LIMITING CONDITIONS FOR OPERATION

3.7. (cont'd.)

B. Standby Gas Treatment System

1. Except as specified in 3.7.B.3 below, both standby gas treatment systems shall be operable at all times when secondary containment integrity is required.

2.a. The results of the in-place cold DOP and halogenated hydrocarbon leak tests at $\leq$ design flow (1780 CFM) and at a reactor building pressure $\leq$-.25" Wg on HEPA filters and charcoal adsorber banks respectively shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show $\geq$99% radioactive methyl iodide removal with inlet conditions of: velocity $\geq$42 FPM, $\geq$1.75 mg/m^3 inlet methyl iodide concentration, $\geq$70% R.H. and $\leq 30^{\circ}$C.

c. Each fan shall be shown to provide 1780 CFM ±10%.

3. From and after the date that one standby gas treatment system is made or found to be inoperable for any reason, reactor operation or fuel handling is permissible only during the succeeding seven days unless such system is sooner made operable, provided that during such seven days all active components of the other standby gas treatment system, and its associated diesel generator, shall be operable.

SURVEILLANCE REQUIREMENTS

4.7 (cont'd.)

B. Standby Gas Treatment System

1. At least once per operating cycle the following conditions shall be demonstrated.

a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at the system design flow rate.

b. Inlet heater input is capable of reducing R.H. from 100 to 70% R.H.

2.a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once per year for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

d. Each system shall be operated with the heaters on at least 10 hours every month.

e. Test sealing of gaskets for housing doors downstream of the HEPA filters and charcoal adsorbers shall be performed at, and in conformance with, each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a.

3. System drains where present shall be inspected quarterly for adequate water level in loop-seals.

3.7.B & 3.7.C BASES (cont'd)

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and HEPA filters. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 99 percent for expected accident conditions. If the performance of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed.

Only one of the two standby gas treatment systems is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If neither system is operable, the plant is brought to a condition where the standby gas treatment system is not required.

4.7.B & 4.7.C BASES

Standby Gas Treatment System and Secondary Containment

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A 7.8 kw heater is capable of maintaining relative humidity below 70%. Heater capacity and pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with ANSI N510-1980. The test cannisters that are installed with the adsorber trays should be used for the charcoal adsorber efficiency test. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced

4.7.B & 4.7.C BASES

with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52, Revision 2, March, 1978. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1980. Any filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d. of Regulatory Guide 1.52, Revision 2, March, 1978.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of the heaters will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If system drains are present in the filter/adsorber banks, loop-seals must be used with adequate water level to prevent by-pass leakage from the banks.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one standby gas treatment system is inoperable, the other system must be tested daily. This substantiates the availability of the operable system and thus reactor operation or refueling operation can continue for a limited period of time.

3.7.D & 4.7.D BASES

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

The maximum closure times for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

These valves are highly reliable, have a low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation

3.12 Additional Safety Related Plant Capabilities

Applicability:

Applies to the operating status of the main control room ventilation system, the reactor building closed cooling water system and the service water system.

Objective:

To assure the availability of the main control room ventilation system, the reactor building closed cooling water system and the service water system upon the conditions for which the capability is an essential response to station abnormalities.

A. Main Control Room Ventilation

1. Except as specified in Specification 3.12.A.3 below, the control room air treatment system, the diesel generators required for operation of this system and the main control room air radiation monitor shall be operable at all times when containment integrity is required.

2.a. The results of the in-place cold DOP and halogenated hydrocarbon tests at $\leq$ design flow (341 CFM) and at control room pressure on HEPA filters and charcoal adsorber banks respectively shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show $\geq$ 99% radioactive methyl iodide removal with inlet conditions of: velocity $\geq$22 FPM, $\geq$1.75 mg/m^3 inlet iodide concentration, $\geq$ 95% R.H. and $\leq$30°C.

c. Each fan shall be shown to provide 341 CFM ±10%.

4.12 Additional Safety Related Plant Capabilities

Applicability:

Applies to the surveillance requirements for the main control room ventilation system, the reactor building closed cooling water system and the service water system which are required by the corresponding Limiting Conditions for Operation.

Objective:

To verify that operability or availability under conditions for which these capabilities are an essential response to station abnormalities.

A. Main Control Room Ventilation

1. At least once per operating cycle, the pressure drop across the combined HEPA filters and charcoal absorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate.

2.a. The tests and sample analysis of Specification 3.12.A.2 shall be performed at least once per year for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal absorber bank or after any structural maintenance on the system housing.

3.12 BASES

A. Main Control Room Ventilation System

The control room ventilation system is designed to filter the control room atmosphere for intake air and/or for recirculation during control room isolation conditions. The system is designed to automatically start upon control room isolation and to maintain the control room pressure to the design positive pressure so that all leakage should be out leakage.

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 intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and HEPA filters. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 99 percent for expected accident conditions. If the performance of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10 CFR Part 50.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation or refueling operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is shutdown and brought to cold shutdown within 24 hours, or refueling operations are terminated.

B. Reactor Building Closed Cooling Water System

The reactor building closed cooling water system has two pumps and one heat exchanger in each of two loops. Each loop is capable of supplying the cooling requirements of the essential services following design accident conditions with only one pump in either loop.

The system has additional flexibility provided by the capability of interconnection of the two loops and the backup water supply to the critical loop by the service water system. This flexibility and the need for only one pump in one loop to meet the design accident requirements justifies the 30 day repair time during normal operation and the reduced requirements during head-off operations requiring the availability of LPCI or the core spray systems.

C. Service Water System

The service water system consists of four vertical service water pumps located in the intake structure, and associated strainers, piping, valving and instrumentation. The pumps discharge to a common header from which independent piping supplies two Seismic Class I cooling water loops and one turbine building loop. Automatic valving is provided to shutoff all supply to the turbine building loop on drop in header pressure thus assuring supply to the Seismic Class I loops each of which feeds one diesel generator, two RHR service water booster pumps, one control room basement fan coil unit and one RBCCW

Revised Technical Specifications for
Low-Low Set Design Change

Revised Pages: 50
52b
83

Reference: 1) NPPD Letter, J. M. Pilant to D. B. Vassallo, "Safety Relief Valve (S/RV) Low-Low Set (LLS) System and Lower Main Steam Isolation Valve (MSIV) Water Level Trip", Dated December 17, 1982

On March 4, 1983, Amendment 83 was issued to the Cooper Nuclear Station Facility Operating License to incorporate into the Technical Specifications the Low-Low Set System. This system was implemented to mitigate excessive loads on Cooper's Mark I Containment brought about by subsequent safety relief valve actuations. The setpoint for Main Steam Isolation Valve Water Level Trip was lowered by the change to provide additional safety margin and to reduce safety relief valve challenges as recommended in NUREG-0737, Item II.K.3.16. A description and safety analysis of this change is contained in NEDE-22223, "Low-Low Set Logic and Lower MSIV Trip for BWR's with Mark I Containment", an enclosure to Reference 1.

The modification was properly implemented and tested with all related drawings and procedures updated. However, the initial Technical Specification change submitted in Reference 1 overlooked several points which should be corrected. These points in no way affect the operation, surveillance, or safety function of the modification. Rather, they are editorial changes to correct errors and to achieve consistency of nomenclature. Accordingly, Nebraska Public Power District proposes the following revisions be made to the Technical Specifications:

1. Delete from page 50 the entry for reactor low-low water level instrument as it no longer initiates primary containment isolation.

2. Change Group 7 isolation signal to reactor low-low-low water level (≥-145.5 in.) on page 52b and reflect this in the Bases for Section 3.2.A on page 83.

3. Delete reference to HPCI and RCIC initiation under the "Primary Containment Isolation Functions" bases section on page 83 because it is unrelated to primary system isolation.

Evaluation of this Revision with Respect to 10CFR50.92

A. The enclosed Technical Specification change is judged to involve no significant hazards based on the following:

   1. Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Evaluation:

Because the proposed change is editorial in nature and does not change existing equipment, surveillances, or procedures, it does not affect the probability or consequence of an accident previously evaluated.

2. Does the proposed license amendment create the possibility for a new or different kind of accident from any accident previously evaluated?

Evaluation:

Because the proposed change does not introduce any new mode of operation, the possibility of an accident of a different type than analyzed in the Final Safety Analysis Report would not result from the change; therefore, the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Evaluation:

Because the proposed change does not change existing facility equipment, surveillance, or procedure and is intended to correct errors and achieve consistency in nomenclature, it does not involve a significant reduction in a margin of safety.

B. Additional basis for proposed no significant hazards consideration determination:

The Commission has provided guidance concerning the application of the standards for making a no significant hazards consideration determination by providing certain examples (48FR14870). The examples include "(i) A purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature." It is the District's belief the proposed change is encompassed by the above example.

COOPER NUCLEAR STATION
TABLE 3.2.A (Page 1)
PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION INSTRUMENTATION

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability is Not Assured (2)
Main Steam Line High Rad.	RMP-RM-251, A,B,C,&D	≤ 3 Times Full Power	2	A or B
Reactor Low Water Level	NBI-LIS-101, A,B,C,&D	≥+12.5" Indicated Level	2(4)	A or B
Reactor Low Low Low Water Level	NBI-LIS-57 A & B #1 NBI-LIS-58 A & B #1	≥-145.5" Indicated Level	2	A or B
Main Steam Line Leak Detection	MS-TS-121, A,B,C,&D 122, 123, 124, 143, 144, 145, 146, 147, 148, 149, 150	≤ 200°F	2(6)	B
Main Steam Line High Flow	MS-dPIS-116 A,B,C,&D 117, 118, 119	≤ 140% of Rated Steam Flow	2(3)	B
Main Steam Line Low Pressure	MS-PS-134, A,B,C,&D	≥ 825 psig	2(5)	B
High Drywell Pressure	PC-PS-12, A,B,C,&D	≤ 2 psig	2(4)	A or B
High Reactor Pressure	RR-PS-128 A & B	≤ 75 psig	1	D
Main Condenser Low Vacuum	MS-PS-103, A,B,C,&D	≥ 7" Hg (7)	2	A or B
Reactor Water Cleanup System High Flow	RWCU-dPIS-170 A & B	≤ 200% of System Flow	1	C

NOTES FOR TABLE 3.2.A (cont'd.)

Isolations

1. Secondary Containment Isolation
2. Start Standby Gas Treatment System

Group 7

Isolation Signals:

1. Reactor Low Low Low Water Level ($\geq$-145.5 in)
2. Main Steam Line High Radiation ($\leq$3 times full power background)

Isolations:

1. Reactor Water Sample Valves

3.2 BASES

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the specifications are (1) to assure the effectiveness of the protective instrumentation when required even during periods when portions of such systems are out of service for maintenance, and (2) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

A. Primary Containment Isolation Functions

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation, set to trip at 176.5" (+12.5") above the top of the active fuel, closes all isolation valves except those in Groups 1, 4, 5, and 7. Details of valve grouping and required closing times are given in Specification 3.7. For valves which isolate at this level this trip setting is adequate to prevent core uncovery in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low low low reactor water level instrumentation is set to trip when the water level is 19" (-145.5") above the top of the active fuel. This trip closes Groups 1 and 7 Isolation Valves (Reference 1), activates the remainder of the CSCS subsystems, and starts the emergency diesel generators. These trip level settings were chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that post accident cooling can be accomplished,

Revised Technical Specifications for
Refueling Interlocks Clarification

Revised Page: 205

The current Technical Specifications for Cooper Nuclear Station require all refueling interlocks, except the one-rod-out interlock, to be operable during multiple control rod removal regardless of whether fuel is present in the core or not. Nebraska Public Power District requests a revision to the Technical Specifications to delete the above requirement when fuel is not present in the reactor vessel. The proposed change conforms to NUREG-0123, Revision 3, Standard Technical Specifications 3.9.10.2 which states the requirement is applicable when there is "fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed."

Evaluation of this Revision with Respect to 10CFR50.92

The enclosed Technical Specification change is judged to involve no significant hazards based on the following:

1. Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Evaluation:

   Because the proposed change conforms with the GE Standard Technical Specifications and clarifies the need for refueling interlock operability during the time no fuel is present in the reactor vessel it does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed license amendment create the possibility for a new or different kind of accident from any accident previously evaluated?

   Evaluation:

   Because the change does not affect the requirements for refueling interlock operability while handling fuel it does not create the possibility for a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Evaluation:

   Because this change clarifies the operability requirements of the refueling interlocks during the time fuel is not present in the core and is in agreement with the GE Standard Technical Specifications it does not involve a significant reduction in a margin of safety.

3.10.A (Cont'd)

6. Any number of control rods may be withdrawn or removed from the reactor core providing the following conditions are satisfied:

a. The reactor mode switch is locked in the "refuel" position. The refueling interlock which prevents more than one control rod from being withdrawn may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. When fuel is present in the reactor vessel, all other refueling interlocks shall be operable.

B. Core Monitoring

During core alterations two SRM's shall be operable, one in the core quadrant where fuel or control rods are being moved and one in an adjacent quadrant. For an SRM to be considered operable, the following conditions shall be satisfied:

1. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)

2. Operable SRM's shall have a minimum of 3 cps except as specified in 3 and 4 below.

3. Prior to spiral unloading, the SRM's shall have an initial count rate of 3 cps. During spiral unloading, the count rate on the SRM's may drop below 3 cps.

4.10 (Cont'd)

B. Core Monitoring

Prior to making any alterations to the core, the SRM's shall be functionally tested and checked for neutron response. Thereafter, while required to be operable, the SRM's will be checked daily for response (or every 12 hours until 3 cps is attained if the spiral reload technique is being used).

Revised Technical Specifications for
Environmental Qualification Program

Revised Pages: 226
226a (deleted)
227

The current Technical Specifications for Cooper Nuclear Station specify a deadline of June 30, 1982, for environmental qualification of all safety-related electrical equipment. On November 19, 1984, the U.S. Nuclear Regulatory Commission issued its final rule (49FR45571) eliminating the June 30, 1982, deadline for environmental qualification.

Nebraska Public Power District requests a revision to the Technical Specifications to delete the above deadline from the Administrative Controls Section. The deletion of the deadline will have no effect on establishing an Environmental Qualification Program at Cooper Nuclear Station since the District has already committed to implement the program in accordance with NRC guidelines.

Evaluation of this Revision with Respect to 10CFR50.92

The proposed amendment involves no significant hazards consideration since it will not 1) involve a significant increase in the possibility or consequences of an accident previously evaluated, 2) create the possibility of a new or different kind of accident from any accident previously evaluated, or 3) involve a significant reduction in a margin of safety. The Commission has provided guidance concerning the application of the standards for making a no significant hazards consideration determination by providing certain examples (48FR14870). The examples include "(i) A purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, a correction of an error, or a change in nomenclature." Another example given is "(vii) A change to make a license conform to changes in regulations, where the license change results in very minor changes to facility operations clearly in keeping with the regulations." It is the District's belief the proposed change is encompassed by the above examples.

6.3 (cont'd)

A. High Radiation Areas

In lieu of the "control device" or "alarm signal" required by Paragraph 20.203 (c) (2) of 10 CFR 20 each High Radiation Area (100 mrem/hr or greater) shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by requiring notification and permission of the shift supervisor. Any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

6.3.5 Temporary Changes

Temporary changes to procedures which do not change the intent of the original procedure may be made, provided such changes are approved by two members of the operating staff holding SRO licenses. Such changes shall be documented and subsequently reviewed by the Division Manager of Nuclear Operations within one month.

6.3.6 Exercise of Procedures

Drills of the Emergency Plan procedures shall be conducted annually, including a check of communications with offsite support groups. Drills on the procedures specified in 6.3.2.A, B, and C above shall be conducted as part of the retraining program.

6.3.7 Programs

The following programs shall be established:

A. Systems Integrity Monitoring Program

A program shall be established to reduce leakage to as low as practical levels from systems outside the primary containment during a serious accident that would or could contain highly radioactive fluids. This program shall include provisions establishing preventive maintenance and periodic visual inspection requirements, and leak testing requirements for each system at a frequency not to exceed refueling cycle intervals.

B. Iodine Monitoring Program

A program shall be established to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include training of personnel, procedures for monitoring and provisions for maintenance of sampling and analysis equipment.

C. Environmental Qualification Program

A. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.3 (cont'd)

B. Post-Accident Sampling System (PASS)

A program shall be established to ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. This program shall include training of personnel, procedures for sampling and analysis and provisions for operability of sampling and analysis equipment.

Revised Technical Specifications for
Standby Liquid Control System
RCIC Test and Calibration Frequencies Table
Typographical Errors

Revised Pages: 111
75

To correct typographical errors discovered on two pages of the CNS Technical Specifications, Nebraska Public Power District proposes the following revisions be made to same.

1. Change Subsection III.8.5 to read Subsection III.9.5 in paragraph 2 of 4.4 Bases, Standby Liquid Control System, page 111.

2. Change Item 1. Logic Buss Power Monitor, under item category Logic Systems (4)(6) on Table 4.2.B, RCIC Test & Calibration Frequencies (page 6), page 75, to read 1. Logic Bus Power Monitor.

Evaluation of this Revision with Respect to 10CFR50.92

The proposed amendment incorporates changes that are of an administrative nature to correct errors and involves no significant hazards considerations since it will not 1) involve a significant increase in the possibility or consequences of an accident previously evaluated, 2) create the possibility of a new or different kind of accident from any accident previously evaluated, or 3) involve a significant reduction in a margin of safety. The Commission has provided guidance concerning the application of the standards for making a no significant hazards consideration determination by providing certain examples (48FR14870). The examples include "(i) A purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature." It is the District's belief that the proposed change is encompassed by the above example.

3.4 BASES (cont'd.)

The volume versus concentration requirement of the solution is such that, should evaporation occur from any point within the curve, a low level alarm will annunciate before the temperature versus concentration requirements are exceeded.

The quantity of stored boron includes an additional margin (25 percent) beyond the amount needed to shutdown the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water.

A minimum quantity of 2650 gallons of solution having a 16 percent sodium pentaborate concentration, or the equivalent as shown in Figure 3.4.1, is required to meet this shutdown requirement. For the minimum required pumping rate of 38.2 gpm, the maximum net storage volume of the boron solution is established as 4780 gallons.

4.4 BASES

STANDBY LIQUID CONTROL SYSTEM

Experience with pump operability indicates that the monthly test, in combination with the tests during each operating cycle, is sufficient to maintain pump performance. The only practical time to fully test the liquid control system is during a refueling outage. Various components of the system are individually tested periodically, thus making unnecessary more frequent testing of the entire system.

The bases for the surveillance requirements are given in subsection III.9.6 of the Final Safety Analysis Report, and the details of the various tests are discussed in subsection III.9.5. The solution temperature and volume are checked at a frequency to assure a high reliability of operation of the system should it ever be required.

COOPER NUCLEAR STATION
TABLE 4.2.B (Page 6)
RCIC TEST & CALIBRATION FREQUENCIES

Item	Item I.D. No.	Functional Test Freq.	Calibration Freq.	Instrument Check
Instrument Channels				
1. Reactor High Water Level	NBI-LIS-101 A & C, #2	Cnce/Month (1)	Once/3 Months	Once/Day
2. Reactor Low Water Level	10A - K79 A & B 10A-K80 A & B	Once/Month (1)	Once/3 Months	Once/Day
3. RCIC High Turbine Exhaust Press.	RCIC-PS-72, A & B	Once/Month (1)	Once/3 Months	None
4. RCIC Low Pump Suction Press.	RCIC-PS-67-1	Once/Month (1)	Once/3 Months	None
5. RCIC Steam Line Space Excess Temp.	RCIC-TS-79, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-80, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-81, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
	RCIC-TS-82, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
6. RCIC Steam Line High ΔP	RCIC-dPIS-83	Once/Month (1)	Once/3 Months	None
	RCIC-dPIS-84	Once/Month (1)	Once/3 Months	None
7. RCIC Steam Supply Press. Low	RCIC-PS-87, A,B,C, & D	Once/Month (1)	Once/3 Months	None
8. RCIC Low Pump Disch. Flow	RCIC-FIS-57	Once/Month (1)	Once/3 Months	None
9. Pump Disch. Line Low Pressure	CM-PS-269	Once/3 Months	Once/3 Months	None
10. RCIC Turbine Conditional Supv. Alarm Timer	RCIC-TDR - K9	Once/Month (1)	Once/Oper. Cycle	None
11. RCIC Steam Line High ΔP Actuation Timer	RCIC-TDR-K-12	Once/Month	Once/Oper. Cycle	None
	RCIC-TDR-K-32	Once/Month	Once/Oper. Cycle	None
Logic Systems (4)(6)				
1. Logic Bus Power Monitor		Once/6 Months	N.A.	
2. RCIC Initiation		Once/6 Months	N.A.	
3. Turbine Trip		Once/6 Months	N.A.	
4. RCIC Automatic Isolation		Once/6 Months	N.A.	

Revised Technical Specifications for
Section 6 Administrative Controls
Editorial Changes

Revised Pages: iv (Table of Contents)
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225a (deleted)
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Section 6, Administrative Controls, of the current CNS Technical Specifications has undergone various changes during the past several years which have introduced discontinuities between the pages of some subsections (i.e.; gaps). To correct this, Nebraska Public Power District proposes a change in Technical Specifications (purely editorial in nature) which simply condenses related subsections which are spread out over several pages. The content of this material remains completely unchanged - there are no deletions, wording modifications, syntax or sequence changes, etc. Page 225a was deleted as a byproduct of compressing the contents of the various affected pages onto fewer pages. This proposed editorial change is being submitted to provide for improved readability and understanding of the Technical Specifications.

Evaluation of this Revision with Respect to 10CFR50.92

A. The enclosed Technical Specification change is judged to involve no significant hazards based upon the following:

1. Does the proposed license amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Evaluation:

No. The proposed amendment does not impact the probability or consequences of any accident previously evaluated.

2. Does the proposed license amendment create the possibility for a new or different kind of accident from any accident previously evaluated?

Evaluation:

No. The proposed amendment does not impact upon any new or old accident analyses since it is purely editorial in nature and does not change the content of the Technical Specifications whatsoever.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Evaluation:

   No. The proposed amendment is intended to clarify the Technical Specifications by introducing improved continuity and should improve safety, if anything.

B. Additional basis for the proposed no significant hazards consideration determination:

The Commission has provided guidance concerning the application of the standards for making a no significant hazards consideration determination by providing certain examples (48CFR14870). The examples include: "(i) A purely administrative change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature . . ." It is the District's belief the proposed change is encompassed by the above example.

f. Investigate all violations of Technical Specifications, including reporting evaluation and recommendations to prevent recurrence, to the Assistant General Manager - Nuclear and to the Chairman of the NPPD Safety Review and Audit Board.

g. Perform special reviews and investigations and render reports thereon as requested by the Chairman of the Safety Review and Audit Board.

h. Review all reportable events specified in Section 50.73 to 10CFR Part 50.

i. Review drills on emergency procedures (including plant evacuation) and adequacy of communication with off site groups.

j. Periodically review procedures required by Specifications 6.3.1, 6.3.2, 6.3.3, and 6.3.4 as set forth in administrative procedures.

5. Authority

a. The Station Operations Review Committee shall be advisory.

b. The Station Operations Review Committee shall recommend to the Division Manager of Nuclear Operations approval or disapproval of proposals under items 4, a through e and j above. In case of disagreement between the recommendations of the Station Operations Review Committee and the Division Manager of Nuclear Operations, the course determined by the Division Manager of Nuclear Operations to be the more conservative will be followed. A written summary of the disagreement will be sent to the Assistant General Manager - Nuclear and to the NPPD Safety Review and Audit Board.

c. The Station Operations Review Committee shall report to the Chairman of the NPPD Safety Review and Audit Board on all reviews and investigations conducted under items 4.f, 4.g, 4.h, and 4.i.

d. The Station Operations Review Committee shall make determinations regarding whether or not proposals considered by the Committee involve unreviewed safety questions. This determination shall be subject to review by the NPPD Safety Review and Audit Board.

6. Records:

Minutes shall be kept for all meetings of the Station Operations Review Committee and shall include identification of all documentary material reviewed; copies of the minutes shall be forwarded to the Chairman of the NPPD Safety Review and Audit Board and the Assistant General Manager - Nuclear within one month.

7. Procedures:

Written administrative procedures for Committee operation shall be prepared and maintained describing the method for submission and content of presentations to the committee, provisions for use of subcommittees, review and approval by members of written Committee evaluations and recommendations, dissemination of minutes, and such other matters as may be appropriate.

B. NPPD Safety Review and Audit Board (SRAB)

Function: The Board shall function to provide independent review and audit of designated activities.

1. Membership:

   a. Chairman

   b. Vice-Chairman

   c. Five Members

   d. Consultants (as required)

   The Board members shall collectively have the capability required to review problems in the following areas: nuclear power plant operations, nuclear engineering, chemistry and radiochemistry, metallurgy, instrumentation and control, radiological safety, mechanical and electrical engineering, quality assurance practices, and other appropriate fields associated with the unique characteristics of the nuclear power plant involved. When the nature of a particular problem dictates, special consultants will be utilized.

   Alternate members shall be appointed in writing by the Board Chairman to serve on a temporary basis; however, no more than two alternates shall serve on the Board at any one time.

2. Meeting frequency: Semiannually, and as required on call of the Chairman.

3. Quorum: Chairman or Vice Chairman, plus four members including alternates. No more than a minority of the quorum shall be from groups holding line responsibility for the operation of the plant.

4. Review: The following subjects shall be reported to and reviewed by the NPPD Safety Review and Audit Board.

   a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

   b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Appendix A Technical Specifications or the CNS Operating License.

e. Violations of applicable codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. All reportable events specified in Section 50.73 to 10CFR Part 50.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Minutes of meetings of the Station Operations Review Committee.

j. Disagreement between the recommendations of the Station Operations Review Committee and the Division Manager of Nuclear Operations.

k. Review of events covered under e,f,g, and h above include reporting to appropriate members of management on the results of investigations and recommendations to prevent or reduce the probability of recurrence.

5. Authority: The NPPD Safety Review and Audit Board shall report to and be advisory to the Assistant General Manager - Nuclear on those areas of responsibility specified in Specifications 6.2.1.B.4 and 6.2.1.B.7.

6. Records:

Minutes shall be recorded for all meetings of the NPPD Safety Review and Audit Board and shall identify all documentary material reviewed. Copies of the minutes shall be forwarded to the Assistant General Manager - Nuclear and the Division Manager of Nuclear Operations, and such others as the Chairman may designate within one month of the meeting.

7. Audits:

Audits of selected aspects of plant operation shall be performed under the cognizance of SRAB with a frequency commensurate with their safety significance. Audits performed by the Quality Assurance Department which meet this specification shall be considered to meet the SRAB audit requirements if the audit results are reviewed by SRAB. A representative portion of procedures and records of the activities performed during the audit period shall be audited and, in addition, observations of performance of operating and maintenance activities shall be included. These audits shall encompass:

a. Verification of compliance with internal rules, procedures (for example: normal, off-normal, emergency, operating, maintenance, surveillance, test, and radiation control procedures) and applicable license conditions at least once per 24 months.

b. The training, qualification, and performance of the operating staff at least once per 24 months.

c. The Emergency Plan and implementing procedures at least once per 12 months.

d. The Security Plan and implementing procedures at least once per 12 months.

e. The facility fire protection and its implementing procedures at least once per 24 months.

f. A fire protection and loss prevention inspection will be performed utilizing either qualified off-site licensee personnel or an outside fire protection consultant at least once per 12 months.

g. An inspection and audit by an outside qualified fire protection consultant shall be performed at least once per 36 months.

6.3 PROCEDURES AND PROGRAMS

6.3.1 Introduction

Station personnel shall be provided detailed written procedures to be used for operation and maintenance of system components and systems that could have an effect on nuclear safety.

6.3.2 Procedures

Written procedures and instructions including applicable check off lists shall be provided and adhered to for the following:

A. Normal startup, operation, shutdown and fuel handling operations of the station including all systems and components involving nuclear safety.

B. Actions to be taken to correct specific and foreseen potential or actual malfunctions of safety related systems or components including responses to alarms, primary system leaks and abnormal reactivity changes.

C. Emergency conditions involving possible or actual releases of radioactive materials.

D. Implementing procedures of the Security Plan and the Emergency Plan.

E. Implementing procedures for the fire protection program.

F. Administrative procedures for shift overtime.

6.3.3 Maintenance and Test Procedures

The following maintenance and test procedures will be provided to satisfy routine inspection, preventive maintenance programs, and operating license requirements.

A. Routine testing of Engineered Safeguards and equipment as required by the facility License and the Technical Specifications.

B. Routine testing of standby and redundant equipment.

C. Preventive or corrective maintenance of plant equipment and systems that could have an effect on nuclear safety.

D. Calibration and preventive maintenance of instrumentation that could affect the nuclear safety of the plant.

E. Special testing of equipment for proposed changes to operational procedures or proposed system design changes.

6.3.4 Radiation Control Procedures

Radiation control procedures shall be maintained and made available to all station personnel. These procedures shall show permissible radiation exposure, and shall be consistent with the requirements of 10 CFR 20.

1. A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, 1/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

2. A summary description of facility changes, tests or experiments in accordance with the requirements of 10CFR50.59(b).

3. Pursuant to 3.8.A, a report of radioactive source leak testing. This report is required only if the tests reveal the presence of 0.005 microcuries or more of removable contamination.

4. Documentation of all challenges to relief valves or safety valves.

D. Monthly Operating Report

Routine reports of operating statistics, shutdown experience, and a narrative summary of operating experience relating to safe operation of the facility, shall be submitted on a monthly basis to the individual designated in the current revision of Reg. Guide 10.1 no later than the tenth of each month following the calendar month covered by the report.

6.5.2 Reportable Events

A Reportable Event shall be any of those conditions specified in Section 50.73 to 10CFR Part 50. The NRC shall be notified and a report submitted pursuant to the requirements of Section 50.73. Each Reportable Event shall be reviewed by SORC and the results of this review shall be submitted to SRAB and the Assistant General Manager - Nuclear.

6.5.3 Unique Reporting Requirements

Reports shall be submitted to the Director, Nuclear Reactor Regulation, USNRC, Washington, D.C. 20555 as follows:

A. Reports on the following areas shall be submitted as noted:

None.

1/ This tabulation supplements the requirements of §20.407 of 10CFR Part 20.

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