5 1/2" E of col. N EL. 607'-5"	I		
(SI)	RSI-H78	34'-7 1/4" N of col. W 15'-9 1/2" E of col. N EL. 601'	I		
(SI)	RSI-H83	17'-5" N of col. W 0'-5 3/4" E of col. N EL. 601'-0"	I		

27

27

Proposed Amendment No. 27
7/12/77

Table TS 3.14-1
Safety Related Hydraulic Shock Suppressors

System Name	Snubber I.D. Number	Approximate Location & Elevation	Accessible or Inaccessible (A or I)	Difficult to Remove (X)	High Radiation Area at Shutdown* (X)
Reactor Coolant RTD Line (RC)	RTD-H2	7'-10" E from ℄ of stm. gen. 1A 6'-10" S from ℄ of stm. gen. 1A	I		X
(RC)	RTD-H6	15'-3 1/2" E from ℄ of stm. gen. 1A 11'-9" S from ℄ of stm. gen. 1A EL 615'-3 3/16"	I		X
(RC)	RTD-H11	6'-2" N from ℄ of stm. gen. 1B 6'-3" W from ℄ of stm. gen. 1B EL. 616'-10 1/4"	I		X
Internal Containment Spray (ICS)	ICS-H7	13'-8 5/16" E of col. N 47'-10" N of ℄ of cont. vessel EL. 626'-8"	I		
(ICS)	ICS-H8	13'-8 5/16" E of col. N 97'-10" N of ℄ of cont. vessel EL. 627'-0"	I		
(ICS)	ICS-H9	8'-7 1/8" E of col. N 52'-2" N of col. E. EL. 649'-6"	I	X	
(ICS)	ICS-H10	49'-6" R from ℄ of cont. vessel 8'-7 1/8" N of col. E EL. 626'-8"	I		
(ICS)	ICS-H11	49'-6" R from ℄ of cont. vessel 8'-7 1/8" N of col. 5 EL. 627'-0"	I		
(ICS)	ICS-H12	52'-1 7/8" from ℄ of cont. vessel 9'-0 5/8" N from ℄ of cont. vessel EL. 649'-6"	I		
Main Steam (MS)	MS-H15A	4'-8" N of col. 6 1'-0 5/16" W of col. J EL. 664'-6"	A		

27

27

Proposed Amendment No. 27
7/12/77

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND
TEST OF INSTRUMENT CHANNELS
(Page 1 of 3)

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S (1) EFPM (3)	D (1) EFPQ (3)	(M) (2)	1) Heat balance 2) Signal to ΔT; bistable action (permissive, rod stop, trips) 27 3) Upper and lower chambers for axial off-set using in-core detectors
2.	Nuclear Intermediate Range	*S (1)	N.A.	P (2)	1) Once/shift when in service 2) Log level; bistable action (permissive, rod stop, trips)
3.	Nuclear Source Range	*S (1)	N.A.	P (2)	1) Once/shift when in service 2) Bistable action (alarm, trips)
4.	Reactor Coolant Temperature	*S	R	M (1) M (2)	1) Overtemperature ΔT 2) Overpower ΔT
5.	Reactor Coolant Flow	S	R **	M	
6.	Pressurizer Water Level	S	R **	M	
7.	Pressurizer Pressure	S	R **	M	
8.	4-KV Voltage & Frequency	N.A.	R	M	Reactor protection circuits only

Table TS4.1-1 (1 of 3)

TABLE TS 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND
TEST OF INSTRUMENT CHANNELS
(Page 3 of 3)

	Channel Description	Check	Calibrate	Test	Remarks
19.	Radiation Monitoring System	*D	R	M	Includes all 24 channels
20.	Boric Acid Make-Up Flow Channel	N.A.	R	N.A.	
21.	Containment Sump Level	N.A.	N.A.	R	
22.	Accumulator Level and Pressure	S	R	N.A.	
23.	Steam Generator Pressure	S	R	M	
24.	Turbine First Stage Pressure	S	A **	M	
25.	Portable Radiation Survey Instruments	*M	A	Q	
26.	Protective System Logic Channel Testing	N.A.	N.A.	M	Includes auto load sequencer
27.	Environmental Monitors	*M	N.A.	N.A.	
28.	Turbine Overspeed Protection Trip Channel	N.A.	R	M	
29.	Seismic Monitoring System	R	R	N.A.	
30.	Fore Bay Water Level	N.A.	R**	R	

A - Annually
D - Daily
M - Monthly
P - Prior to each startup if not done previous week
Q - Quarterly

R - Each refueling shutdown
S - Each shift
B/W - Every two weeks
N.A. - Not applicable
W - Weekly
EFPM - Effective Full Power Month
EFPQ - Effective Full Power Quarter

Proposed Amendment No. 27
7/12/77

*See Specification 4.1.d
** Only if test indicates calibration required