UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 173
License No. DPR-46

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nebraska Public Power District (the licensee) dated June 28, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-46 is hereby amended to read as follows:

    2. Technical Specifications

    The Technical Specifications contained in Appendix A, as revised through Amendment No. 173, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Hall

James R. Hall, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: November 8, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 173

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

REMOVE PAGES	INSERT PAGES
i	i
6	6
12	12
30	30
77	77
107	107
108	108
109	109
110	110
111	111
112	112
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RADIOLOGICAL TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

1.1 FUEL CLADDING INTEGRITY

Applicability

The SAFETY LIMITS established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective

The objective of the SAFETY LIMITS is to establish limits below which the integrity of the fuel cladding is preserved.

Action

If a SAFETY LIMIT is exceeded, the reactor shall be in at least HOT SHUTDOWN within 2 hours.

Specifications

A. Reactor Pressure ≥800 psia and Core Flow ≥10% of Rated

The existence of a MINIMUM CRITICAL POWER RATIO (MCPR) less than 1.06 for two recirculation loop operation (1.07 for single-loop operation) shall constitute violation of the fuel cladding integrity SAFETY LIMIT.

B. Core Thermal Power Limit (Reactor Pressure <800 psia and/or Core Flow (10%)

When the reactor pressure is (800 psia or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of RATED POWER.

C. Power Transient

To ensure that the SAFETY LIMIT established in Specification 1.1.A and 1.1.B is not exceeded, each required scram shall be initiated by its expected scram signal. The SAFETY LIMIT shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

2.1 FUEL CLADDING INTEGRITY

Applicability

The LIMITING SAFETY SYSTEM SETTINGS apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity SAFETY LIMITS from being exceeded.

Objective

The objective of the LIMITING SAFETY SYSTEM SETTINGS is to define the level of the process variables at which automatic PROTECTIVE ACTION is initiated to prevent the fuel cladding integrity SAFETY LIMITS from being exceeded.

Specifications

A. Trip Settings

The LIMITING SAFETY SYSTEM SETTINGS shall be as specified below:

1. Neutron Flux Trip Settings

a. APRM Flux Scram Trip Setting (Run Mode)

(1) Flow Referenced Scram Trip Setting

When the Reactor Mode Selector Switch is in the RUN position, the APRM flow referenced flux scram trip setting shall be:

$S \leq 0.58\ W + 62\% - .58\ \Delta W$

where:

S = Setting in percent of RATED POWER (2381 MWt).

W = Two-loop recirculation flow rate in percent of rated (rated loop recirculation flow rate is that recirculation flow rate which provides 100% core flow at 100% power).

ΔW = Difference between two-loop and single-loop effective drive flow at the same core flow.

1.1 Bases: (Cont'd)

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the SAFETY LIMIT of Specification 1.1A or 1.1B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a SAFETY LIMIT violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a SAFETY LIMIT provided scram signals are OPERABLE is supported by the extensive plant safety analysis.

The computer provided with Cooper Nuclear Station has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C will be relied on to determine if a SAFETY LIMIT has been violated.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is SHUTDOWN, consideration must also be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the SAFETY LIMIT at 18 inches above the top of the fuel provides adequate margin.

References for 1.1 Bases

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-(latest approved revision).

2. "Cooper Nuclear Station Single-Loop Operation," NEDO-24258, May, 1980.

NOTES FOR TABLE 3.1.1

1. There shall be two OPERABLE or tripped TRIP SYSTEMS for each function. If the minimum number of OPERABLE INSTRUMENT CHANNELS for a TRIP SYSTEM cannot be met, the affected TRIP SYSTEM shall be placed in the safe (tripped) condition, or the appropriate actions listed below shall be taken.

   A. Initiate insertion of OPERABLE rods and complete insertion of all OPERABLE rods within four hours.

   B. Reduce power to less than 30% of RATED POWER.

   C. Reduce power level to IRM range and place Reactor Mode Selector Switch in the STARTUP position within eight hours and depressurize to less than 1000 psig.

   D. Reduce turbine load and close Main Steam Isolation Valves within eight hours.

2. Permissible to bypass, with control rod block, for Reactor Protection System reset in REFUEL and SHUTDOWN positions of the Reactor Mode Selector Switch.

3. This note deleted.

4. Permissible to bypass when turbine first stage pressure is less than 30% of full load.

5. IRMs are bypassed when APRMs are onscale and the Reactor Mode Selector Switch is in the RUN position.

6. The design permits closure of any two lines without a full scram being initiated.

7. When the reactor is subcritical, fuel is in the vessel, and the reactor water temperature is less than 212°F, only the following trip functions need to be OPERABLE:

   A. Reactor Mode Selector Switch in SHUTDOWN.
   B. Manual scram.
   C. IRM high fluxa at 120/125 indicated scale.
   D. APRM (15%) high flux scram.

8. Not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

9. Not required to be OPERABLE while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

TABLE 4.2.C
SURVEILLANCE REQUIREMENTS FOR ROD WITHDRAWAL BLOCK INSTRUMENTATION

Function	Functional Test Freq.	Calibration Freq.	Instrument Check
APRM Upscale (Flow Bias)	(1) (3)	Once/3 Months	Once/Day
APRM Upscale (Startup Mode)	(1) (3)	Once/3 Months	Once/Day
APRM Downscale	(1) (3)	Once/3 Months	Once/Day
APRM Inoperative	(1) (3)	N.A.	Once/Day
RBM Upscale (Power Referenced)	(1) (3)	Once/6 Months	Once/Day
RBM Power Range	(3)	Once/6 Months	N.A.
RBM Downscale	(1) (3)	Once/6 Months	Once/Day
RBM Inoperative	(1) (3)	N.A.	Once/Day
IRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
IRM Detector Not Full In	(2) (Once/OPERATING CYCLE)	Once/OPERATING CYCLE (10)	Once/Day
IRM Inoperative	(1) (2) (3)	N.A.	N.A.
SRM Upscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Downscale	(1) (2) (3)	Once/3 Months	Once/Day
SRM Detector Not Full In	(2) (Once/OPERATING CYCLE)	Once/OPERATING CYCLE (10)	N.A.
SRM Inoperative	(1) (2) (3)	N.A.	N.A.
Flow Bias Comparator	(1) (8)	Once/OPERATING CYCLE	N.A.
Flow Bias Upscale	(1) (8)	Once/3 Months	N.A.
Rod Block Logic	(9)	N.A.	N.A.
SDV High Water Level	Quarterly	Once/OPERATING CYCLE	N.A.

3.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the operating status of the Standby Liquid Control (SLC) System.

Objective:

To assure the OPERABILITY of a system with the capability to SHUTDOWN the reactor and maintain the SHUTDOWN condition without the use of control rods.

Specification:

A. Normal System Operation

1. During periods when fuel is in the reactor and prior to startup from a Cold Condition, the Standby Liquid Control System shall be operable, except as specified in 3.4.B below. This system need not be operable when the reactor is in the Cold Condition and all control rods are fully inserted and Specification 3.3.A is met.

4.4 STANDBY LIQUID CONTROL SYSTEM

Applicability:

Applies to the surveillance requirements of the Standby Liquid Control (SLC) System.

Objective:

To verify the OPERABILITY of the SLC System.

Specification:

A. Normal System Operation

The OPERABILITY of the SLC System shall be shown by the performance of the following tests:

1. At least once each 3 months each subsystem shall be tested for OPERABILITY by recirculating demineralized water to the test tank and verifying each pump develops a flow rate $\geq$ 38.2 gpm at a discharge pressure $\geq$ 1300 psig.

2. At least once during each OPERATING CYCLE:

a. Check that the settings of the subsystem relief valves are 1450 < P < 1680 psig and the valves will reset at P $\geq$ 1300 psig.

b. Manually initiate the system, except explosive valves, and pump boron solution from the SLC Storage Tank through the recirculation path. After pumping boron solution the system will be flushed with demineralized water.

3.4

4.4.A.2 (Cont'd.)

c. Manually initiate one of the SLC System Pumps and pump demineralized water into the reactor vessel from the test tank.

These tests check the actuation of the explosive charge of the tested subsystem, proper operation of the valves, and pump OPERABILITY. The replacement charges to be installed will be selected from the same manufactured batch as a previously tested charge.

d. Both subsystems, including both explosive valves, shall be tested in the course of two OPERATING CYCLES.

B. Operation with Inoperable Components:

1. From and after the date that one subsystem is made or found to be inoperable, Specification 3.4.A.1 shall be considered fulfilled and continued REACTOR POWER OPERATION permitted provided that the OPERABLE subsystem remains OPERABLE and the inoperable subsystem is returned to an OPERABLE condition within seven days.

B. Surveillance with Inoperable Components:

1. When a subsystem is found to be inoperable, the OPERABLE subsystem shall be verified to be OPERABLE immediately and daily thereafter until the inoperable subsystem is returned to an OPERABLE condition.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.4

C. Sodium Pentaborate Solution

At all times when the SLC System is required to be OPERABLE the following conditions shall be met:

1. The net volume versus concentration of the liquid control solution in the SLC Storage Tank shall be maintained as required in Figure 3.4.1.

2. The temperature of the liquid control solution shall be maintained above the curve shown in Figure 3.4.2.

D. If specification 3.4.A through C cannot be met, the reactor shall be placed in a Cold Shutdown Condition with all operable control rods fully inserted within 24 hours.

4.4.C

C. Sodium Pentaborate Solution

The following tests shall be performed to verify the availability of the Liquid Control Solution:

1. Volume: Check and record at least once per day.

2. Temperature: Check and record at least once per day.

3. Concentration: Check and record at least once per month. Also check concentration anytime water or boron is added to the solution or solution temperature is below the temperature required in Figure 3.4.2.

3.4 BASES

STANDBY LIQUID CONTROL SYSTEM

A. The Standby Liquid Control (SLC) System consists of two, distinct subsystems, each containing one positive displacement pump and independent suction from the SLC storage tank, and discharge to a common injection header through parallel explosive valves. The purpose of the SLC System is to provide the capability of bringing the reactor from RATED POWER to a cold, xenon-free SHUTDOWN CONDITION assuming that none of the withdrawn control rods can be inserted. To meet this objective, the system is designed to inject a quantity of boron that produces a concentration of 660 ppm of boron in the reactor pressure vessel in less than 125 minutes. The 660 ppm concentration in the reactor pressure vessel is required to bring the reactor from RATED POWER to a 3.0 percent Δk subcritical condition, considering the hot to cold reactivity difference, xenon poisoning, etc. The time requirement for inserting the boron solution was selected to override the rate of reactivity insertion caused by cooldown of the reactor following the xenon poison peak.

The conditions under which the SLC System must provide shutdown capability are identified in Limiting Conditions for Operation. If no more than one OPERABLE control rod is withdrawn, the basic shutdown reactivity requirement for the core is satisfied and the SLC System is not required. Thus, the basic reactivity requirement for the core is the primary determinant of when the SLC System is required.

The minimum limitation on the relief valve setting is intended to prevent the recycling of liquid control solution via the lifting of a relief valve at too low a pressure. The upper limit on the relief valve setting provides system protection from overpressure.

B. Only one of the two SLC subsystems is needed for operating the system. One inoperable subsystem does not immediately threaten shutdown capability, and reactor operation can continue while the inoperable subsystem is being repaired. The seven day completion time is based on the availability of an OPERABLE subsystem capable of performing the intended SLC system function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive (CRD) system to shut down the plant.

BASES (cont'd.)

C. Level indication and alarm indicate whether the solution volume has changed, which might indicate a solution concentration change. The test interval has been established in consideration of these factors. Temperature and liquid level alarms for the system are annunciated in the control room.

The solution is kept at least 10°F above the saturation temperature to guard against boron precipitation. The margin is included in Figure 3.4.2.

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

The quantity of stored boron includes an additional margin (25 percent) beyond the amount needed to shutdown the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water. A minimum quantity of 3132 gallons of solution having a 16.0 weight percent sodium pentaborate concentration, or the equivalent as shown in Figure 3.4.1, is required to meet this shutdown requirement.

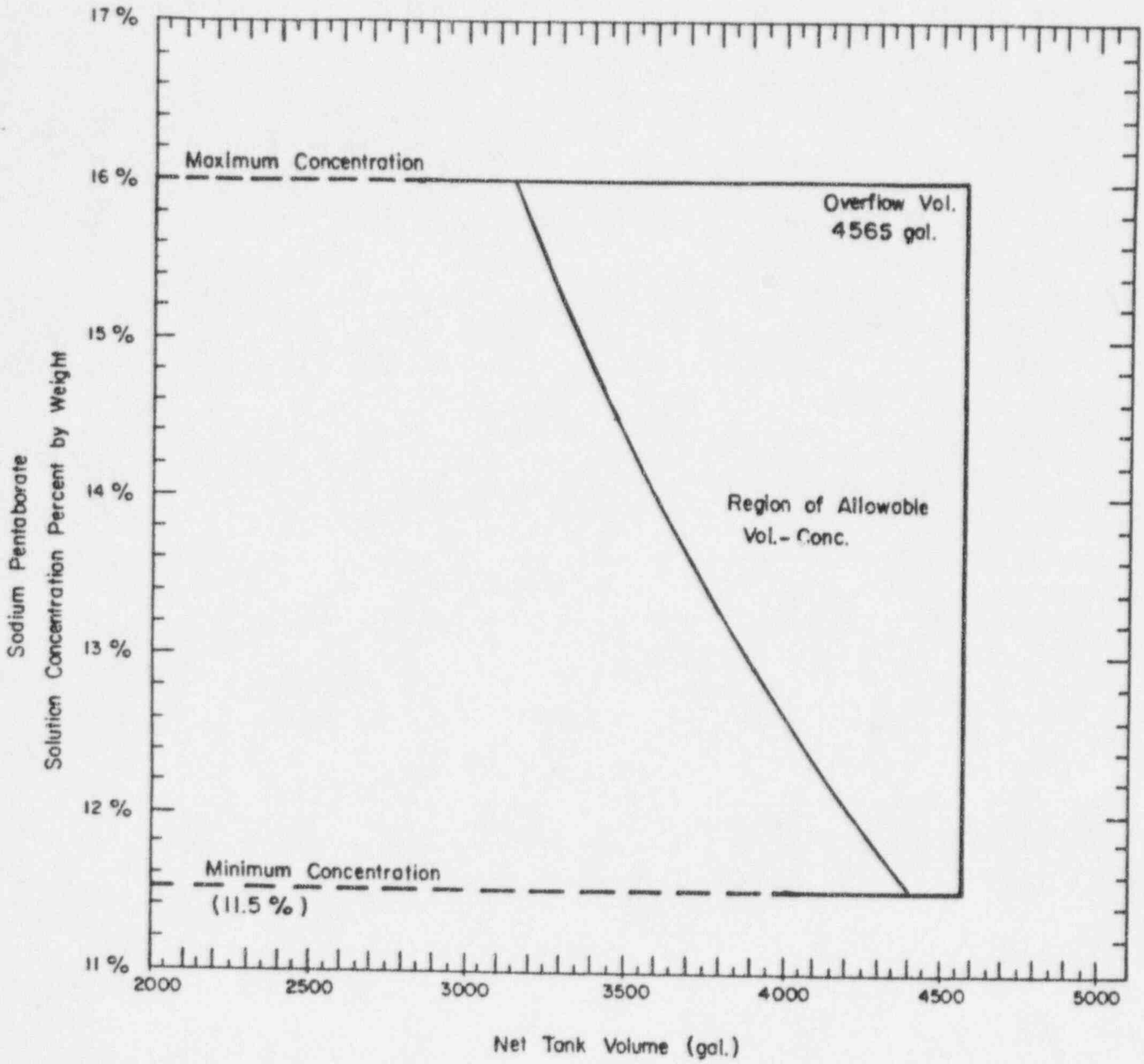
The NRC's final rule on Anticipated Transients Without Scram (ATWS), 10CFR50.62, requires that the SLC System be modified to provide a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent sodium pentaborate solution for a 251 inch I.D. vessel. For Cooper Nuclear Station, with a 218 inch I.D. vessel, the equivalent minimum flow rate is 66 gpm of 13 weight percent sodium pentaborate. This equivalence is met with both SLC pumps supplying their minimum flow rate of 38.2 gpm with a solution concentration of at least 11.5 weight percent of sodium pentaborate. Because ATWS is a very low probability event and is considered to be beyond the design basis for CNS, the surveillance and limiting condition for operation requirements need not be more stringent than the original SLC System design basis requirements. The SLC System changes made as a result of the ATWS rule do not invalidate the original system design basis.

4.4 BASES

STANDBY LIQUID CONTROL SYSTEM

Experience with pump OPERABILITY indicates that testing once each three months, in combination with the tests during each OPERATING CYCLE, is sufficient to maintain pump performance. The only practical time to fully test the SLC system is during a REFUELING OUTAGE. Various components of the system are individually tested periodically, thus making unnecessary more frequent testing of the entire system.

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



NEBRASKA PUBLIC POWER DISTRICT
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Sodium Pentaborate
Solution Volume-Concentration
Requirements

FIGURE 3.4.1

BASES:

3.6.H and 4.6.H

Snubbers

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during STARTUP and SHUTDOWN. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be OPERABLE during REACTOR POWER PERATION.

Because the snubber protection is required only during relatively low probability events, a period of 72 hours is allowed for repairs or replacement. Since plant STARTUP should not commence with knowingly defective safety related equipment, Specification 3.6.H.3 prohibits STARTUP with inoperable snubbers.

All safety related snubbers are visually inspected for overall integrity and OPERABILITY.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions, such as temperature, radiation and vibration.