TECHNICAL SPECIFICATION REVISION CONTROL

WPS Amend. No.	Date	Page Submitted	NRC Amend. No.	Date	Pages Removed	Pages Inserted
53	01/12/83	Table TS 4.1-2, TS 6-28	50 51	10/06/83 12/28/83	Table TS 4.1-2, TS 6-26 Table TS 4.1-2. TS 6-26	Table TS 4.1-2, TS 6-26 Table TS 4.1-2, TS 6-26
54	04/29/83	TSv, TSvi, TS 3.4-1, TS 3.4-2, TS 3.5-6, TS 3.5-7, TS 3.8-4, TS 3.10-6, Table TS 3.5-2 (2 and 3 of 3), Table TS 3.14-1 (11 of 11), Table TS 3.15-1, Table TS 3.15-2, TS 4.1-2a, TS 4.2-4, TS 4.4-5, Table TS 4.1-1 (3 of 4), Table TS 4.1-2, Table TS 4.10-1 (2 of 6), TS 6-1, TS 6-2, TS 6-7, TS 6-24, TS 6-26, Figure TS 6.2-1	51	12/28/83	TSv, TSvi, TS 3.4-1, TS 3.4-2, TS 3.5-6, TS 3.5-7, TS 3.8-4, TS 3.10-6, Table TS 3.5-2 (2 and 3 of 3), Table TS 3.14-1 (11 of 11), Table TS 3.15-1, Table TS 3.15-2, TS 4.2-4, TS 4.4-5, Table TS 4.1-1 (3 of 4), Table TS 4.1-2, Table TS 4.10-1 (2 of 6), TS 6-1, TS 6-2, TS 6-7, TS 6-24, TS 6-26, Figure TS 6.2-1	TSv, TSvi, TS 3.4-1, TS 3.4-2. TS 3.5-6, TS 3.5-7, TS 3.8-4, TS 3.10-6, Table TS 3.5-2 (2 and 3 of 3), Table TS 3.14-1 (11 of 11), Table TS 3.15-1, Table TS 3.15-2, TS 4.1-2a, TS 4.2-4, TS 4.4-5, Table TS 4.1-1 (3 of 4), Table TS 4.1-2, Table TS 4.10-1 (2 of 6), TS 6-1, TS 6-2, TS 6-7, TS 6-24, TS 6-26, Figure TS 6.2-1
55	08/24/83	TSi, TSii, TSv, TS3.1-1, TS3.1.1a, TS3.1-2, TS3.1-2b, TS3.1-2c, TS3.2-2, TS3.3-1, TS3.3-2, TS3.3-3, TS3.3-4, TS3.3-5, TS3.3-6, TS3.3-7, TS3.3-8, TS3.3-9, TS3.3-10, TS3.3-11, TS3.3-12, TS3.4-1, TS3.4-2, TS3.4-3, TS3.7-2, TS3.8-1, TS3.8-2, TS3.8-2a, Table TS3.5-3 (page 1 and 2 of 2), TS4.4-5, TS4.4-6, TS4.4-7, TS4.4-11, TS4.4-12, TS4.12-1, TS4.12-2, TS4.12-3, Table TS4.1-2 (page 1 and 2 of 2), TS6-4, TS6-5, TS6-23				

TECHNICAL SPECIFICATION REVISION CONTROL

WPS Amend. No.	Date	Page Submitted	NRC Amend. No.	Date	Pages Removed	Pages Inserted
54a	11/21/83	Figs. TS 6.2-1 & TS 6.2-2	51	12/28/83	Figs. TS 6.2-1 & TS 6.2-2	Figs. TS 6.2-1 & TS 6.2-2
56	12/14/83	Figs. TS 3.10-6				

physical form for sample analysis or instrument calibration or associated with radioactive apperatus or components:

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility".

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR, Chapter 1: Part 20, Section 30.34 of Part 30 Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensees are authorized to operate the facility at steady state reactor core power levels not in excess of 1650 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 51, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications. 51

(3) The licensee may proceed with and is required to complete the modifications identified in Paragraphs 3.1.1, 3.1.2 and 3.1.4 through 3.1.28 of the Fire Protection Safety Evaluation Report. These modifications shall be completed by the dates specified in Table 3.1. Dates for resolution of items are specified in Table 3.2. In the event that these dates for completion cannot be met, the licensee shall submit a report explaining the circumstances and propose a revised schedule.

(4) Physical Protection

The licensee shall fully implement and maintain in effect all provisions of the following Commission approved documents, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). These approved documents consist of information withheld from public disclosures pursuant to 10 CFR 2.790 (d).

a) "Industrial Security Manual" dated May 25, 1977, January 9, 1978, December 18, 1978, January 30, 1979, March 7, 1979 and March 27, 1979.

b) Kewaunee Nuclear Power Plant Safeguards Contingency Plan, as originally submitted by letter of March 27, 1979, and subsequently revised and re-submitted by letter of February 20, 1981, pursuant to 10 CFR 73.40. The Safeguards Contingency Plan shall be fully implemented, in accordance with 10 CFR 73.40(b) within 30 days of this approval by the Commission.

51

LIST OF TABLES

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the Steam and Power Conversion System.

Objective

To assure minimum conditions of steam-relieving capacity and auxiliary feedwater supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentrations of water activity that might be released by steam relief to the atmosphere.

Specification

a. The reactor shall not be heated above 350°F unless the following conditions are satisfied.

   1. Five main steam safety valves per operable steam generator are available. 51

   2. Three auxiliary feedwater pumps are operable.

   3. System piping and valves directly associated with the above components are operable.

   4. A minimum of 30,000 gallons of water is available in the condensate storage tanks and the Service Water System is capable of delivering an unlimited supply from Lake Michigan.

   5. The iodine-131 activity on the secondary side of the steam generators does not exceed 1.0 μCi/cc.

b. If, when the reactor is above 350°F, any of the conditions of Specification 3.4.a cannot be met within 48 hours, and except for the conditions of 3.4.c, the reactor shall be shutdown and cooled below 350°F using normal operating procedures.

Basis

A reactor shutdown from power requires removal of core decay heat. Decay heat removal requirements are normally satisfied by the steam bypass to the condenser and by continued feedwater flow to the steam generators. Normal feedwater flow to the steam generators is provided by operation of the turbine-cycle feedwater system.

The ten main steam safety valves (five per steam generator) have a total com-
bined rated capability of 7,765,000 lbs/hr. The maximum full-power steam flow
is 7,449,000 lbs/hr; therefore, the main steam safety valves will be able to
relieve the total maximum steam flow if necessary. Below 10% power, only one 51
steam generator is required to be operable. The requirement that five main
steam safety valves per operable steam generator are available will assure
sufficient steam relief capability during this mode of operation.

In the unlikely event of complete loss of electrical power to the plant, continued capability of decay heat removal would be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary feedwater pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump is normally aligned with one steam generator: the discharge of the turbine-driven pump, which starts automatically, is manually valved as necessary to backup either or both motor-driven pumps, or to replace the standby function of either motor-driven pump when it is out of service. Any single auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the reactor.

The specified minimum water supply in the condensate storage tanks is sufficient for ninety minutes of hot shutdown plus a suitable margin to prevent loss of net positive suction head prior to switching suction to the service water system. Unlimited replenishment of the condensate storage supply is available from Lake Michigan through the Service Water System.

The secondary coolant activity is based on a postulated release of the contents of one steam generator to the atmosphere. This could happen, for example, as a result of a steam break accident combined with failure of a steam line isolation valve. The limiting dose for this case results from iodine-131

Each relay in the undervoltage protection channels will fail safe and is alarmed to alert the operator to the failure.

A blackout signal which occurs during the sequence loading following a safety injection signal will result in a reinitiation of the sequence loading logic at time step 0 as long as the Safety Injection signal has not been re-set. The Kewaunee Emergency Procedures warn the operators that a Blackout Signal occurring after reset of Safety Injection will not actuate the sequence loading and instructs to re-initiate Safety Injection if needed.

Turbine Overspeed Protection

Turbine overspeed protection is provided to limit the possibility of turbine missiles. Overspeed protection is provided by three independent systems based on diverse operating principles. The three systems are the electro-hydraulic (E-H) system, the mechanical trip system, and the Redundant Overspeed Trip System (ROST). The E-H and mechanical systems are single channel and operate on a one-out-of-one to trip logic; the ROST system is a three channel system, requiring two out of three channels to trip.(4) 51

Instrument Operating Conditions

During plant operations, the complete protective instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection Systems, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the Reactor Control and Protection System when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel on process control equipment is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. The source and intermediate range nuclear instrumentation system channels are not intentionally placed in a tripped mode since these are one-out-of-two trips, and the trips are therefore bypassed during testing. Testing does not trip the system unless a trip condition exists in another channel.

The operability of the instrumentation noted in Table 3.5-5 assures that sufficient information is available on these selected plant parameters to aid the operator in identification of an accident and assessment of plant conditions during and following an accident. In the event the instrumentation noted in Table 3.5-5 is not operable, the operator is given instruction on compensatory actions.

References:

(1) FSAR Section 7.5
(2) FSAR Section 14.3
(3) FSAR Section 14.2.5
(4) FSAR Section 10.2.2 51

therein through interlocks and administrative procedures. Placement of additional spent fuel racks will be controlled by detailed procedures to prevent traverse directly above spent fuel.

The one hundred hour decay time following plant shutdown is consistent with the assumption used in the dose calculation for the fuel handling accident.

The requirement for the spent fuel pool sweep system, including charcoal adsorbers, to be operating when spent fuel movement is being made provides 51
added assurance that the offsite doses will be within acceptable limits in the event of a fuel handling accident. The spent fuel pool sweep system is designed to sweep the atmosphere above the refueling pool and release to the Auxiliary Building vent during fuel handling operations. Normally, the charcoal adsorbers are bypassed but for purification operation, the bypass dampers are closed routing the air flow through the charcoal adsorbers. If the dampers do not close tightly, bypass leakage could exist to negate the usefulness of the charcoal adsorber. If the spent fuel pool sweep system is found not to be operating fuel handling within the Auxiliary Building will be terminated until the system can be restored to the operating condition.

The bypass dampers are integral to the filter housing. The test of the bypass leakage around the charcoal adsorbers will include the leakage through these dampers.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential radioiodine releases to the atmosphere. Bypass leakage for the charcoal adsorbers and particulate removal efficiency for HEPA filters are determined by halogenated hydrocarbon to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one high worth rod inserted and the part length rods fully withdrawn.

e. Rod Misalignment Limitations

This specification defines allowable limits for misaligned rod cluster control assemblies. In specifications 3.10.e.1 and 3.10.e.2, the magnitude, in steps, of an indicated rod misalignment may be determined by comparison of the respective bank demand step counter to the analog individual rod position indicator, the rod position as noted on the plant process computer, or through the conditioning module output voltage via a correlation of rod position vs. voltage. Rod misalignment limitations do not apply during physics testing. 51

1. When reactor power is greater than or equal to 85% of rating the rod cluster control assembly shall be maintained within ± 12 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 12 steps when reactor power is greater than or equal to 85%, the rod will be realigned or the core power peaking factors shall be determined within 4 hours, and specification 3.10.b applied. If peaking factors are not determined within 4 hours, the reactor power shall be reduced to less than 85% of rating.

2. When reactor power is less than 85% but greater than or equal to 50% of rating, the rod cluster control assemblies shall be maintained within ± 24 steps from their respective banks. If a rod cluster control assembly is misaligned from its bank by more than ± 24 steps when reactor power is less than 85% but greater than or equal to 50%, the rod will be realigned or the core power peaking factors shall be determined within 4 hours, and specification 3-10.b applied. If the peaking factors are not determined within 4 hours, the reactor power shall be reduced to less than 50% of rating. 51

3. And, in addition to 3.10.e.1 and 3.10.e.2 above, if the misaligned rod cluster control assembly is not realigned within 8 hours, the rod shall be declared inoperable.

TABLE TS 3.5-2

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP
(Page 2 of 3)

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
8	High Pressurizer Pressure	3	2	2	-		Maintain hot shutdown
9	Pressurizer High Water Level	3	2	2	-		Maintain hot shutdown
10	Low Flow In One Loop (< 50% full power)	3/loop	2/loop (any loop)	2	-		Maintain hot shutdown
	Low Flow Both Loops (10-50% full power)	3/loop	2/loop (any loop)	2	-		Maintain hot shutdown
11	Turbine Overspeed Protection***	3	1	2	-		Maintain < 50% of rated power
12	Lo-Lo Steam Generator Water Level	3/loop	2/loop	2/loop	-		Maintain hot shutdown
13	Undervoltage 4-KV Bus	2/bus	1/bus (both buses)	1/bus	-		Maintain hot shutdown
14	Underfrequency 4-KV Bus	2/bus	1/bus (both buses)	1/bus	-		Maintain hot shutdown

TABLE TS 3.5-2

INSTRUMENT OPERATION CONDITIONS FOR REACTOR TRIP
(Page 3 of 3)

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
15	Control Rod Misalignment Monitor						
	a. Rod position deviation	1	-	1	-		Log individual rod positions once/shift and after a load change >10% or after >30 in. of control rod motion
	b. Quadrant power tilt monitor (upper and lower ex-core neutron detectors)	1	-	1	-		Log individual rod positions once/shift and after a load change >10% or after >30 in. of control rod motion
16	Steam Flow/Feedwater Flow Mismatch	2	1	1	-		Maintain hot shutdown

NOTE 1: When block condition exists, maintain normal operation.

* When one channel is out of service, a bypass may be used to allow testing the other channels; a channel shall not be bypassed longer than 4 hours.

**** One additional channel may be taken out of service for zero power physics testing.**

*** There are three independent turbine overspeed trip systems. This specification requires two of these systems to be operable or go to column six. When only two systems are operable, an individual system may be blocked for no longer than four (4) hours to allow for testing.

System Name	Snubber I.D. Number	Approximate Location & Elevation	Accessible or Inaccessible (A or I)	Difficult to Remove (X)	High Radiatic Area at Shutdown (X)
(CVC)	RCVC-H-34	Inside Containment NW Quadrant El. 600'-0"	I		X
(CVC)	RCVC-H-35	Inside Containment NW Quadrant El. 626'-0"	I		X
(CVC)	RCVC-H-36	Inside Containment Pen. 13 N El. 612'-0"	I		
(CVC)	CVC-H-84	4'-9" N of col. (6) 5'-0" E of col. (J) El. 606'-7 7/8"	A		
(CVC)	CVC-H-96	1'-10" N of col. (6) 7'-0" E of col. (HW) El. 597'-0"	A		
Containment Spray (ICS)	CS-H-33A	11'-0" N of col. (4) 7'-0" W of col. (H) El. 607'-0"	A		
(ICS)	CS-H-39	3'-0" N of col. (6N) 11'-0" W of col. (J) El. 597'-6"	A		

51

TABLE TS 3.15-1

FIRE DETECTION INSTRUMENTATION

Fire Area		Detectors	Minimum # Required	Required Actions
AX-21	4160 Switchgear Room	3	2	Establish an hourly fire watch inspection
AX-23	Special Vent Filter Housings	9	9	If filter housing is in operation with charcoal filters in service establish an hourly fire watch inspection. If not in service establish a 4-hour inspection frequency.
AX-23	Auxiliary Building	4	2	Establish an hourly fire watch inspection
AX-24	Fuel Handling Area	3	2	Establish an hourly fire watch inspection 51
AX-30	Relay Room	19	6	Establish an hourly fire watch inspection
AX-32	Cable run area	11	8	Establish an hourly fire watch inspection
AX-35	Control Room	13	0	Control room is continuously manned
AX-37	CRD Room	7	4	Establish an hourly fire watch inspection
SB-65	Shield Building	6	2	Establish a four hour fire watch inspection
SC-70	Screenhouse	4	2	Establish an hourly fire watch inspection
TU-90/91	D/G 1A and day tank room	7	5	Establish an hourly fire watch inspection
TU-92/93	D/G 1B and day tank room	7	5	Establish an hourly fire watch inspection
TU 94	Cardox Room	1	1	Establish an hourly fire watch inspection
TU 95	Air Compressor & Pump Room	5	4	Establish an hourly fire watch inspection
TU 97	Battery Room 1A	1	1	Establish an hourly fire watch inspection
TU 98	Battery Room 1B	1	1	Establish an hourly fire watch inspection

TABLE TS 3.15-2

SAFETY RELATED FIRE HOSE STATIONS

Fire Hose Station No.	Location
0	Screenhouse, north stairway leading to lower level
1	Adjacent to Oil Storage Room "B" and SWPT Pressure Filter Assembly
4	Adjacent to D/G 1B and D/G 1B day tank rooms
5	Adjacent to D/G 1A and D/G 1A day tank rooms
7	Air Compressor and Pump Room near Auxiliary Feedwater Area Panel
9	Adjacent to S/G Blowdown Tank and 4160 V Switchgear Rooms
14	Adjacent to Battery Rooms 1A and 1B
21	Adjacent to Control Room, 626' elevation near stairs
23	Adjacent to Main Shop, Tank and Pump Room near Door 78
28	Aux. Building Basement North of Freight elevator (A)
29	Aux. Building Basement North of Laundry Pumps on south wall of valve gallery.
30	Aux. Building Basement solid radwaste handling area, west of MCC 1-45G
31	Aux. Building Mezz. Southwest of BA Transfer Pumps
32	Aux. Building Mezz. South of S/G Blowdown Tank
33	Stair well at 616 elevation next to "G" wall
34	Aux. Building Operating Floor East Side of RWST
35	Aux. Building Operating Floor West of entrance to BA Tank Room

Turbine Overspeed Protection

Surveillance on the turbine overspeed protection system varies depending on the system. The Electro-Hydraulic system is tested once per refueling cycle. The mechanical trip system has a calibration check performed once per refueling cycle and certain portions of it are tested on a monthly cycle. 51
The Redundant Overspeed Trip system is calibrated once per refueling cycle and tested monthly. In addition to these tests, the turbine governor and stop valves are tested monthly.

2. Steam Generator Tube Sample Selection and Inspection - The tubes selected for each in-service inspection shall:

   a. Include at least 3% of the total number of tubes in all steam generators. The tubes selected for these inspections shall be selected on a random basis except as noted in 4.2.b.2.b.

   b. Concentrate the inspection by selection of at least 50% of the tubes to be inspected from critical areas where experience in similar plants with similar water chemistry indicates higher potential for degradation.

   c. Include the inspection of all non-plugged tubes which previous inspections revealed in excess of 20% degradation. The previously degraded tubes need only be inspected about the area of previous degradation indication if their inspection is not employed to satisfy 4.2.b.2.a and 4.2.b.2.b above.

   d. The second and third sample inspections during each inservice inspection may be less than the full length of each tube by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

   e. If a tube does not permit the passage of the eddy current inspection probe the entire length and through the U-bend, this shall be recorded and an adjacent tube shall be inspected. The tube which did not allow passage of the eddy current probe shall be considered degraded.

The results of each sample inspection shall be classified into one of the following three categories.

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

51

to this value.

c. Residual Heat Removal System

1. Those portions of the Residual Heat Removal System external to the isolation valves at the containment shall be hydrostatically tested at 350 psig at each major refueling outage, or they shall be tested during their use in normal operation at least once between successive major refueling outages.

2. The total leakage from either train shall not exceed two gallons per hour. Visible leakage that cannot be stopped at test conditions shall be suitably measured to demonstrate compliance with this Specification. 51

3. Any repairs necessary to meet the specified leak rate shall be accomplished within seven days of resumption of power operation.

d. Shield Building Ventilation System

1. At least once per operating cycle or once every 18 months whichever occurs first, the following conditions shall be demonstrated:

   A. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 10 inches of water and the pressure drop across any HEPA filter bank is less than 4 inches of water at the system design flow rate (+10%).

   B. Automatic initiation of each train of the system.

   C. Operability of heaters at rating and the absence of defects by visual inspection.

2. A. The in-place DOP test for HEPA filters shall be performed (1) at least once per 18 months and (2) following painting, fire, or chemical release in any ventilation zone communicating with the system.

   B. The laboratory tests for activated carbon in the charcoal filters shall be performed (1) at least once per 18 months for filters in a standby status or after 720 hours of filter operation, and

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND
TEST OF INSTRUMENT CHANNELS

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				
a. Electro-Hydraulic System	N.A.	N.A.	R	
b. Mechanical System	N.A.	R (See Remarks)	M	A calibration check is performed for the Mechanical System once per refueling cycle; repairs are made if necessary.
c. Redundant Overspeed Trip System	N.A.	R	M	
29. Seismic Monitoring System	R	R	N.A.	
30. Fore Bay Water Level	N.A.	R**	R	

51

TABLE TS 4.1-2

MINIMUM FREQUENCIES FOR SAMPLING TESTS

	Sampling Tests	Test	Frequency	Maximum Time Between Tests (Days)	
1.	Reactor Coolant Samples	Gross Beta-Gamma activity (excluding tritium)	5/week	3	
		Tritium activity	Monthly	37	
		*Chemistry (Cl, F, O_2)	3/week	4	51
2.	Reactor Coolant Boron	*Boron concentration	2/week	5	
3.	Refueling Water Storage Tank Water Sample	Boron concentration	Monthly *****	37	
4.	Boric Acid Tanks	Boron concentration	Weekly	8	
5.	Accumulator	Boron concentration	Monthly	37	
6.	Spent Fuel Pool	Boron concentration	Monthly **	37	
7.	Secondary Coolant	pH	5/week	3	
		Ammonia	5/week	3	
		Sodium	5/week	3	
		Gross Beta-Gamma activity	Weekly	8	
		Iodine concentration	Weekly when gross Beta-Gamma activity $\geq 1.0\ \mu Ci/cc$	8	
8.	Waste Disposal System Liquid Effluent Monitor	Gross Beta-Gamma activity	During each batch release	N.A.	51
9.	Circulating Water Monitor	Radioactivity	Continuous ***	N.A.	51
10.	Auxiliary Building Vent Monitor	Gross Beta-Gamma activity	Continuous ****	N.A.	
11.	Containment Vessel Air Particulate Monitor	Fission gas particulate activity	Continuous ***	N.A.	51
12.	Containment Vessel Radiogas Monitor	Fission gas	Continuous ***	N.A.	51

Notes

* See Spec 4.1.D
** Sample will be taken monthly when fuel is in the pool.
*** Continuous monitoring takes place when reactor is in operation.
**** Operable during refueling also.
***** And after adjusting tank contents.

TABLE TS 4.10-1 (Page 2 of 6)

Operational Environmental Radiological Surveillance Program

Type of Sample	Location	Sampling Frequency	Type of Analysis	Frequency of Analysis	Reporting Units	Approximate Minimum Detectable Level	Comments
D. Well Water	K-1h K-1g	Monthly	Gross alpha (TR)[a]	Monthly	pCi/l	0.16 pCi/l	On all samples
			Gross beta (TR)	Monthly	pCi/l	0.28 pCi/l	On all samples
			K-40	Monthly	pCi/l	3.4 pCi/l	On all samples
			Tritium (actual level)	Quarterly	pCi/ml	0.005 pCi/ml	Quarterly composite on one well
			Sr-89	Quarterly	pCi/l	0.6 pCi/l	Quarterly composite on one well
			Sr-90	Quarterly	pCi/l	0.3 pCi/l	Quarterly composite on one well
			Radium-226	--	pCi/l	0.3 pCi/l	Only on samples when alpha is > 3 pCi/l
			Gamma Scan	--	pCi/l	---	Only on samples when beta is > 30 pCi/l
	K-10 K-11 K-12 K-13	Quarterly	Gross alpha (TR)	Quarterly	pCi/l	0.16 pCi/l	On all samples
			Gross beta (TR)		pCi/l	0.28 pCi/l	On all samples
			K-40		pCi/l	3.4 pCi/l	On all samples
			Sr-89; Sr-90		pCi/l	0.6 pCi/l 0.3 pCi/l	Only on samples when beta is > 10 pCi/l
			Radium-226		pCi/l	0.3 pCi/l	Only on samples when alpha is > 3 pCi/l
			Gamma Scan		pCi/l	---	Only on samples when beta is > 30 pCi/l
E. Precipitation	K-11	Monthly	Tritium	Monthly	pCi/ml	0.005 pCi/ml	On all samples 51
F. Milk	K-3 K-4 K-5 K-6 K-12 K-19	Weekly	I-131	Weekly*	pCi/l	0.5 pCi/l	On all samples
		Monthly	Gamma Scan	Monthly	pCi/l	---	Monthly composite
			Sr-89	Monthly	pCi/l	0.6 pCi/l	Monthly composite
			Sr-90	Monthly	pCi/l	0.3 pCi/l	Monthly composite
			Stable Potassium	Monthly	g/l	0.04 g/l	Monthly composite
			Stable Calcium	Monthly	g/l	0.01 g/l	Monthly composite

* - Weekly during grazing reason - monthly at all other times

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager has overall on-site responsibility for plant operation. In the absence of the Plant Manager, the succession to this responsibility shall be in the following order:

a. Maintenance Superintendent

b. Operations Superintendent

c. Assistant Superintendent Operations

d. Plant Services Superintendent

e. Shift Supervisor

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for plant management and technical support shall be as shown on Figure TS 6.2-1.

FACILITY STAFF

6.2.2 The plant organization shall be as shown on Figure TS 6.2-2 and:

a. Each on-duty shift complement shall consist of at least:

(1) One Shift Supervisor (SRO)
(2) Two licensed Reactor Operators
(3) One Auxiliary Operator
(4) One Equipment Operator
(5) One Radiation Technologist

b. In the event that one of the shift members becomes incapacitated due to illness or injury or the Radiation Technologist has to accompany an injured person to the hospital, reactor operations may continue with the reduced complement until a replacement arrives. In all but severe weather conditions, a replacement is required within two hours. 51

c. At least one licensed operator shall be in the control room when fuel is in the reactor.

d. At least two licensed operators shall be present in the control room during reactor startup, turbine generator synchronization to the grid, and during recovery from reactor trips.

e. Deleted 51

f. Refueling operations shall be directed by a licensed Senior Reactor Operator assigned to the refueling operation who has no other concurrent responsibilities during the refueling operation.

g. A five man fire response team, consisting of 3 Fire Brigade members and 2 Assistant Fire Brigade personnel, shall be maintained If a member of the fire response team becomes incapacitated due to illness or injury this requirement is deemed satisfied if a replacement arrives within two hours in all but the severest weather.

h. When the reactor is above the cold shutdown condition, a qualified Shift Technical Advisor shall be within 10 minutes of the control room.

ORGANIZATIONAL CHANGES

6.2.3 Changes not affecting safety may be made to the offsite and facility staff organizations. Such changes shall be reported to the Commission in the form of an application for license amendment within 60 days of the implementation of the change.

6.3 PLANT STAFF QUALIFICATIONS

6.3.1 Qualification of each member of the Plant Staff shall meet or exceed the minimum acceptable levels of ANSI-N18-.1-1971 for comparable positions, except for the Radiation Protection Supervisor 51 who shall meet or exceed the recommendation of Regulatory Guide 1-8, Revision 1-R, September 1975, or their equivalent as further clarified in Attachment 1 to the Safety Evaluation Report enclosed with Amendment No. 46 to Facility Operating License DPR-43.

6.3.2 The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in the design of the Kewaunee Plant and plant transient and accident analysis.

6. Review and/or prepare safety evalutions of all plant design changes.
7. Audits as required by the Quality Assurance Program and as outlined in Section 6.5.3.8.

AUTHORITY

6.5.2.4 Members of the Fuel and Fossil Operations, Power Plant Design and Construction, and System Planning and Engineering groups, although not directly responsible to the Vice President - Nuclear Power, are available for special projects and support of the Kewaunee Plant.

The Nuclear Design Change, Nuclear Services, and Nuclear Training Groups are responsible to the Manager - Nuclear Power. The Nuclear Licensing and Systems Group is responsible to the Vice President - Nuclear Power. 51

6.5.3 NUCLEAR SAFETY REVIEW AND AUDIT COMMITTEE (NSRAC)

FUNCTION

6.5.3.1 The NSRAC shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear Power Plant Operations
b. Nuclear Engineering
c. Chemistry and Radio-Chemistry
d. Metallurgy
e. Instrumentation
f. Radiological Safety
g. Mechanical and Electrical Engineering
h. Quality Assurance Practices
i. Other appropriate fields as determined by the Committee, to be associated with the unique characteristics of the nuclear power plant.

COMPOSITION

6.5.3.2 The NSRAC shall be composed of, but not necessarily limited to:

6.10.2 The following records shall be retained for the duration of the Plant Operating License.

a. Records of a complete set of as-built drawings for the plant as originally licensed and all print changes showing modifications made to the plant.

b. Records of new and spent fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of plant radiation and contamination surveys.

d. Records of radiation exposure of all plant personnel, and others who enter radiation control areas.

e. Records of radioactivity in liquid and gaseous wastes released to the environment.

f. Records of transient or operational cycles for these facility components.

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of meetings of the NSRAC and PORC.

j. Records for Environmental Qualification. 51

6.11 RADIATION PROTECTION PROGRAM

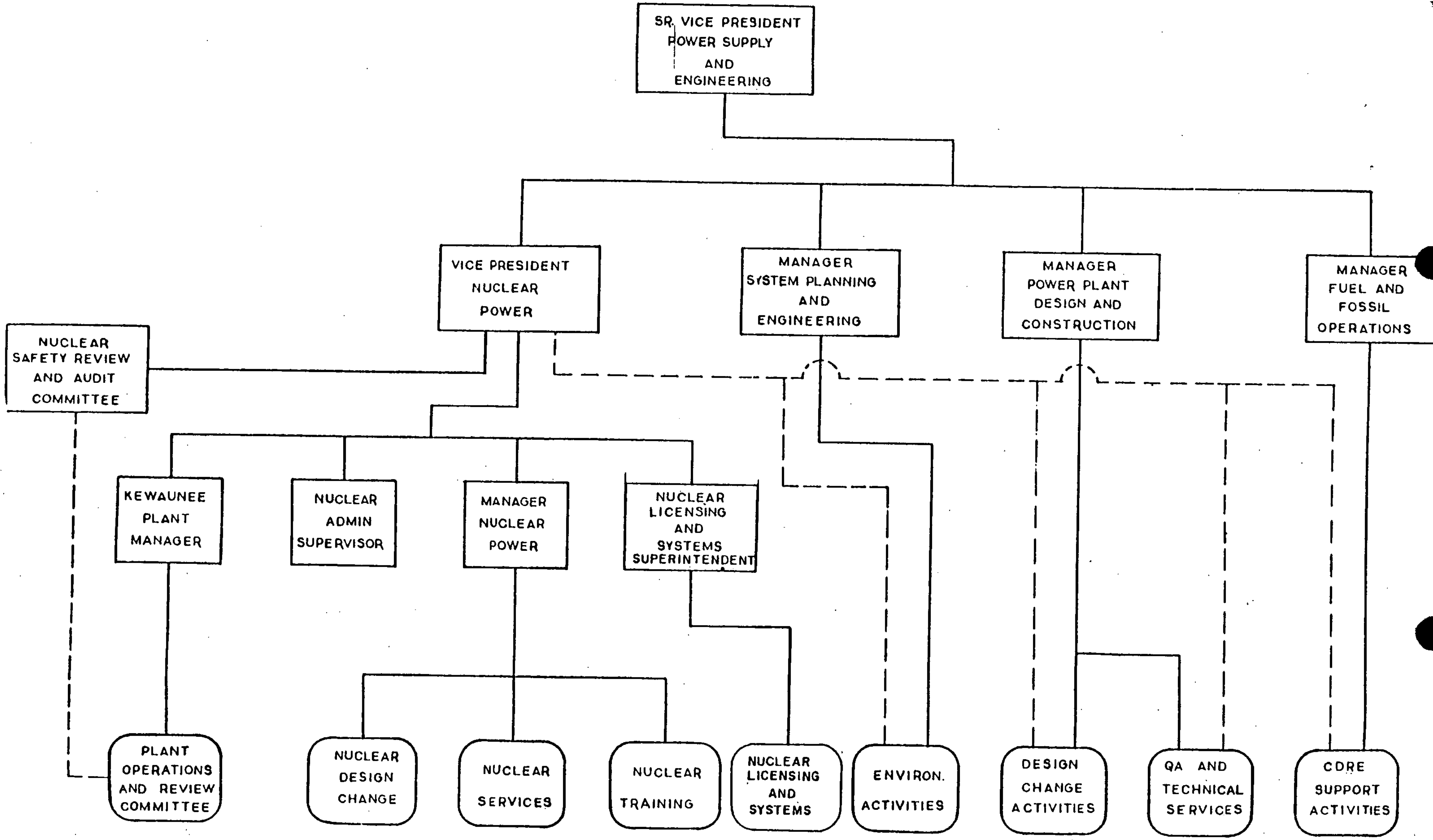
Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.14 DELETED

6.15 Secondary Water Chemistry

The licensee shall implement a secondary water chemistry monitoring program. The intent of this program will be to control corrosion thereby inhibiting steam generator tube degradation. The secondary water chemistry program shall act as a guide for the chemistry group in their routine as well as non-routine activities.

PAGE TS 6-28 DELETED



Legend

_____ Direct Responsibility

----- Lines of Communication

FIGURE TS 6.2-1

Amendment No. 51
12/28/83

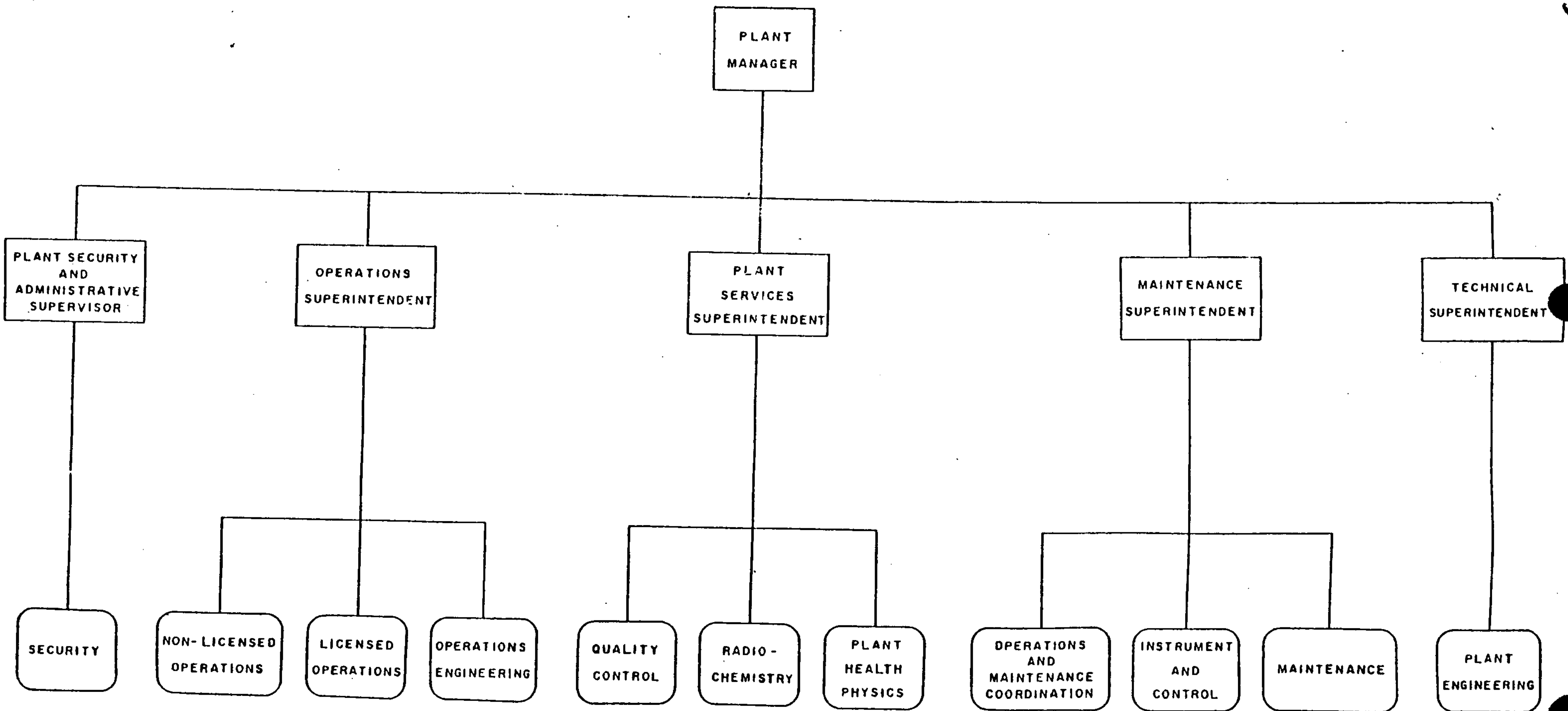


FIGURE TS 6.2-2

Amendment No. 51
12/28/83