Attachment 1

Revised Technical Specifications for
Fire Protection System
Clean Water Supply

Revised Pages: 216b
216c
216d
216k

Reference 1) Letter from J. M. Pilant to D. B. Vassallo dated June 28, 1982, "Fire Protection Rule 10CFR50, Appendix R"

As discussed in Sections 1.4 and 7.0 of Reference 1, the District is providing a Clean Water Fire Protection System for CNS which upgrades the existing system that takes suction from the Missouri River. This change is not an NRC requirement but is being performed with direction from the CNS insurance company. The electric and diesel fire pumps will be separate and independent in the modified system and the requirements of 10CFR50 Appendix R, and Branch Technical Position 9.5-1 Appendix A will be met.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.14 FIRE DETECTION SYSTEM	4.14 FIRE DETECTION SYSTEM
APPLICABILITY	APPLICABILITY
Applies to the operational status of the Fire Detection System.	Applies to the operational status of the Fire Detection System.
OBJECTIVE	
To assure continuous automatic surveillance throughout the Main Plant.	
SPECIFICATIONS	SPECIFICATIONS
A. The Fire Detection System instumentation for each fire detection zone shown in Table 3.14 shall be operable.	A. Each detector on Table 3.14 shall be demonstrated operable every 6 months by performance of a channel functional test.
B. With one or more of the fire detection instrument(s) shown in Table 3.14 inoperable: 1. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, and 2. Restore the inoperable instrument(s) to OPERABLE status within 14 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.7.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to OPERABLE status.	B. The NFPA Code 72.D Class B supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.
3.15 FIRE SUPPRESSION WATER SYSTEM	4.15 FIRE SUPPRESSION WATER SYSTEM
APPLICABILITY	APPLICABILITY
Applies to the availability of water for fire fighting purposes.	Applies to the availability of water for fire fighting purposes.
OBJECTIVE	
To assure a continuous operable water supply for fire fighting systems from 2 fire pumps.	

LIMITING CONDITIONS FOR OPERATION

3.15 (cont'd)

SPECIFICATIONS

A. The fire suppression water system shall be OPERABLE with:

1. Two fire pumps, each with a capacity of at least 2000 gpm, with their discharge aligned to the fire suppression header.

2. An OPERABLE flow path capable of taking suction from either of two 500,000 gallon water storage tanks or the Missouri River and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant valves and the front valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

B. If the requirement of 3.15.A cannot be met, restore the inoperable equipment to OPERABLE status within 7 days or prepare and submit a Special Report to the Commission pursuant to Specification 6.7.2 within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

C. With the fire suppression system inoperable:

1. Establish a backup fire suppression water system within 24 hours, and

2. Submit a Special Report in accordance with Specification 6.7.2;

a) By telephone within 24 hours, and

b) In writing no later than the first working day following the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.15 (cont'd)

SPECIFICATIONS

A. The Fire Suppression Water Supply System shall be demonstrated operable:

1. At least once per 31 days by starting each pump on a staggered start-up basis and operating it for:

a) A minimum of 15 minutes for a diesel engine-driven fire pump, and

b) A minimum of 7 minutes for an electrical motor-driven fire pump.

2. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

3. At least once per 12 months by cycling each testable valve in the flow path through at least one complete cycle of full travel.

4. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

a) Verifying that each automatic valve in the flow path actuates to its correct position on a test signal,

b) Verifying that each pump developes at least 2000 gpm with at least 110 psi,

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

4.15 (cont'd)

c) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle or full travel, and

d) Verifying that each high pressure pump starts (sequentially) to maintain the fire suppression water system pressure $\geq$ 65 psig.

5. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

6. The fire pump diesel engine shall be demonstrated OPERABLE:

a) At least once per 31 days by verifying;

1) The fuel storage tank contains at least 150* gallons of fuel, and

2) The diesel starts from ambient conditions and operates for at least 15 minutes.

b) At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM-D975-74 for viscosity water content and sediment.

c) At least once per 18 months by:

1) Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service, and

*This number shall become 250 gallons when the clean water fire protection system becomes operable.

INSTRUMENT LOCATION		INSTRUMENT ID NO.
2	Control Room	FP-SD-17-1 FP-SD-17-2 FP-SD-17-3
3	Cable Spreading Room	FP-SD-16-1 FP-SD-16-2 FP-SD-16-3 FP-SD-16-4 FP-SD-16-5 FP-SD-16-6
	Cable Expansion Room	FP-SD-16-7 FP-SD-16-8
4	Switchgear Rooms	
	DC Switchgear Rooms	FP-SD-15-2 FP-SD-15-3
	Critical Switchgear Room	FP-SD-22-1 FP-SD-22-2
5	Station Battery Rooms	FP-SD-15-1 FP-SD-15-4 FP-SD-15-1A FP-SD-15-4A
6	Diesel Generator Rooms	FP-SD-10-1 FP-SD-10-2 FP-SD-10-3 FP-SD-10-4 CO2-SD-DG-1A CO2-SD-DG-1B CO2-SD-DG-1C CO2-SD-DG-1D CO2-SD-DG-2A CO2-SD-DG-2B CO2-SD-DG-2C CO2-SD-DG-2D
7	Diesel Fuel Storage Rooms	CO2-TD-DG-1A CO2-TD-DG-1B
8	Safety Related Equipment not in Reactor Building	
	RHR Service Water Booster Pumps	FP-SD-14-3
	Emergency Condensate Storage Tanks	FP-SD-14-1
	Service Water Pumps	FP-FD-32-1 FP-FD-32-2
9	Auxiliary Relay Room & Reactor Protection System Rooms	
	Auxiliary Relay Room	FP-SD-15-9
	Reactor Protection System Room 1A	FP-SD-15-7
	Reactor Protection System Room 1B	FP-SD-15-8

Attachment 2

Revised Technical Specifications for
Scram Discharge Volume Modifications

Revised Pages: 28
33
40
61

The original 12" Scram Discharge Instrument Volume (SDIV) was set to initiate a scram at a level in the volume corresponding to $\leq$ 36 gallons. The 36 gallons was based on an availability consideration giving the operator 20 minutes to respond to an inadvertently closed drain valve assuming each control rod leaked 50 cc/minute. There are now two instrument volumes of approximately 22 gallons each, one for each group of hydraulic control units in the reactor building. Each group has approximately one-half of the hydraulic control units.

The new instrument volumes initiate alarms, rod blocks, and scrams at specified levels rather than volumes. A level transmitter or level switch measures level rather than volume. The surveillance program to provide functional checks of the SDIV level instrumentation is provided. Station procedures provide for periodic verification of the correlation between level and volume. An SDV not drained alarm has been established at $\leq$ 11½ inches. The references for all levels are the center lines of the lower instrument tap on each SDIV. The scram level for each instrument volume assures an adequate scram discharge volume exists so that all control rods can insert fully. It should be noted that CNS now has larger scram discharge volumes (excluding the instrument volumes) than existed before this modification.

COOPER NUCLEAR STATION
TABLE 3.1.1
REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Reactor Protection System Trip Function	Applicability Conditions Mode Switch Position: Shutdown	Startup	Refuel	Run	Trip Level Setting	Minimum Number of Operable Channels Per Trip Systems (1)	Action Requir When Equipmen Operability i Not Assured (
Mode Switch in Shutdown	X(7)	X	X	X		1	A
Manual Scram	X(7)	X	X	X		1	A
IRM (17) High Flux	X(7)	X	X	(5)	≤ 120/125 of indicated scale	3	A
Inoperative		X	X	(5)		3	A
APRM (17) High Flux (Flow biased)				X	≤ (0.66W+54%) $\frac{FRP}{MFLPD}$ (14)	2	A or C
High Flux	X(7)	X(9)	X(9)	(16)	≤ 15% Rated Power		A or C
Inoperative		X(9)	X(9)	X	(13)	2	A or C
Downscale		(11)		X(12)	≥ 2.5% of indicated scale	2	A or C
High Reactor Pressure NBI-PS-55 A,B,C, & D		X(9)	X(10)	X	≤ 1045 psig	2	A
High Drywell Pressure PC-PS-12 A,B,C, & D		X(9)(8)	X(8)	X	≤ 2 psig	2	A or D
Reactor Low Water Level NBI-LIS-101 A,B,C, & D		X	X	X	≥ + 12.5 in. indicated level	2	A or D
Scram Discharge Volume High Water Level CRD-LS-231 A & B CRD-LS-234 A & B CRD-LT-231 C & D CRD-LT-234 C & D		X	X(2)	X	≤ 92 inches	3	A

COOPER NUCLEAR STATION
TABLE 4.1.1 (Page 2)
REACTOR PROTECTION SYSTEM (SCRAM INSTRUMENTATION) FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

Instrument Channel	Group (2)	Functional Test	Minimum Frequency (3)
High Water Level in Scram Discharge Volume CRD-LS-231 A & B CRD-LS-234 A & B CRD-LT-231 C & D CRD-LT-234 C & D	A	Trip Channel and Alarm	Once/3 Months
Main Steam Line High Radiation RMP-RM-251 A,B,C, & D	B	Trip Channel and Alarm (4)	Once/Week
Main Steam Line Isolation Valve Closure MS-LMS-86 A,B,C, & D MS-LMS-80 A,B,C, & D	A	Trip Channel and Alarm	Once/Month (1)
Turbine Control Valve Fast Closure TGF-63/OPC -1,2,3,4	A	Trip Channel and Alarm	Once/Month (1)
Turbine First Stage Pressure Permissive MS-PS-14 A,B,C, & D	A	Trip Channel and Alarm	Once/3 Months
Turbine Stop Valve Closure SVOS-1 (1), SVOS-1 (2) SVOS-2 (1), SVOS-2 (2)	A	Trip Channel and Alarm	Once/Month (1)
Reactor Pressure Permissive NBI-PS-51 A,B,C & D	A	Trip Channel and Alarm	Once/3 Months

3.1 BASES (cont'd.)

against short reactor periods in these ranges.

The control rod drive scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. The scram discharge volume accommodates in excess of 36 gallons of water and is the low point in the piping. No credit was taken for this volume in the design of the discharge piping as concerns the amount of water which must be accommodated during a scram.

During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated which would result in slow scram times or partial control rod insertion. To preclude this occurrence, diverse indication (two level switches and two level transmitters for each discharge volume) has been provided in the instrument volumes which alarm and scram the reactor when the volume of water reaches 92 inches. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or amount of insertion of the control rods. This function shuts the reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function adequately.

A source range monitor (SRM) system is also provided to supply additional neutron level information during start-up but has no scram functions (reference paragraph VII.5.4 FSAR). Thus, the IRM and APRM are required in the "Refuel" and "Start/Hot Standby" modes. In the power range the APRM system provides required protection (refer-

4.1 BASES (cont'd.)

revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in the Group (B) devices to calculate their "unsafe" failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 20×10^{-6} failures/hour. The bi-stable trip circuits are predicted to have an unsafe failure rate of less than 2×10^{-6} failures/hour. Considering the two hour monitoring interval for the analog devices as assumed above, and a weekly test interval for the bi-stable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

The bi-stable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1.1. There are numerous identical bi-stable devices used throughout the plant's instrumentation system. Therefore, significant data on the failure rates for the bi-stable devices should be accumulated rapidly.

The frequency of calibration of the APRM Flow Biasing Network has been established as each refueling outage. The flow biasing network is functionally tested at least once per month and, in addition, cross calibration checks of the flow input to the flow biasing network can be made during the functional test by direct meter reading. There are several instruments which must be calibrated and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a

TABLE 3.2.C
CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Trip Level Setting	Minimum Number Of Operable Instrument Channels/Trip System(5)
APRM Upscale (Flow Bias)	$\leq (0.66W + 42\%) \frac{FRP}{MFLPD}$ (2)	2(1)
APRM Upscale (Startup)	$\leq$ 12%	2(1)
APRM Downscale (9)	$\geq$ 2.5%	2(1)
APRM Inoperative	(10b)	2(1)
RBM Upscale (Flow Bias)	$\leq$ (0.66W + 40%) (2)	1
RBM Downscale (9)	$\geq$ 2.5%	1
RBM Inoperative	(10c)	1
IRM Upscale (8)	$\leq$ 108/125 of Full Scale	3(1)
IRM Downscale (3)(8)	$\geq$ 2.5%	3(1)
IRM Detector Not Full In (8)		3(1)
IRM Inoperative (8)	(10a)	3(1)
SRM Upscale (8)	$\leq 1 \times 10^5$ Counts/Second	1(1)(6)
SRM Detector Not Full In (4)(8)	($\geq$ 100 cps)	1(1)(6)
SRM Inoperative (8)	(10a)	1(1)(6)
Flow Bias Comparator	$\leq$ 10% Difference In Recirc. Flows	1
Flow Bias Upscale/Inop.	$\leq$ 110% Recirc. Flow	1
SRM Downscale (8)(7)	$\geq$ 3 Counts/Second (11)	1(1)(6)
SDV Water Level High CRD-231E, 234E	$\leq$ 46 inches	1(12)

Attachment 3

Revised Technical Specification for
RPS Power Monitoring System

Revised Pages: 193
195
197
199

Reference 1) Letter from D. B. Vassallo to J. M. Pilant dated May 4, 1982, "Reactor Protection System (RPS) Power Monitoring System Design Modification"

2) Letter from D. B. Vassallo to J. M. Pilant dated July 8, 1982, same subject

During the Spring 1982 refueling outage, eight Class 1E Electrical Protection Assemblies (EPA's) were installed in the RPS power monitoring system. This change was in response to the concern that the original RPS was not seismically qualified and could degrade during a seismic event. The proposed Technical Specifications are in accordance with General Electric verified time delays as required in Reference 1 and the model Technical Specification of Reference 2. Please note that exception is taken to the Model Technical Specification surveillance requirement of a "channel functional" test every six months. This would require deenergizing each half of the RPS system either during the test or transfer to the alternate supply. This puts unwanted transients on critical equipment (especially the Main Steam Line Radiation Monitors) and induces an unnecessary risk of a plant scram. The 18-month test frequency proposed for the functional test and channel calibration is consistent with other Technical Specifications for electrical breakers in essential systems.

LIMITING CONDITIONS FOR OPERATION

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the auxiliary electrical power system.

Objective:

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification:

A. Auxiliary Electrical Equipment

The reactor shall not be made critical from a Cold Shutdown Condition unless all of the following conditions are satisfied:

1. Both off-site sources (345 KV and 69 KV) and the startup transformer and emergency transformer are available and capable of automatically supplying power to the 4160 Volt emergency buses 1F and 1G.

2. Both diesel generators shall be operable and there shall be a minimum of 45,000 gal. of diesel fuel in the fuel oil storage tanks.

3. The 4160V critical buses 1F and 1G and the 480V critical buses 1F and 1G are energized.

   a. The loss of voltage relays and their auxiliary relays are operable.

   b. The undervoltage relays and their auxiliary relays are operable.

4. The four unit 125V/250V batteries and their chargers shall be operable.

5. The power monitoring system for the inservice RPS MG set or alternate source shall be operable.

SURVEILLANCE REQUIREMENTS

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Auxiliary Electrical Equipment

1. Emergency Buses Undervoltage Relays

   a. Loss of voltage relays

      Once every 18 months, loss of voltage on emergency buses is simulated to demonstrate the load shedding from emergency buses and the automatic start of diesel generators.

   b. Undervoltage relays

      Once every 18 months, low voltage on emergency buses is simulated to demonstrate disconnection of the emergency buses from the offsite power source. The undervoltage relays shall be calibrated once every 18 months.

2. Diesel Generators

   a. Each diesel-generator shall be started manually and loaded to not less than 35% of rated load for no less than 2 hours once each month to demonstrate operational readiness.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.A	4.9.A (cont'd.)
	cell and overall battery voltage shall be measured and logged.
	b. Every three months the measurements shall be made of the voltage of each cell to nearest 0.1 Volt, specific gravity of each cell, and temperature of every sixth cell. These measurements shall be logged.
	c. Once each operating cycle, the stated batteries shall be subjected to a rated load discharge test. The specific gravity and voltage of each cell shall be determined after the discharge and logged.
B. Operation with Inoperable Equipment	4. Power Monitoring System for RPS System
Whenever the reactor is in Run Mode or Startup Mode with the reactor not in a Cold Condition, the availability of electric power shall be as specified in 3.9.A, except as specified in 3.9.B.1.	The above specified RPS power monitoring system instrumentation shall be determined operable:
1. From and after the date incoming power is not available from a startup or emergency transformer, continued reactor operation is permissible under this condition for seven days. At the end of this period, provided the second source of incoming power has not been made immediately available, the NRC must be notified of the event and the plan to restore this second source. During this period, the two diesel generators and associated critical buses must be demonstrated to be operable.	a. At least once per operating cycle by demonstrating the operability of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a channel calibration including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints.
2. From and after the date that incoming power is not available from both start-up and emergency transformers, continued operation is permissible, provided the two diesel generators and associated critical buses are demonstrated to be	1. Over-voltage $\leq$ 132 VAC, with time delay $\leq$ 2 sec.
	2. Under-voltage $\geq$ 108 VAC, with time delay $\leq$ 2 sec.
	3. Under-frequency $\geq$ 57 Hz. with time delay $\leq$ 2 sec.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.9.B.5 (cont'd.) From and after the date that one of the 125 or 250 volt battery systems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding ten days within electrical safety considerations, provided repair work is initiated in the most expeditious manner to return the failed component to an operable state, and Specifications 3.5.A.5 and 3.5.F are satisfied. The NRC shall be notified within 24 hours of the situation, the precautions to be taken during this period and the plans to return the failed components to an operable state.	4.9.B
6. With one RPS electric power monitoring channel for an inservice RPS MG set or alternate power supply inoperable, restore the inoperable channel to operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service.	
7. With both RPS electric power monitoring channels for an inservice RPS MG set or alternate power supply inoperable, restore at least one to operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.	

4.9 BASES

The monthly test of the diesel generator is conducted to check for equipment failures and deterioration. Testing is conducted up to equilibrium operating conditions to demonstrate proper operation at these conditions. The diesel generator will be manually started, synchronized and connected to the bus and load picked up. The diesel generator should be loaded to at least 35% of rated load to prevent fouling of the engine. It is expected that the diesel generator will be run for at least two hours. Diesel generator experience at other generating stations indicates that the testing frequency is adequate and provides a high reliability of operation should the system be required.

Each diesel generator has two air compressors and two air receivers for starting. It is expected that the air compressors will run only infrequently. During the monthly check of the diesel generator, each receiver in each set of receivers will be drawn down below the point at which the corresponding compressor automatically starts to check operation and the ability of the compressors to recharge the receivers.

The diesel generator fuel consumption rate at full load is approximately 275 gallons per hour. Thus, the monthly load test of the diesel generators will test the operation and the ability of the fuel oil transfer pumps to refill the day tank and will check the operation of these pumps from the emergency source.

The test of the diesel generator during the refueling outage will be more comprehensive in that it will functionally test the system; i.e, it will check diesel generator starting and closure of diesel generator breaker and sequencing of load on the diesel generator. The diesel generator will be started by simulation of a loss-of-coolant accident. In addition, an undervoltage condition will be imposed to simulate a loss of off-site power.

Periodic tests between refueling outages verify the ability of the diesel generator to run at full load and the core and containment cooling pumps to deliver full flow. Periodic testing of the various components, plus a functional test once-a-cycle, is sufficient to maintain adequate reliability.

Although station batteries will deteriorate with time, utility experience indicates there is almost no possibility of precipitous failure. The type of surveillance described in this specification is that which has been demonstrated over the years to provide an indication of a cell becoming irregular or unserviceable long before it becomes a failure. In addition, the checks described also provide adequate indication that the batteries have the specified ampere-hour capability.

The diesel fuel oil quality must be checked to ensure proper operation of the diesel generators. Water content should be minimized because water in the fuel could contribute to excessive damage to the diesel engine.

When it is determined that some auxiliary electrical equipment is out of service, the increased surveillance required in Section 4.5.F is deemed adequate to provide assurance that the remaining equipment will be operable.

The Reactor Protection System (RPS) is equipped with a seismically qualified, Class 1E power monitoring system. This system consists of eight Electrical Protection Assemblies (EPA) which isolate the power sources from the RPS if the input voltage and frequency are not within limits specified for safe system operation. Isolation of RPS power causes that RPS division to fail safe.

Attachment 4

Revised Technical Specification for
Plant Staff Working Hours

Revised Page: 226

Generic Letter 82-12 dated June 15, 1982, stated:

> "Our letter of February 8, 1982, requested that you take action as necessary to revise the administrative section of your technical specifications to assure that your plant administrative procedures follow the revised working hour guidelines, including a provision for documentation of authorized deviations which should be available for NRC review. You should review your past actions to assure that they are consistent with the attached revised policy statement. Note that the revised guidelines are to be incorporated by October 1, 1982."

In discussions with the Staff, the District was directed to revise the Technical Specifications to state that working hours will be controlled in accordance with a CNS Station Operating Procedure. This proposed change is attached.

6.3 Station Operating Procedures

6.3.1 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 Written integrated and system 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 forseen 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 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 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.

Attachment 5

Revised Technical Specification for
SRAB Duties

Revised Page: 222

In the recent Amendment 80 to the License, in Specification 6.2.1.B, the word "approved" was changed to "approve" as an apparent typographical error. As a recent I&E inspection report pointed out, this minor alteration actually changed the duties which SRAB must perform. The word "approve" is being changed back to "approved" because it is now clear that this word was not a typographical error as previously thought.

tary material reviewed; copies of the minutes shall be forwarded to the Chairman of the NPPD Safety Review and Audit Board and the Division Manager of Power Operations 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.

The board must: verify that operation of the plant is consistent with company policy and rules, approved operating procedures and operating license provisions; review safety related plant changes, proposed tests and procedures; verify that unusual events are promptly investigated and corrected in a manner which reduces the probability of recurrence of such events; and detect trends which may not be apparent to a day-to-day observer.

Audits of selected aspects of plant operation shall be performed with a frequency commensurate with their safety significance and in such a manner as to assure that an audit of all nuclear safety related activities is completed within a period of two years. Periodic review of the audit programs should be performed by the Board at least twice a year to assure that such audits are being accomplished in accordance with requirements of Technical Specifications. The audits shall be performed in accordance with appropriate written instructions or procedures and should include verification of compliance with internal rules, procedures (for example, normal, off-normal, emergency, operating, maintenance, surveillance, test and radiation control procedures and the emergency and security plans), regulations involving nuclear safety and operating license provisions; training, qualification and performance of operating staff; and corrective actions following abnormal occurrences or unusual events. 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. Written reports of such audits shall be reviewed at a scheduled meeting of the Board and by appropriate members of management including those having responsibility in the area audited. Follow-up action, including reaudit of deficient areas, shall be taken when indicated.

In addition to the above, the Safety Review and Audit Board will audit the facility fire protection and its implementing procedures at least once every 24 months.

Attachment 6

Revised Technical Specification for
Listing of Snubbers

Revised Pages: 137a
137b
137e
137f-137m

The current Technical Specifications for Cooper Nuclear Station lists snubbers under three different categories on Tables 3.6.1, 3.6.2., and 3.6.3.

Nebraska Public Power District requests a revision to the Technical Specifications as shown on the attached pages. In addition to revising the existing tables, this request will add a new table, Table 3.6.4, which lists Inaccessible Safety Related Hydraulic Shock Suppressors (Snubbers).

This request is made for reasons as follows:

(a) In order to add a new category (table).

(b) To have a listing that is compatible with what is contained in the District's computer system and to facilitate data withdrawals and entries.

(c) Some snubber listings have been added/deleted as a result of modifications to Torus Attached Piping.

(d) Locations given in the tables were not entirely accurate and needed to be more specific.

This change will provide revised tables that are functionally superior to the existing tables.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.6.H Shock Suppressors (Snubbers)

1. During all modes of operation except Cold Shutdown and Refuel, all safety related snubbers shall be operable except as noted in 3.6.H.2 through 3.6.H.5 below.

2. The snubbers listed in Tables 3.6.1, 3.6.2, 3.6.3, and 3.6.4 are required to protect the primary coolant system or other safety related systems or components. All others are therefore exempt from these specifications.

3. With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.6.H.4 on the supported component or declare the supported system or subsystem inoperable and follow the appropriate ACTION statement for that system.

4. If a snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

5. Snubbers may be added to, removed, or substituted for, by analysis, from safety related systems without prior License Amendment to Tables 3.6.1, 3.6.2, 3.6.3, and 3.6.4, provided that a revision to these tables is included with a subsequent License Amendment request.

4.6.H Shock Suppressors (Snubbers)

The following surveillance requirements apply to all snubbers listed in Tables 3.6.1, 3.6.2, 3.6.3, and 3.6.4.

1. All snubbers shall be visually inspected in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
8 or more	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation and by type, hydraulic or mechanical. These four groups may be inspected independently according to the above schedule.

2. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

4.6.H Shock Suppressors (Snubbers) (cont'd)

structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; or (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.6.H.6 or 4.6.H.7 as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

3. At least once per 18 months during shutdown, a representative sample, 10% of the total of each type of snubber in use in the plant, shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.6.H.5 or 4.6.H.6, an additional 10% of that type of snubber shall be functionally tested.

4. The representative sample selected for functional testing shall include various configuration, operating environments and the range of size and capacity of snubbers. Tables 3.6.1, 3.6.2, 3.6.3, and 3.6.4 may be used jointly or separately as the basis for the sampling plan.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.6.H Shock Suppressors (Snubbers) (cont'd) Concurrent with the first in-service visual inspection and at least once per 18 months thereafter, the installation and maintenance records of each snubber listed in Tables 3.6.1, 3.6.2, 3.6.3, and 3.6.4 shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

Table 3.6.1

ACCESSIBLE SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS)

Snubber	Location
CS-SNUB-(CS-S1)	R-903-SE
CS-SNUB-(CS-S10)	R-931-NE
CS-SNUB-(CS-S11)	R-931-NE
CS-SNUB-(CS-S2)	R-903-SE
CS-SNUB-(CS-S3)	R-931-SE
CS-SNUB-(CS-S6)	R-881-SE QUAD
CS-SNUB-(CS-S7)	R-881-SE QUAD
CS-SNUB-(CS-VE7)	R-881-SE QUAD
HPCI-SNUB-(HP-S11)	R-859-HPCI RM
HPCI-SNUB-(HP-S15)	R-859-HPCI RM
HPCI-SNUB-(HP-S18)	R-859-SW QUAD
HPCI-SNUB-(HP-S18A)	R-859-HPCI RM
HPCI-SNUB-(HP-S22A)	R-859-HPCI RM
HPCI-SNUB-(HP-S4)	R-859-SW QUAD
HPCI-SNUB-(RF-S3)	R-859-HPCI RM
HPCI-SNUB-(RF-S4)	R-881-SW TORUS
HPCI-SNUB-(RF-S5)	R-881-SW TORUS
MS-SNUB-(BS-S1)	R-881-SW TORUS
MS-SNUB-(BS-S113A)	R-881-NW TORUS
MS-SNUB-(BS-S113B)	R-881-NW TORUS
MS-SNUB-(BS-S116A)	R-881-NW TORUS
MS-SNUB-(BS-S116B)	R-881-NW TORUS
MS-SNUB-(BS-S125A)	R-881-SW TORUS
MS-SNUB-(BS-S125B)	R-881-SW TORUS
MS-SNUB-(BS-S2)	R-881-SW TORUS
MS-SNUB-(BS-S3)	R-881-NW TORUS
MS-SNUB-(BS-S4)	R-903-A RHR HX RM
MS-SNUB-(BS-S5)	R-903-A RHR HX RM
MS-SNUB-(MS-S1)	R-859-HPCI RM
MS-SNUB-(MS-S10)	R-881-SW TORUS
MS-SNUB-(MS-S11)	R-881-SW TORUS
MS-SNUB-(MS-S111A)	R-903-A RHR HX RM
MS-SNUB-(MS-S11A)	R-881-SW TORUS
MS-SNUB-(MS-S12)	R-881-SW TORUS
MS-SNUB-(MS-S12A)	R-881-SW TORUS
MS-SNUB-(MS-S13)	R-903-B RHR HX RM
MS-SNUB-(MS-S13A)	R-903-B RHR HX RM
MS-SNUB-(MS-S13B)	R-903-B RHR HX RM
MS-SNUB-(MS-S14)	R-903-B RHR HX RM
MS-SNUB-(MS-S15)	R-931-B RHR HX RM
MS-SNUB-(MS-S15A)	R-931-B RHR HX RM
MS-SNUB-(MS-S16A)	R-881-NW TORUS
MS-SNUB-(MS-S16B)	R-881-NW TORUS
MS-SNUB-(MS-S17)	R-903-A RHR HX RM
MS-SNUB-(MS-S18)	R-903-A RHR HX RM
MS-SNUB-(MS-S19)	R-903-A RHR HX RM
MS-SNUB-(MS-S2)	R-859-HPCI RM
MS-SNUB-(MS-S20)	R-931-A RHR HX RM
MS-SNUB-(MS-S20A)	R-931-A RHR HX RM
MS-SNUB-(MS-S23)	R-881-NE TORUS
MS-SNUB-(MS-S24)	R-881-NE TORUS
MS-SNUB-(MS-S25)	R-859-NE QUAD
MS-SNUB-(MS-S26)	R-859-NE QUAD

Table 3.6.1

ACCESSIBLE SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS) (cont'd)

Snubber	Location
MS-SNUB-(MS-S3)	R-859-HPCI RM
MS-SNUB-(MS-S4)	R-859-HPCI RM
MS-SNUB-(MS-S75)	R-931-A RHR HX RM
MS-SNUB-(MS-S76)	R-931-B RHR HX RM
MS-SNUB-(MS-S7A)	R-859-HPCI RM
MS-SNUB-(MS-S7B)	R-859-HPCI RM
MS-SNUB-(MS-S8)	R-881-SW TORUS
RCIC-SNUB-(RF-S1)	R-881-NE QUAD
RCIC-SNUB-(RF-S1A)	R-881-NE QUAD
RCIC-SNUB-(RF-S45C)	R-881-NE QUAD
RCIC-SNUB-(RF-S45D)	R-881-NE QUAD
RCIC-SNUB-(RF-S46A)	R-881-NE QUAD
RCIC-SNUB-(RF-S51A)	R-881-NE TORUS
RCIC-SNUB-(RF-S51B)	R-881-NE TORUS
REC-SNUB-(RCC-S20)	R-931-NW
REC-SNUB-(RCC-S21)	R-931-NW
REC-SNUB-(RCC-S22)	R-931-SW
REC-SNUB-(RCC-S3)	R-931-NE
REC-SNUB-(RCC-S4)	R-931-NE
RF-SNUB-(RF-S2)	R-881-NE TORUS
RF-SNUB-(RF-S6)	R-881-SE TORUS
RHR-SNUB-(RH-S103A)	R-859-SW QUAD
RHR-SNUB-(RH-S107A)	R-859-NW QUAD
RHR-SNUB-(RH-S20)	R-903-INJ V RM
RHR-SNUB-(RH-S21)	R-903-INJ V RM
RHR-SNUB-(RH-S22)	R-881-NW TORUS
RHR-SNUB-(RH-S23)	R-881-NW TORUS
RHR-SNUB-(RH-S24)	R-881-NW TORUS
RHR-SNUB-(RH-S25)	R-903-NW
RHR-SNUB-(RH-S25A)	R-903-NW
RHR-SNUB-(RH-S26)	R-903-NW
RHR-SNUB-(RH-S27A)	R-931-A RHR HX RM
RHR-SNUB-(RH-S29)	R-903-INJ V RM
RHR-SNUB-(RH-S30A)	R-881-SW TORUS
RHR-SNUB-(RH-S30B)	R-881-SW TORUS
RHR-SNUB-(RH-S32)	R-881-SW TORUS
RHR-SNUB-(RH-S33D)	R-881-NW TORUS
RHR-SNUB-(RH-S34)	R-903-SW
RHR-SNUB-(RH-S35)	R-903-B RHR HX RM
RHR-SNUB-(RH-S36)	R-903-B RHR HX RM
RHR-SNUB-(RH-S37)	R-903-B RHR HX RM
RHR-SNUB-(RH-S38)	R-903-B RHR HX RM
RHR-SNUB-(RH-S39)	R-903-B RHR HX RM
RHR-SNUB-(RH-S40)	R-903-B RHR HX RM
RHR-SNUB-(RH-S41)	R-859-SW QUAD
RHR-SNUB-(RH-S42)	R-859-SW QUAD
RHR-SNUB-(RH-S43)	R-881-SW TORUS
RHR-SNUB-(RH-S44)	R-881-SW QUAD
RHR-SNUB-(RH-S45)	R-881-SW QUAD
RHR-SNUB-(RH-S48)	R-881-NW QUAD
RHR-SNUB-(RH-S49)	R-881-NW QUAD
RHR-SNUB-(RH-S51)	R-903-A RHR HX RM
RHR-SNUB-(RH-S52)	R-903-A RHR HX RM

Table 3.6.1

ACCESSIBLE SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS) (cont'd)

Snubber	Location
RHR-SNUB-(RH-S54)	R-859-NW QUAD
RHR-SNUB-(RH-S55)	R-859-NW QUAD
RHR-SNUB-(RH-S56)	R-903-A RHR HX RM
RHR-SNUB-(RH-S57)	R-903-A RHR HX RM
RHR-SNUB-(RH-S59)	R-881-NW TORUS
RHR-SNUB-(RH-S65)	R-881-SW QUAD
RHR-SNUB-(RH-S66)	R-903-INJ V RM
RHR-SNUB-(RH-S76A)	R-881-SW TORUS
RHR-SNUB-(RH-S76B)	R-881-SW TORUS
RHR-SNUB-(RH-S77)	R-881-SW TORUS
RHR-SNUB-(RH-S78A)	R-881-NW TORUS
RHR-SNUB-(RH-S78B)	R-881-NW TORUS
RHR-SNUB-(RH-S80)	R-881-NW QUAD
RHR-SNUB-(RH-S96A)	R-903-NW
RHR-SNUB-(RH-S98)	R-881-NW QUAD
RWCU-SNUB-(CU-S89)	R-881-SE TORUS
SW-SNUB-(SW-H23A)	IS-SWP RM
SW-SNUB-(SW-H23D)	IS-SWP RM
SW-SNUB-(SW-H23E)	IS-SWP RM
SW-SNUB-(SW-H23H)	IS-SWP RM

Table 3.6.2

ACCESSIBLE SAFETY RELATED MECHANICAL SHOCK SUPPRESSORS (SNUBBERS)

Snubber	Location
MS-SNUB-(MS-S149B)	R-903-STM TUNNEL
MS-SNUB-(MS-S16)	R-881-NW TORUS
MS-SNUB-(MS-S9A)	R-881-SW TORUS
MS-SNUB-(MS-S9B)	R-881-SW TORUS
RCIC-SNUB-(RF-S51C)	R-881-NE TORUS
RHR-SNUB-(RH-S58)	R-903-A RHR HX RM
SGT-SNUB-(PSSP-40)	R-881-SW TORUS
SGT-SNUB-(PSSP-74)	R-881-SW TORUS
SW-SNUB-(SW-H23B)	IS-SWP RM
SW-SNUB-(SW-H23C)	IS-SWP RM
SW-SNUB-(SW-H23F)	IS-SWP RM
SW-SNUB-(SW-H23G)	IS-SWP RM

Table 3.6.3

INACCESSIBLE SAFETY RELATED MECHANICAL SHOCK SUPPRESSORS (SNUBBERS)

Snubber	Location
CS-SNUB-(CS-S4)	DW-934
CS-SNUB-(CS-S5)	DW-934
CS-SNUB-(CS-S8)	DW-934
CS-SNUB-(CS-S9)	DW-934
MS-SNUB-(MS-S21)	DW-901
MS-SNUB-(MS-S22)	DW-901
MS-SNUB-(MS-S63)	DW-921
MS-SNUB-(SS-A2)	DW-921
MS-SNUB-(SS-A3)	DW-921
MS-SNUB-(SS-B2)	DW-921
MS-SNUB-(SS-B3)	DW-921
MS-SNUB-(SS-C2)	DW-921
MS-SNUB-(SS-C3)	DW-921
MS-SNUB-(SS-D2)	DW-921
MS-SNUB-(SS-D3)	DW-921
MS-SNUB-(VR-55-23-X)	DW-901
MS-SNUB-(VR-55-26-Z)	DW-901
MS-SNUB-(VR-55-9-Y)	DW-901
MS-SNUB-(VR-55-9-Z)	DW-901
MS-SNUB-(VR-56-12-Y)	DW-901
MS-SNUB-(VR-56-24-X)	DW-901
MS-SNUB-(VR-58-12-Y)	DW-921
MS-SNUB-(VR-59-7-X)	DW-921
MS-SNUB-(VR-59-7-Z)	DW-901
MS-SNUB-(VR-60-7-X)	DW-921
MS-SNUB-(VR-60-7-Z)	DW-901
MS-SNUB-(VR-61-17-X)	DW-901
MS-SNUB-(VR-61-8-X)	DW-901
MS-SNUB-(VR-61-8-Z)	DW-921
MS-SNUB-(VR-62-17-X)	DW-901
MS-SNUB-(VR-62-8-X)	DW-901
MS-SNUB-(VR-62-8-Z)	DW-921
MS-SNUB-(VR-H61D)	DW-888
MS-SNUB-(VR-H62B)	DW-888
MS-SNUB-(VR-H62C)	DW-888
MS-SNUB-(VR-H63B)	DW-888
MS-SNUB-(VR-H63C)	DW-888
MS-SNUB-(VR-H64D)	DW-888
MS-SNUB-(VR-S1)	DW-901
MS-SNUB-(VR-S10)	DW-901
MS-SNUB-(VR-S11)	DW-921
MS-SNUB-(VR-S12)	DW-901
MS-SNUB-(VR-S14)	DW-888
MS-SNUB-(VR-S2)	DW-901
MS-SNUB-(VR-S20)	DW-921
MS-SNUB-(VR-S21)	DW-921
MS-SNUB-(VR-S22)	DW-901
MS-SNUB-(VR-S23A)	DW-901
MS-SNUB-(VR-S23B)	DW-901
MS-SNUB-(VR-S24A)	DW-901
MS-SNUB-(VR-S24B)	DW-901
MS-SNUB-(VR-S25)	DW-901

Table 3.6.3

INACCESSIBLE SAFETY RELATED MECHANICAL SHOCK SUPPRESSORS (SNUBBERS) (cont'd)

Snubber	Location
MS-SNUB-(VR-S26)	DW-888
MS-SNUB-(VR-S27)	DW-901
MS-SNUB-(VR-S3)	DW-888
MS-SNUB-(VR-S30)	DW-921
MS-SNUB-(VR-S31)	DW-921
MS-SNUB-(VR-S32)	DW-888
MS-SNUB-(VR-S4)	DW-901
MS-SNUB-(VR-S40)	DW-921
MS-SNUB-(VR-S41)	DW-921
MS-SNUB-(VR-S42A)	DW-921
MS-SNUB-(VR-S42B)	DW-921
MS-SNUB-(VR-S43)	DW-888
MS-SNUB-(VR-S50A)	DW-921
MS-SNUB-(VR-S50B)	DW-921
MS-SNUB-(VR-S51)	DW-888
MS-SNUB-(VR-S5A)	DW-901
MS-SNUB-(VR-S5B)	DW-901
MS-SNUB-(VR-S6)	DW-901
MS-SNUB-(VR-S60)	DW-921
MS-SNUB-(VR-S61)	DW-921
MS-SNUB-(VR-S62A)	DW-921
MS-SNUB-(VR-S62B)	DW-921
MS-SNUB-(VR-S63)	DW-921
MS-SNUB-(VR-S70A)	DW-901
MS-SNUB-(VR-S70B)	DW-901
MS-SNUB-(VR-S71A)	DW-901
MS-SNUB-(VR-S71B)	DW-901
MS-SNUB-(VR-S72)	DW-901
MS-SNUB-(VR-S73)	DW-901
MS-SNUB-(VR-S74)	DW-901
MS-SNUB-(VR-S7A)	DW-888
MS-SNUB-(VR-S7B)	DW-888
MS-SNUB-(VR-S8)	DW-888
MS-SNUB-(VR-S80)	DW-901
MS-SNUB-(VR-S81)	DW-901
MS-SNUB-(VR-S82)	DW-901
MS-SNUB-(VR-S83A)	DW-901
MS-SNUB-(VR-S83B)	DW-901
MS-SNUB-(VR-S84)	DW-901
MS-SNUB-(VR-S85)	DW-901
MS-SNUB-(VR-S86A)	DW-901
MS-SNUB-(VR-S86B)	DW-901
MS-SNUB-(VR-S87A)	DW-888
MS-SNUB-(VR-S87B)	DW-888
MS-SNUB-(VR-S88)	DW-888
RF-SNUB-(RF-S10)	DW-921
RF-SNUB-(RF-S11)	DW-921
RF-SNUB-(RF-S12)	DW-921
RF-SNUB-(RF-S13)	DW-921
RF-SNUB-(RF-S14)	DW-921
RF-SNUB-(RF-S15)	DW-921
RF-SNUB-(RF-S16)	DW-921

Table 3.6.3

INACCESSIBLE SAFETY RELATED MECHANICAL SHOCK SUPPRESSORS (SNUBBERS) (cont'd)

Snubber	Location
RF-SNUB-(RF-S17)	DW-921
RF-SNUB-(RF-S18)	DW-921
RF-SNUB-(RF-S19)	DW-921
RF-SNUB-(RF-S8)	DW-921
RF-SNUB-(RF-S9)	DW-921
RHR-SNUB-(RH-S10)	DW-901
RHR-SNUB-(RH-S11)	DW-901
RHR-SNUB-(RH-S13)	DW-921
RHR-SNUB-(RH-S14)	DW-921
RHR-SNUB-(RH-S15)	DW-921
RHR-SNUB-(RH-S16)	DW-901
RHR-SNUB-(RH-S17)	DW-901
RHR-SNUB-(RH-S18)	DW-901
RHR-SNUB-(RH-S19)	DW-901
RHR-SNUB-(RH-S3)	DW-FLG AREA
RHR-SNUB-(RH-S4)	DW-FLG AREA
RHR-SNUB-(RH-S5)	DW-921
RHR-SNUB-(RH-S6)	DW-921
RHR-SNUB-(RH-S67)	DW-901
RHR-SNUB-(RH-S68)	DW-901
RHR-SNUB-(RH-S69A)	DW-901
RHR-SNUB-(RH-S69B)	DW-901
RHR-SNUB-(RH-S7)	DW-921
RHR-SNUB-(RH-S70)	DW-901
RHR-SNUB-(RH-S71)	DW-901
RHR-SNUB-(RH-S72)	DW-901
RHR-SNUB-(RH-S72A)	DW-901
RHR-SNUB-(RH-S73)	DW-901
RHR-SNUB-(RH-S8A)	DW-901
RHR-SNUB-(RH-S8B)	DW-901
RHR-SNUB-(RH-S9)	DW-901
RR-SNUB-(SS-1A)	DW-888
RR-SNUB-(SS-1B)	DW-888
RR-SNUB-(SS-2A)	DW-888
RR-SNUB-(SS-2B)	DW-888
RR-SNUB-(SS-3A1)	DW-901
RR-SNUB-(SS-3A2)	DW-901
RR-SNUB-(SS-3B1)	DW-901
RR-SNUB-(SS-3B2)	DW-901
RR-SNUB-(SS-4A)	DW-901
RR-SNUB-(SS-4B)	DW-901
RR-SNUB-(SS-5A)	DW-888
RR-SNUB-(SS-5B)	DW-888
RR-SNUB-(SS-8A1)	DW-901
RR-SNUB-(SS-8A2)	DW-901
RWCU-SNUB-(CU-S3A)	DW-921
RWCU-SNUB-(CU-S3B)	DW-921

Table 3.6.4

INACCESSIBLE SAFETY RELATED HYDRAULIC SHOCK SUPPRESSORS (SNUBBERS)

Snubber	Location
RR-SNUB-(SS-7A1)	DW-901
RR-SNUB-(SS-7A2)	DW-901
RR-SNUB-(SS-7B1)	DW-901
RR-SNUB-(SS-7B2)	DW-901

Attachment 7

Revised Technical Specification for
SBGT Testing Requirement

Revised Pages: 165
182
183
215
215d
215e

References 1) ANSI N510-1980

2) NRC Inspection Report 50-298/82-02, Item 10.a(2)

3) Regulatory Guide 1.52

The current Technical Specifications for Cooper Nuclear Station lists requirements for the testing of charcoal filters in the Standby Gas Treatment System and the Control Room Ventilation System.

Nebraska Public Power District requests a revision to the Technical Specifications as shown on the attached pages. This request is made in order to bring testing criteria for filters, as contained in the Technical Specifications, into line with current industry standards and guidance (References 1 and 3). In relation to this request, Messrs. L. Wilborn and B. Murray conducted a routine inspection (Reference 2) on January 11-15, 1982. Their inspection report also recommended this change in Item 10.a(2).

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7. (cont'd.)	4.7 (cont'd.)
B. Standby Gas Treatment System	B. Standby Gas Treatment System
1. Except as specified in 3.7.B.3 below, both circuits of the standby gas treatment system and the diesel generators required for operation of such circuits shall be operable at all times when secondary containment integrity is required.	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 results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$99% DOP removal and $\geq$99% halogenated hydrocarbon removal.	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. The results of laboratory carbon sample analysis shall show $\geq$99% radioactive methyl iodide removal at a velocity within 20 percent of actual system design, $\geq$1.75 mg/m^3 inlet methyl iodide concentration, $\geq$70% R.H. and $\leq$30°F.	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. Fans shall be shown to operate within $\pm$10% design flow.	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 circuit 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. From and after the date that one circuit of the 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 circuit is sooner made operable, provided that during such seven days all active components of the other standby gas treatment circuit shall be operable.	3. System drains where present shall be inspected quarterly for adequte 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 a HEPA efficiency of at least 99 percent removal of DOP particulates. 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 efficiencies 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. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

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 circuit 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

with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. 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.

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.12 Additional Safety Related Plant Capabilities	4.12 Additional Safety Related Plant Capabilities
Applicability:	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.	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:	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.	To verify that operability or availability under conditions for which these capabilities are an essential response to station abnormalities.
A. Main Control Room Ventilation	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.	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 results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal absorber banks shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal.	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. The results of laboratory carbon sample analysis shall show $\geq$ 99% radioactive methyl iodide removal at a velocity within 20% of system design, 1.75 mg/m^3 inlet iodide concentration, $\geq$ 95% R.H. and $\leq$30°F.	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. Fans shall be shown to operate within $\pm$ 10% design flow.	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 a HEPA efficiency of at least 99 percent removal of DOP particulates. 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 efficiencies 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. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is not 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

3.12 BASES (cont'd)

heat exchanger. Valves are included in the common discharge header to permit the Seismic Class I service water system to be operated as two independent loops. The heat exchangers are valved such that they can be individually backwashed without interrupting system operation.

During normal operation two or three pumps will be required. Three pumps are used for a normal shutdown.

The loss of all a-c power will trip all operating service water pumps. The automatic emergency diesel generator start system and emergency equipment starting sequence will then start one selected service water pump in 30-40 seconds. In the meantime, the drop in service water header pressure will close the turbine building cooling water isolation valve guaranteeing supply to the reactor building, the control room basement, and the diesel generators from the one service water pump.

Due to the redundance of pumps and the requirement of only one to meet the accident requirements, the 30 day repair time is justified.

D. Battery Room Ventilation

The temperature rise and hydrogen buildup in the battery rooms without adequate ventilation is such that continuous safe operation of equipment in these rooms cannot be assured.

4.12 BASES

A. Main Control Room Ventilation System

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. Pressure drop should be determined at least once per operating cycle to show system performance capability.

Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant should be performed in accordance with ANSI N510-1980.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. 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 with an adsorbent qualified according to Table 1 of Regulatory Guide 1.52. The replacement tray for the absorber 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 HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture built up on the adsorber.

Attachment 8

Revised Technical Specification for
10CFR50 Appendix J Testing

Revised Pages:	159	176	
	160	177	
	161	178	
	162	184	
	162a	Deleted:	159a
	163		178a
	164		183a
	168		
	173		

Reference 1) Letter from D. G. Eisenhut to J. M. Pilant dated September 3, 1982, "Exemption to Appendix J to 10CFR Part 50 and Safety Evaluation Report"

This change is in response to Reference 1. The feedwater check valves will be tested with air or nitrogen as required.

The requested Technical Specifications for the containment air locks are proposed. It should be noted that the SER of Reference 1 stated that "test performance requires shutting down the reactor and opening the equipment hatch in order to install a strongback on the inner airlock door . . ." It is true that the strongback has to be attached to the inside door of the containment airlock in order to pressurize the airlock to accident pressure (Pa), but the drywell does not have to be entered to install the strongback because the strongback is stored inside the containment airlock. To attach the strongback during full power operation will expose personnel to radiation levels of approximately 500 mr/hr. Total exposure for this testing is estimated to be approximately 2 rem, which the District feels is excessive. Even though it is not necessary to enter the drywell to do the containment airlock test at Pa, the District still followed the NRC recommendation in the proposed Technical Specification.

The District will calculate a new correlation of reduced pressure leakage rates to full pressure leakage rate for the bellows leakage test and the containment airlock test. The calculation will be based on the Franklin Research Center Technical Evaluation Report, Appendix A, Procedure 'B'. This calculation will be added to the appropriate procedure for the local leak rate tests.

Since resolution of the above three issues concludes the Staff's review of CNS as regards Appendix J, minor format changes are being proposed for Section 3/4.7 of the Technical Specifications which should make this section easier to utilize.

The action on the initiating signal for the Reactor Water Sample Valves in Table 3.7.1 (page 168) is being changed since the valves actually go closed on a signal.

LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. Suppression Pool

At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2. and 3.5.F.5.

a. Minimum water volume - 87,650 ft^3

b. Maximum water volume - 91,000 ft^3

c. Maximum suppression pool temperature during normal power operation - 90^oF. For 45 days, commencing July 16, 1981, the suppression pool temperature may be increased to 95° whenever the river water temperature is such that the pool temperature cannot be maintained below 90^oF.

d. During testing which adds heat to the suppression pool, the water temperature shall not exceed 10^oF above the normal power operation limit specified in c. above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in c. above within 24 hours.

e. The reactor shall be scrammed from any operating condition if the pool temperature reaches 110^oF. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in c. above.

SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. Suppression Pool

a. The suppression pool water level and temperature shall be checked once per day.

b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160^oF or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

3.7.A.1 (cont'd.)

f. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.

2. Containment Integrity

Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

4.7.A (cont'd.)

2. Leak Rate Testing

a. Integrated leak rate tests (ILRT's) shall be performed to verify primary containment integrity. Primary containment integrity is confirmed if the leakage rate does not exceed the equivalent of 0.635 percent of the primary containment volume per 24 hours at 58 psig.

b. Integrated leak rate tests may be performed at either 58 psig or 29 psig, the leakage rate test period, extending to 24 hours of retained internal pressure. If it can be demonstrated to the satisfaction of those responsible for the acceptance of the containment structure that the leakage rate can be accurately determined during a shorter test period, the agreed-upon shorter period may be used.

Prior to initial operation, integrated leak rate tests must be performed at 58 and 29 psig (with the 29 psig test being performed prior to the 58 psig test) to establish the allowable leak rate, L_t (in percent of containment volume per 24 hours) at 29 psig as the lesser of the following values:

(L_a is 0.635 percent)

$$L_t = 0.635 \frac{L_{tm}}{L_{am}}$$

for $\frac{L_{tm}}{L_{am}} \leq 0.7$

where

L_{tm} = measured ILR at 29 psig

L_{am} = measured ILR at 58 psig, and

$\frac{L_{tm}}{L_{am}} \leq 1.0$

$$L_t = 0.635 \frac{P_t}{P_a}^{1/2}$$

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A (cont'd.)	4.7.A.2.b. (cont'd.)

where

P_a = peak accident pressure, 58 psia

P_t = appropriately measured test pressures (psia)

for $\frac{L_{tm}}{L_{am}} > 0.7$

c. The ILRT's shall be performed at the following minimum frequency:

1. Prior to initial unit operation.

2. At approximately three and one-third year intervals so that any ten-year interval would include four ILRT's. These intervals may be extended up to eight months if necessary to coincide with refueling outage.

d. The measured leakage rates, L_{tm} and L_{am}, shall be less than 0.75 L_t and 0.75 L_a for the reduced pressure tests and peak pressure test respectively.

e. Except for the initial ILRT, all ILRT's shall be performed without any preliminary leak detection surveys and leak repairs immediately prior to the test. If an ILRT has to be terminated due to excessive leakage through identified leakage paths, the leakage through such paths shall be determined by a local leakage test and recorded. After repairs are made another ILRT shall be conducted.

If an ILRT is completed but the acceptance criteria of Specification 4.7.A.2.d is not satisfied and repairs are necessary, the ILRT need not be

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A (Cont'd)	4.7.A.2.e (cont'd) repeated provided locally measured leakage reductions, achieved by repairs, reduce the containment's overall measured leakage rate sufficiently to meet the acceptance criteria. f. Local Leak Rate Tests 1. With the exceptions specified below, local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves at a pressure of 58 psig during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years. Table 3.7.2 specifies testable penetrations with double O-ring seals, Table 3.7.3 specifies testable penetrations with testable bellows, and Table 3.7.4 specifies primary containment testable isolation valves. The test duration of all valves and penetrations shall be of sufficient length to determine repeatable results. The total acceptable leakage for all valves and penetrations other than the MSIV's is 0.60 La. 2. Bolted double-gasket seals (Table 3.7.2) shall be tested after each opening and during each reactor shutdown for refueling, or other convenient intervals but in no case at intervals greater than two years. 3. The main steam isolation valves (MSIV's) shall be tested at a pressure of 29 psig. If a total leakage rate of 11.5 scf/hr for any one MSIV is exceeded, repairs and retest shall be performed to correct the condition. This is an exemption to Appendix J of 10CFR50.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A (Cont'd)	4.7.A.2.F (cont'd) 4. Main steam line and feedwater line expansion bellows as specified in Table 3.7.3 shall be tested by pressurizing between the laminations of the bellows at a pressure of 5 psig. This is an exemption to Appendix J of 10CFR50. 5. The personnel airlock shall be tested at 58 psig at intervals no longer than six months. This testing may be extended to the next refueling outage (not to exceed 24 months) provided that there have been no airlock openings since the last successful test at 58 psig. In the event the personnel airlock is not opened between refueling outages, it shall be leak checked at 3 psig at intervals no longer than six months. Within three days of opening (or every three days during periods of frequent opening) when containment integrity is required, test the personnel airlock at 3 psig. This is an exemption to Appendix J of 10CFR50. g. <u>Continuous Leak Rate Monitor</u> When the primary containment is inerted the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable. h. <u>Drywell Surfaces</u> The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of torus corrosion or leakage.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A (cont'd.)	4.7.A (cont'd.)
3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers	3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers
a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building air actuated vacuum breakers shall be 0.5 psid. The self actuated vacuum breakers shall open fully when subjected to a force equivalent to 0.5 psid acting on the valve disc.	a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentation, including set points shall be checked for proper operation every three months.
b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, the vacuum breaker switch shall be secured in the closed position and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.	b. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specifications 3.7.A.3.a and each vacuum breaker shall be inspected and verified to meet design requirements.
4. Drywell-Pressure Suppression Chamber Vacuum Breakers	4. Drywell-Pressure Suppression Chamber Vacuum Breakers
a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be operable at the 0.5 psid setpoint and positioned in the fully closed position as indicated by the position indicating system except during testing and except as specified in 3.7.A.4.b and .c below.	a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle every 30 days.
b. Three drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening provided they are secured in the fully closed position or that the requirement of 3.7.A.4.c is demonstrated to be met.	b. When it is determined that a vacuum breaker valve is inoperable for opening at a time when operability is required all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A.4 (cont'd.)	4.7.A.4 (cont'd.)
c. The total leakage between the drywell and suppression chamber shall be less than the equivalent leakage through a 1" diameter orifice.	c. Once each operating cycle, each vacuum breaker valve shall be visually inspected to insure proper maintenance and operation of the position indication switch. The differential pressure setpoint shall be verified.
d. If specifications 3.7.A.4.a, b or c, cannot be met, the situation shall be corrected within 24 hours or the reactor will be placed in a cold shutdown condition within the subsequent 24 hours.	d. Prior to reactor startup after each refueling, a leak test of the drywell to suppression chamber structure shall be conducted to demonstrate that the requirement of 3.7.A.4.c is met.
5. Oxygen Concentration	5. Oxygen Concentration
a. After completion of the startup test program and demonstration of plant electrical output, the primay containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.	a. The primary containment oxygen concentration shall be measured and recorded at least twice weekly.
b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown.	b. The quantity of liquid nitrogen in the liquid nitrogen storage tank shall be determined twice per week when the volume requirements of 3.7.A.5.c are in effect.
c. When the containment atmosphere oxygen concentration is required to be less than 4%, the minimum quantity of liquid nitrogen in the liquid nitrogen storage tank shall be 500 gallons.	
d. If the specifications of 3.7.A.5.a thru c cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.	
e. The specifications of 3.7.A.5.a thru d are not applicable during a 48 hour continuous period between the dates of March 22, 1982 and March 25, 1982.	

COOPER NUCLEAR STATION
TABLE 3.7.1 (Page 1)
PRIMARY CONTAINMENT ISOLATION VALVES

Valve & Steam	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (Sec) (1)	Normal Position (2)	Action On Initiating Signal (3)
Main Steam Isolation Valves					
MS-AO-80- A,B,C, & D	4		3 < T < 5	O	GC
MS-AO-86- A,B,C, & D		4	3 < T < 5	O	GC
Drywell Floor Drain Iso. Valves RW-AO-82, RW-AC-83	1	1	15	O	GC
Drywell Equipment Drain Iso. Valves RW-AO-94, RW-AO-95	1	1	15	O	GC
Main Steam Line Drain Valves MS-MO-74, MS-MO-77	1	1	30	C	SC
Reactor Water Sample Valves RRV-740AV, RRV-741AV	1	1	15	O	GC
Reactor Water Cleanup System Iso. Valves RWCU-MO-15, RWCU-MO-18	1	1	60	O	GC
RHR Reactor Head Spray Iso. Valves RHR-MO-32, RHR-MO-33	1	1	60	C	SC
RHR Suction Cooling Iso. Valve RHR-MO-17, RHR-MO-18	1	1	40	C	SC
RHR Discharge to Radwaste Iso. Valves RHR-MO-57, RHR-MO-67	1	1	20	C	SC
Suppression Chamber Purge & Vent PC-245AV, PC-230MV		2	15	C	SC
Suppression Chamber N_2 Supply PC-237AV, PC-233MV		2	15	C	SC

TABLE 3.7.4
PRIMARY CONTAINMENT TESTABLE ISOLATION VALVES

PEN. NO.	VALVE NUMBERS	TEST MEDIA
X-7A	MS-AO-80A and MS-AO-86A, Main Steam Isolation Valves	Air
X-7B	MS-AO-80B and MS-AO-86B, Main Steam Isolation Valves	Air
X-7C	MS-AO-80C and MS-AO-86C, Main Steam Isolation Valves	Air
X-7D	MS-AO-80D and MS-AO-86D, Main Steam Isolation valves	Air
X-8	MS-MO-74 and MS-MO-77, Main Steam Line Drain	Air
X-9A	RF-15CV and RF-16CV, Feedwater Check Valves	Air
X-9A	RCIC-AO-22, RCIC-MO-17, and RWCU-15CV, RCIC/RWCU Connection to Feedwater	Air
X-9B	RF-13CV and RF-14CV, Feedwater Check Valves	Air
X-9B	HPCI-AO-18 and HPCI-MO-57, HPCI Connection to Feedwater	Air
X-10	RCIC-MO-15 and RCIC-MO-16, RCIC Steam Line	Air
X-11	HPCI-MO-15 and HPCI-MO-16, RPCI Steam Line	Air
X-12	RHR-MO-17 and RHR-MO-18, RHR Suction Cooling	Air
X-13A	RHR-MO-25A and RHR-MO-27A, RHR Supply to RPV	Air
X-13B	RHR-MO-25B and RHR-MO-27B, RHR Supply to RPV	Air
X-14	RWCU-MO-15 and RWCU-MO-18, Inlet to RWCU System	Air
X-16A	CS-MO-11A and CS-MO-12A, Core Spray to RPV	Air
X-16B	CS-MO-11B and CS-MO-12B, Core Spray to RPV	Air
X-17	RHR-MO-32 and RHR-MO-33, RPV Head Spray	Air
X-18	RW-732AV and RW-733AV, Drywell Equipment Sump Discharge	Air
X-19	RW-765AV and RW-766AV, Drywell Floor Drain Sump Discharge	Air
X-25	PC-232MV and PC-238AV, Purge and Vent Supply to Drywell	Air
X-25	ACAD-1305MV and ACAD-1306MV, Supply to Drywell	Air
X-26	PC-231MV and PC-246AV, Purge and Vent Exhaust [illegible] Drywell	Air
X-26	ACAD-1310MV, Bleed from Drywell	Air

3.7.A & 4.7.A BASES

Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination, limit the off-site doses to values less than those suggested in 10CFR100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10CFR100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 58 psig which is below the maximum of 62 psig. Maximum water volume of 91,000 ft^3 results in a downcomer submergence of 5' and the minimum volume of 87,650 ft^3 results in a submergence approximately 12 inches less. The majority of the Bodega tests were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humbolt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to drain the suppression chamber, this should only

be done when there is no requirement for core standby cooling systems operability as explained in bases 3.5.F.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Inerting

Safety Guide 7 assumptions for Metal-Water reaction result in hydrogen concentration in excess of the Safety Guide 7 flammability limit. By keeping the oxygen concentration less than 4% by volume the requirements of Safety Guide 7 are satisfied.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during periods when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance.

The 500 gallon conservative limit on the nitrogen storage tank assures that adequate time is available to get the tank refilled assuming normal plant operation. The estimated maximum makeup rate is 1500 SCFD which would require about 160 gallons for a 10 day makeup requirement. The normal leak rate should be about 200 SCFD.

Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the

results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a greater assurance that the valve will be operable when needed.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment. A program for periodic testing and examination of the excess flow check valves is performed as follows:

1. Vessel at pressure sufficient to actuate valves. This could be at time of vessel hydro following a refueling outage.

2. Isolate sensing line from its instrument at the instrument manifold.

3. Provide means for observing and collecting the instrument drain or vent valve flow.

4. Open vent or drain valve.

   a. Observe flow cessation and any leakage rate.

   b. Reset valve after test completion.

5. The head seal leak detection line cannot be tested in this manner. This valve will not be exposed to primary system pressure except under unlikely conditions of seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source and therefore this valve need not be tested. This valve is in a sensing line that is not safety related.

6. Valves will be accepted if a marked decrease in flow rate is observed and the leakage rate is acceptable.

3.7.E Bases

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed as described in the licensee's letter of October 4, 1976, which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of drywell-suppression chamber differential pressure of 1.0 psid and a suppression chamber water level corresponding to a downcomer submergence range of three to four feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.