Attachment 3

SAFETY REVIEW REPORT
FOR
NUCLEAR INSTRUMENTATION SYSTEM (NIS)

San Onofre Nuclear Generating Station Unit 1

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Prepared by

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1.0 INTRODUCTION

Southern California Edison has contracted with Westinghouse to supply equipment for upgrading to the Nuclear Instrumentation System (NIS) at the San Onofre Generating Station Unit 1 (SONGS 1). The upgraded system is functionally identical to the currently-installed NIS, with the exception of the intermediate range channel. Incorporated into the intermediate range channel is the capability for post accident monitoring (PAM), a function which makes the SONGS 1 design unique from other Westinghouse NIS designs. Other individual features of the SONGS 1 design include an asymmetric power range detector configuration and two-section fission chambers for the intermediate and power range detectors. These features will be described in more detail later. The basic design, however, is very similar to that supplied by Westinghouse to a number of other nuclear plants. Installation of the NIS is to be accomplished by Southern California Edison Company (SCE).

This document, a Safety Review Report (SRR), describes the new system design and evaluates how the licensing basis safety analyses are affected by the changes, as well as how the NIS upgrade for the Westinghouse scope of supply conforms to appropriate industry and regulatory standards. For this upgrade, information is presented herein to demonstrate that the mechanical and electrical portions of the NIS are (or will be) environmentally qualified and are capable of performing their designated safety-related functions while exposed to applicable normal, abnormal, test, accident, and post-accident environment conditions. Seismic qualification of safety-related mechanical and electrical equipment is also presented for the NIS.

Changes to the SONGS 1 Technical Specifications required as a result of the upgrade are described in section 5.0 of this report, and the marked-up pages are presented in Appendix A.

2.0 SYSTEM DESCRIPTION

2.1 General Description

The upgraded NIS provides output signals which are an integral part of the Westinghouse Nuclear Steam Supply System (NSSS) for SONGS 1, a pressurized water reactor plant. The function of this NIS is to measure the neutron flux from the source level to 200% of full reactor power. The NIS provides, in the control room, flux level information in the source, intermediate, and power ranges to monitor the power level of the plant at all times. It provides signals for tripping the reactor, inhibiting control rod withdrawal and alarming off-normal operating conditions, and provides signals to the Reactor Control and Protection System for automatic plant operations. In addition, the NIS provides post-accident monitoring (PAM) of neutron flux from 10^{-7} to 200% of full reactor power. A block diagram of the Nuclear Instrumentation System is shown in Figure 2-1. Functional requirements for the NIS are presented in Appendix D.

The NIS being installed at SONGS 1 consists of four nuclear instrumentation channels. Two channels consist of power range only and two channels consist of power, intermediate, and source ranges. Both control and protective functions are provided by the NIS signals. Control functions are provided by source range and intermediate range high start up rate rod stop signals. In addition, the power range provides input signals to the Rod Control System for rod speed control. Protective functions are provided by intermediate range high start up rate reactor trip signals and power range low, mid and high overpower reactor trip signals. In addition to the control and protective functions. Indications for auxiliary functions are derived from the source, intermediate, and power range channels. These auxiliary functions include audio-visual indication of count rate, average power comparison, start-up rate and flux deviation. Outputs from the upgraded NIS are divided into three overlapping ranges consisting of the source, intermediate, and power ranges. The three overlapping ranges of instrumentation provide signals for reactor protection at increasing levels during startup and at power operation. The

overlap of instrument ranges provides reliable, continuous protection throughout the range of neutron flux. Additional detail describing the functions of the various channels is provided in the following sections.

At SONGS 1, there are two different detector thimble sizes. Two of the eight thimbles are 5.00 inches in diameter, and the remaining six thimbles are 6.999 inches in diameter. In the current SONGS 1 NIS, one of the 5.00-inch thimbles is being used for a power range detector; however, current detector designs (as in the NIS upgrade for SONGS 1) dictate the use of a thimble larger than 5.00 inches, since the detectors are 6.75 inches in diameter. This larger detector size thus requires that the detector used in the NIS upgrade be relocated from Thimble 4 (a 5.00-inch thimble) to Thimble 3 (a 6.999-inch thimble). Figure 2-2 shows the detector locations for the upgrade.

The relocation of one power range excore detector from Thimble 4 to Thimble 3 alters the radial separation between each of the power range detectors from 90 degrees in the current NIS. In the design of the NIS upgrade, the detector in Thimble 3 is 45 degrees from the adjacent detector in one direction and 135 degrees from the adjacent detector in the opposite direction. This relocation potentially changes what the detectors will sense, given an asymmetric power distribution and thus the resulting value for asymmetry. Although there is no impact on the conclusion of the safety analyses (see section 3.0), there is a technical specification change associated with the asymmetry limit. This is discussed in section 5.2.1 and reflected on the mark-up of Technical Specification 2.1 provided in Appendix A.

2.2 Instrumentation Channels

2.2.1 Source Range Channels (Ref: W Dwg. 2D33790)

The source range consists of two independent channels, each utilizing a boron triflouride proportional counter located in separate thimble locations outside of the reactor vessel. The neutron flux produces pulses in the detectors, proportional in number to the neutron flux present at the detector location.

These pulses are fed through a series of triaxial cables to a preamplifier, and then to the four bay NIS console. The source range preamplifier increases the signal to noise ratio by amplifying the signal prior to transmission and is located outside of containment as close to the detector as possible to minimize signal loss. At shutdown the source range channels provide alarm signals in the control room and inside containment in the event there is an inadvertent increase in reactivity. The source range detector high voltage is turned off and on automatically by the intermediate range drawers. For details see Intermediate (Wide) Range Channels (section 2.2.2) below. The source range drawer high voltage may also be controlled manually in the event of an intermediate range drawer failure or for maintenance purposes. Included on the source range drawer is an indicator lamp which illuminates when the source range high voltage cutoff switch is out of the normal (automatic) position.

Reactor control is provided during start-up by supplying a high start up rate rod stop signal.

2.2.2 Intermediate (Wide) Range Channels (Ref: W Dwg. 2D33791)

The intermediate range consists of two independent channels, each utilizing the lower sections of a two-section fission chamber located in separate thimble locations outside the reactor vessel. The two-section fission chambers are utilized for both the intermediate range (lower section) and the power range (upper section) channels. The signal output of the fission chamber is proportional to the neutron flux present at the detector location. The low level neutron signal detected by the lower section of the two-section fission chamber is fed through a series of triaxial and quadaxial cables to the out-of-containment wide range pre-amplifier which amplifies these signals before coupling to the wide range amplifier in the four bay NIS console. The wide range pre-amplifier also conditions the signals from the lower sections of the two-section fission chambers for use in the power range channel.

The intermediate range channel also generates automatic internal signals (internal to the NIS) to turn the source range high voltage off or on. The source range high voltage is cut off automatically, upon increasing neutron flux, when the setpoint is reached. Each channel (#1 and #2) has its own setpoint adjustment for source range high voltage cutoff. When a setpoint is reached, intermediate range channel #1 or channel #2 will cut off the respective source range channel high voltage. Upon decreasing neutron flux and below the setpoint, the source range high voltage is automatically turned on. No operator action is required to turn the source range high voltage off or on except for intermediate range drawer failure or for test purposes. Each intermediate range drawer includes a source range high voltage cutoff indication lamp.

Reactor protection is provided by a high start up rate reactor trip signal. Reactor control is provided by a high start-up-rate rod stop signal.

The intermediate range channels also provide the post accident monitoring function for neutron flux monitoring. (The post accident hardware includes the lower sections of the two-section fission chambers, wide range preamplifiers, wide range amplifiers, and analog indicators.) The intermediate range channels exhibit a range from 10^{-7} to 200% power. Class 1E outputs from the intermediate range drawers are provided for the Class 1E control board analog indicators. The system arrangement meets the requirements of Regulatory Guide 1.97 Revision 2 requirements for Category 1 instrumentation.

In addition, the intermediate range drawers provide Class 1E outputs for the remote shutdown panel qualified analog indicators. The outputs provided meet the requirements of 10CFR Part 50, Criterion 19 for providing neutron flux indication necessary to maintain the plant in a safe shutdown condition from outside the control room.

2.2.3 Power Range Channels (Ref: W Dwgs 2D33793, 2D33794)

The power range consists of four independent channels, consisting of the two two-section fission chambers and two uncompensated ion chambers. Each of the

power range detectors is located in separate thimble locations outside of the reactor vessel. Signals from each detector section are fed through a series of triaxial cables to signal conditioning equipment in the four bay NIS console.

Reactor protection is provided by the following reactor trip signals: high range overpower reactor trip, mid range overpower reactor trip and low range overpower reactor trip. The power range channels provide a low, mid and high power rod stop output to develop the P-1 permissive. The power range channels also provide signals to develop permissives (P) P-1, P-3, P-7, and P-8 described in section 2.5.

2.2.4 Coincidenter Channels (Ref: W Dwg 2D33797 and 2D33798)

The coincidentor channels receive isolated input signals from the source, intermediate, and power range channels and provide logic arrangement of the signals to the Reactor Control and Protection System.

2.2.5 Auxiliary Channels (Ref: W Dwg 2D33795 and 2D33796)

These non-safety related channels receive isolated inputs from the source, intermediate and power range detectors. These channels provide the following: power range high flux deviation signal, rod speed control signal, axial offset signal, audio count rate signal, source and intermediate range startup rate and power range deviation signals.

2.3 Nuclear Overpower Trips

2.3.1 Power Range High Range Neutron Flux Trip (Ref: W Dwg 2D33797)

A selector switch on the control board provides a means to set the reactor trip for low power, mid-power or high power. Existing alarms (not part of this upgrade) warn the operator at preselected setpoints to change the switch mode when increasing or decreasing power level.

The power range neutron flux trip circuit, through the coincidenter, provides a reactor trip signal when two out of four power range high range neutron flux signals exceed the trip setpoint (the selector switch to be set for high range). Two of the power range signals originate from the upper and lower sections of the two-section fission chambers. The other two signals originate from the uncompensated ion chambers.

2.3.2 Power Range Mid Range Neutron Flux Trip (Ref: W Dwg 2D33797)

A selector switch on the control board provides a means to set the reactor trip for low power, mid-power or high power. Existing alarms (not part of this upgrade) warn the operator at preselected set points to change the switch mode when increasing or decreasing level.

The power range neutron flux trip circuit, through the coincidenter, provides a reactor trip signal when two out of four power range mid range neutron flux signals exceed the trip setpoint (the selector switch to be set for mid-range). Two of the power range signals originate from the upper and lower sections of the two-section fission chambers. The other two signals originate from the uncompensated ion chambers.

2.3.3 Power Range Low Range Neutron Flux Trip (Ref: W Dwg 2D33797)

A selector switch on the control board provides a means to set the reactor trip for low power, mid-power or high power.

The power range neutron flux trip circuit, through the coincidenter, provides a reactor trip signal when two out of four power range low range neutron flux signals exceed the trip setpoint (the selector switch to be set for low range). Two of the power range signals originate from the upper and lower sections of the two-section fission chambers. The other two signals originate from the uncompensated ion chambers.

2.3.4 Intermediate Range High Start-Up-Rate Trip (Ref: W Dwg 2D33798)

The intermediate range high-start-up rate circuit, through the coincidenter, provides a reactor trip signal when one out of two intermediate channels exceeds the trip setpoint. Both of these nuclear signals originate from the lower sections of the two-section fission chambers and feed into the wide range pre-amplifier before being fed into the intermediate range drawer.

2.4 Nuclear Rod Stops

2.4.1 Power Range Overpower Rod Stop (Ref: W Dwg 2D33797)

The power range circuitry provides low, mid, and high power rod stop outputs to form an overpower rod stop signal in the coincidentor when one out of four power range channels exceeds the selected rod stop setpoint. A selector switch on the control board provides the capability for the operator to select one of the three (low, mid, and high) rod stop setpoints. Two of the power range signals originate from the upper and lower sections of the two-section fission chambers. The other two signals originate from the uncompensated ion chambers.

2.4.2 Power Range Dropped Rod Rod Stop (Ref: W Dwg 2D33797)

The power range circuitry provides a dropped rod rod stop signal when one out of four power range channels exceeds the setpoint. Two of the power range signals originate from the upper and lower sections of the two-section fission chambers. The other two signals originate from the uncompensated ion chambers.

2.4.3 Source/Intermediate Range High Start-Up-Rate Rod Stop
(Ref: W Dwg 2D33798)

The source/intermediate range circuitry provides a high start-up-rate rod stop signal on one out of two signals when either the source range high start-up-rate rod stop or intermediate range high start-up-rate rod stop has a coincidence of one out of two. This configuration provides a single rod stop

signal when either the source range or intermediate range high start-up-rate rod stop exceeds the setpoint. The source range signals originate from boron triflouride proportional counters and feed into source range preamplifiers. The intermediate range signal originates from the lower sections of the two-section fission chamber which feeds into the wide range amplifier.

2.5 Power Range Permissives

2.5.1 Permissive P-1 - Low, Mid and High Power Range Overpower Rod Stops (Ref: W Dwg 2D33797)

A manual selector switch selects power range overpower rod stop signals for low, mid or high power ranges. P-1 is activated upon receipt of a signal greater than a preset value from any one of four power range nuclear flux channels. P-1 blocks the "rods out" signal for all rods under manual or automatic control.

2.5.2 Permissive P-3 - Power Range Dropped Rod Rod Stop (Ref: W Dwg 2D33797)

P-3 is activated upon receipt of a rate of change signal from any one of four power range detectors. P-3 blocks the automatic "rods out" signal for the control group. P-3 is also actuated from the rod bottom device in the Rod Position Indicating System.

2.5.3 Permissive P-7 - Power Range Low Power (Ref: W Dwg 2D33798)

P-7 is activated upon receipt of signals from three out of four power range channels below 10 percent power level in coincidence with a turbine generator signal (MWe) below this value (from turbine first stage pressure). P-7 blocks reactor trip operation initiated by low flow, low pressure and turbine trips. In addition, P-7 unblocks the startup rate reactor trip and rod stops.

2.5.4 Permissive P-8 - Power Range Mid Power (Ref: W Dwg 2D33797)

P-8 is activated upon receipt of signals from two out of four power range channels above 50 percent power level in coincidence with a generator signal (MWe) above this value. In coincidence with P-8, a signal indicating loss of flow in any one loop causes a reactor trip. Below 50% power level, a coincidence of two low flow signals from two different loops is necessary to cause a reactor trip.

2.6 Bistable Outputs

The bistable output is 0 VAC for the required action to occur.

2.6.1 Source Range (Ref: W Dwg 2D33790)

High Flux at Shutdown; 0 VAC output to actuate horn.

High start-up-rate rod stop; 0 VAC output for rod stop.

Loss of detector voltage; 0 VAC output upon loss of S/R detector voltage.

2.6.2 Intermediate Range (Ref: W Dwg 2D33791)

Intermediate range high start-up-rate rod stop; 0 VAC output for rod stop.

Intermediate range high start-up-rate reactor trip; 0 VAC output for reactor trip signal.

Intermediate range loss of detector voltage; 0 VAC output upon loss of I/R detector voltage.

2.6.3 Power Range (Ref: W Dwg 2D33793 and 2D33794)

Overpower trip high range; 0 VAC output for reactor trip output signal.

Overpower trip mid range; 0 VAC output for reactor trip output signal.

Overpower trip low range; 0 VAC output for reactor trip output signal.

Dropped rod rod stop (P-3) (coincidenter logic for trains A & B); 0 VAC output for dropped rod rod stop.

Low power, mid power and high power overpower rod stop (P-1) (coincidenter logic for trains A & B); 0 VAC output for rod stop.

Low power permissive (P-7) (coincidenter logic for trains A & B); 0 VAC output for P-7.

Mid power permissive (P-8) (coincidenter logic for trains A & B); 0 VAC output for mid power permissive.

2.7 Bypasses/Manual Blocks/Channel on Test

2.7.1 Source Range (Ref: W Dwg 2D33790)

Source range #1 high voltage cutoff;
Switch S104 in source range for high voltage: Normal, On, or Off

High start-up-rate rod stop bypass - A keyed bypass switch permits blocking of high startup-rate rod stop during testing; 118 VAC output when bypassed.

Shutdown level manual block at high flux level - A manual block switch permits block of S/R high shutdown level annunciator and plant horn; 118 VAC output when switch is activated.

Channel on Test - Test annunciator is energized when test switch is not in normal position. Normally closed contacts are utilized (switch in normal position).

2.7.2 Intermediate Range (Ref: W Dwg 2D33791)

High startup-rate reactor trip and high startup-rate rod stop bypass - A keyed bypass switch permits blocking of intermediate range high startup-rate reactor trip and high startup-rate rod stop; 118 VAC output when the switch is in bypass.

Channel on Test - Test annunciator is energized when test switch is not in normal position. Normally closed contacts are utilized (switch in normal position).

2.7.3 Power Range (Ref: W Dwg 2D33793 and 2D33794)

Rod Stop Bypass - A keyed bypass switch permits blocking of the low power, mid power and high power rod stops and the dropped rod rod stop; 118 VAC output when bypassed.

Channel on Test - Test annunciator is energized when test switch is not in normal position. Normally closed contacts are utilized (switch in normal position).

2.8 Isolated Outputs Supplied by the NIS

2.8.1 Source Range (Ref: W Dwg 2D33790)

The following source range isolated output signals can be used for remote neutron flux level indications.

Audio count rate - 0 to 10 VDC internal NIS signal to remote audio count rate selector switch

Control Room NIS recorder - 0 to 10 MVDC (1 to 10^6 counts per second [cps])

Control Room meter - 0 to 1 MADC (1 to 10^6 cps)

Computer - 0 to 5 VDC

High startup-rate - 0 to 10 VDC internal NIS signal to non-safety related startup-rate meter in auxiliary channel.

2.8.2 Intermediate Range (Ref: W Dwg 2D33791)

The following intermediate range isolated output signals can be used for remote level indication from 0.1 to 10^6 counts per second and 10^{-7} to 200% of full reactor power.

Shutdown panel - Two channels of 0 to 1 MADC, wide range power 10^{-7} to 200% output is provided. Two channels of 0.1 to 10^6 cps is provided.

Control Room NIS recorder - 0 to 10 MVDC (10^{-7} to 200% power)

Control Room meter - 0 to 1 MADC (10^{-7} to 200% power)

Startup-rate - Internal NIS signal to non-safety related auxiliary channel

2.8.3 Power Range (Ref: W Dwg 2D33793 and 2D33794)

The following power range isolated output signals can be used for remote neutron level indication.

Upper flux comparator - 0 to 10 VDC internal NIS signal to non-safety related auxiliary channel

Upper axial offset - 0 to 5 VDC internal NIS signal to non-safety related auxiliary channel

Control room NIS recorder - 0 to 10 MVDC

Control room meter - 0 to 1MADC (0 to 120% of full reactor power)

Computer - 0 to 5 VDC

Rod speed control - 0 to 10 VDC internal NIS signal to non-safety related auxiliary channel

Comparator - 0 to 5 VDC internal NIS signal to non-safety related auxiliary channel

Overpower recorder - 0 to 10 MVDC (0 to 200% of full reactor power)

Lower flux comparator - 0 to 10 VDC internal NIS signal to non-safety related auxiliary channel

Lower axial offset - 0 to 5 VDC internal NIS signal to axial offset panel in the non-safety related auxiliary channel

Upper axial offset - 0 to 5 VDC internal NIS signal to axial offset panel in the non-safety related auxiliary channel

2.9 Non-Safety Related Auxiliary Channel

Bistable Outputs Flux Deviation (Ref: W Dwg 2D33795)

Power range upper detector high flux deviation - Spare

Power range lower detector high flux deviation - Spare

Power range channel deviation - 0 VAC output with bistable tripped; 118 VAC output with bistable not tripped

Meters and Alarms (Ref: W Dwgs 2D33795 and 2D33796)

Source range startup-rate meters (2) - (-0.1 to 0.9 MADC) (-0.5 to 5.0 DPM)

Intermediate range startup-rate meters (2) - (-0.1 to 0.9 MADA) (-0.5 to 5.0 DPM)

Percent axial offset - (-20 to 20%) (-1 to +1 VDC)

Audio count rate (control room and containment)

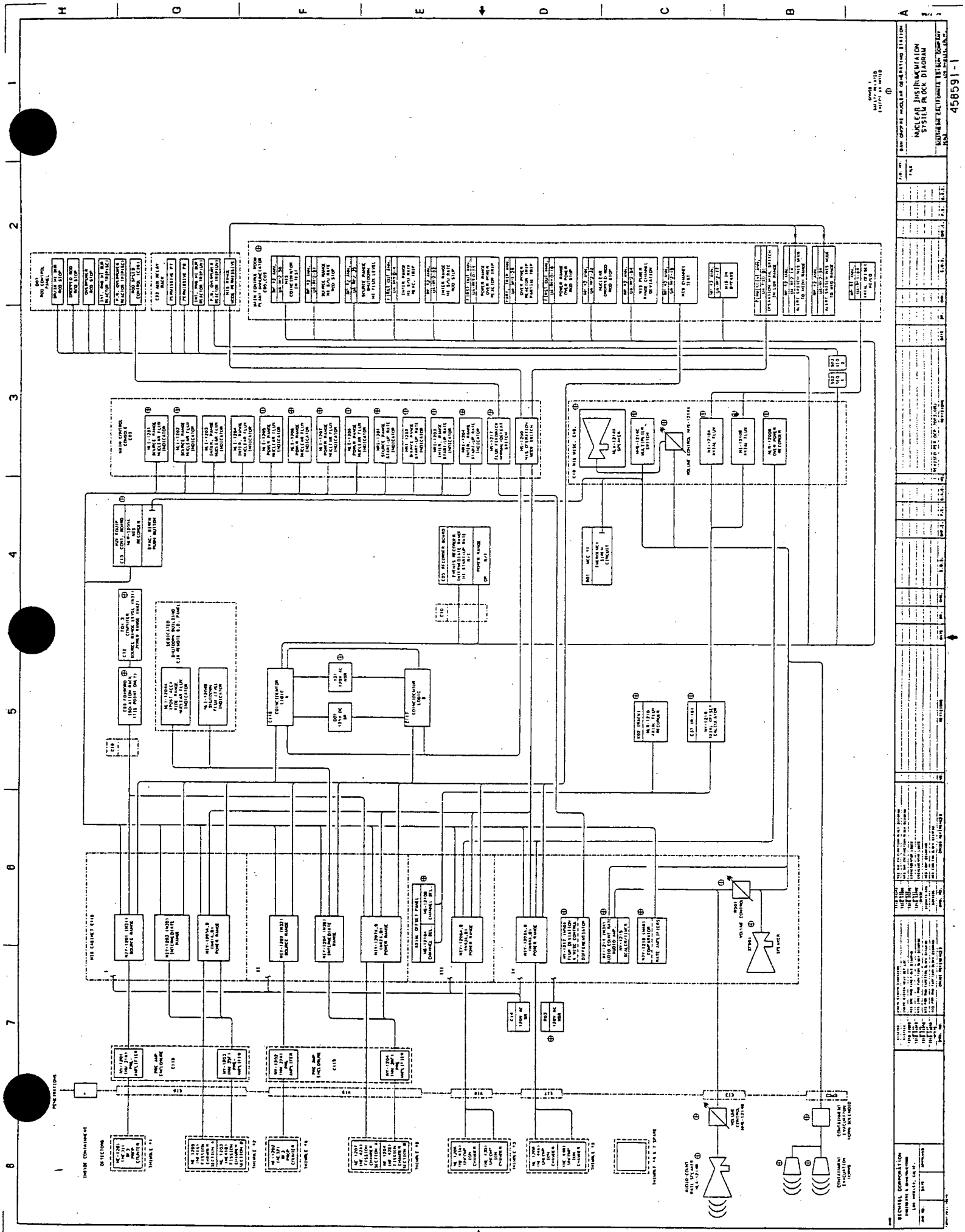
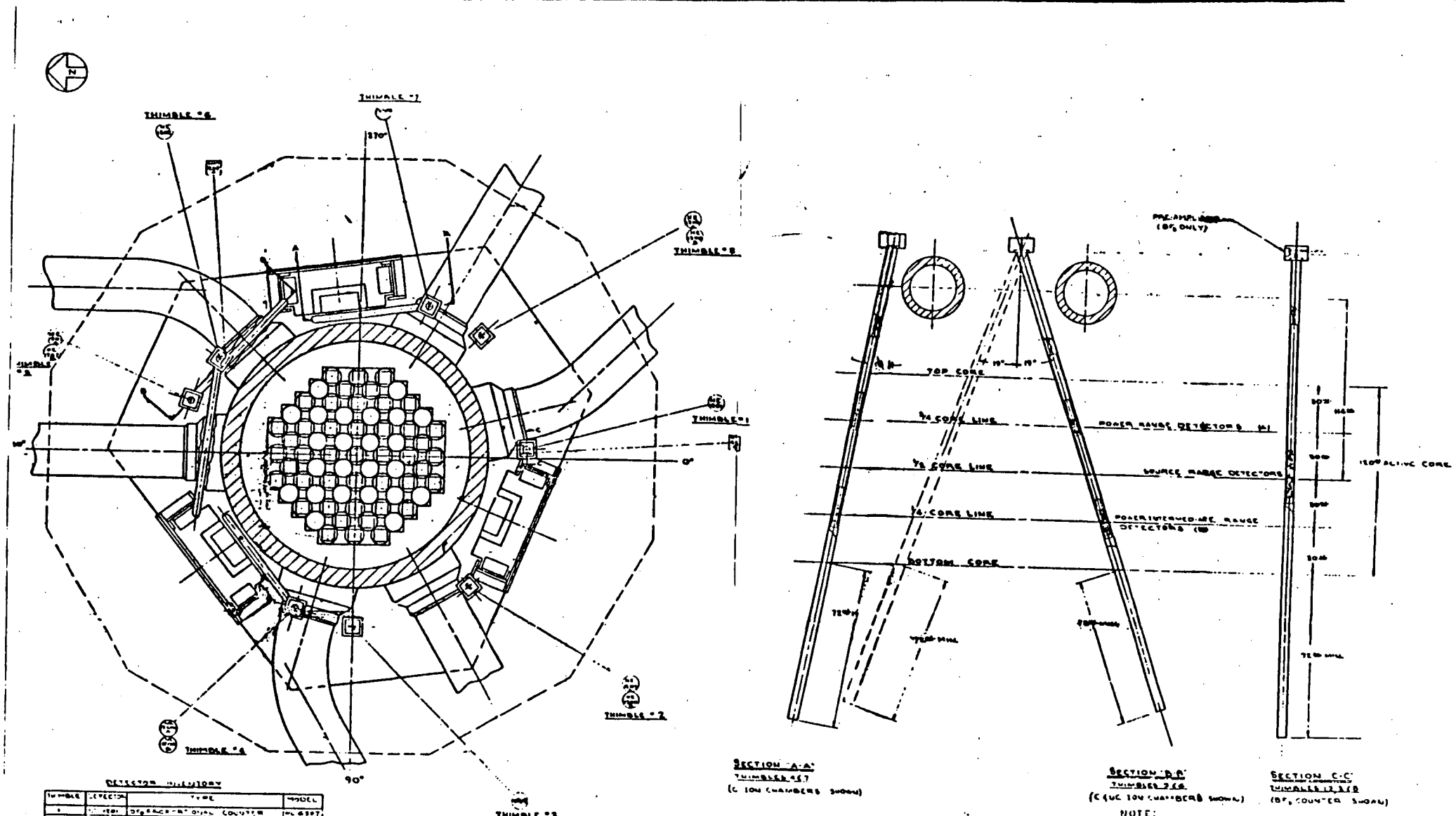


Figure 2-1



DETECTOR INVENTORY

THIMBLE	DETECTOR	TYPE	MODEL
1	[illegible]	BF3 PROPORTIONAL COUNTER	[illegible]
2	[illegible]	UNCOMPENSATED ION CHAMBER	[illegible]
3	[illegible]	COMPENSATED ION CHAMBER	[illegible]
4	[illegible]	FUTURE	—
5	[illegible]	UNCOMPENSATED ION CHAMBER	[illegible]
6	[illegible]	UNCOMPENSATED ION CHAMBER	[illegible]
7	[illegible]	COMPENSATED ION CHAMBER	[illegible]
8	[illegible]	BF3 PROPORTIONAL COUNTER	[illegible]
	—	FUTURE	—
	[illegible]	UNCOMPENSATED ION CHAMBER	[illegible]
	[illegible]	UNCOMPENSATED ION CHAMBER	[illegible]

Figure 2-2
Detector Location

3.0 ACCIDENT ANALYSIS

3.1 Introduction

The NIS upgrade to be accomplished at SONGS 1 has been previously described in section 2.0 of this document. Section 3.0 describes the evaluations that have been completed for the impact of the upgrade on the licensing basis safety analyses for non-Loss of Coolant Accidents (non-LOCAs), LOCAs, and steam generator tube rupture (SGTR) events analyzed for SONGS 1 by Westinghouse.

3.2 Non-LOCA Evaluation

3.2.1 Basis for Evaluation of the NIS Upgrade

The Nuclear Instrumentation System upgrade has been evaluated with respect to the SONGS 1 non-LOCA safety analyses. The relocation of the power range detector from Thimble 4 to Thimble 3 is the most noteworthy change to the NIS from the non-LOCA safety analysis perspective. In addition, the NIS upgrade includes adding a keyed switch to bypass the power range dropped rod and overpower rod stops when the power range channel is out of service. The non-LOCA evaluation of this feature is provided in section 3.2.5.

The primary use of the NIS in the non-LOCA safety analyses is to provide input to the Reactor Control and Protection System. In the upgraded NIS design, this input is used to generate rod stop signals on high startup rate in the source and intermediate ranges, and on negative flux rate and on overpower for the power range settings. Reactor trips are generated on high startup rate in the intermediate range, and on high flux in the power range (low, mid, and high settings). The NIS also supplies the rod control system with core power input for automatic rod control.

For the reactor trip settings of the NIS upgrade, the nominal setpoints which will be programmed will be unchanged. Also, the time responses of the upgraded NIS to generate trip signals (see Appendix D) remain consistent with the description of the current NIS provided in FSA Section 5.1. The non-LOCA

safety analyses take credit for only the reactor trip signal and the dropped rod rod stop signal provided by the power range channels. The time responses of these trip signals remain unchanged. For all non-LOCA transients except a single dropped rod cluster control assembly (RCCA), moving one of the power range detectors from Thimble 4 to Thimble 3 does not affect the ability of the NIS to measure core power or changes in core power. To generate reactor power indication, the total current output of each power range channel, both sections, is correlated with reactor power as determined by secondary side calorimetric power measurements. From this, a calibration on each channel is performed to allow it to indicate power in percent of Rated Thermal Power units. Relocation radially around the core and the tilt of these channels will only affect the magnitude of the calibration coefficients and not the validity of the calibration. This is because the calibration is made against total core power and not quadrant core power. The only concern is the ability of the power range detectors, due to the asymmetric configuration, to detect the small localized negative flux rate produced by a single dropped RCCA.

The following presents discussions of the impact of the NIS upgrade for each SONGS 1 non-LOCA transient. The list of SONGS 1 non-LOCA transients includes the incidents presented in the FSA (reference 3-1) and other non-LOCA safety analyses as noted in the following discussions. In all cases, except the Dropped RCCA transient, only an evaluation was required to support the NIS upgrade. The evaluation of the NIS upgrade for the Dropped RCCA transient required supporting reanalysis. Table 3-1 presents a summary of the impact of the NIS upgrade on the SONGS 1 non-LOCA analyses.

3.2.2 Non-LOCA Incidents Evaluated

3.2.2.1 Uncontrolled Rod Withdrawal from Subcritical (FSA 7.1)

This event is analyzed to show that the core and reactor coolant system are not adversely affected. This is done by showing the DNBR limit is met, and that the peak RCS pressure does not exceed 110% of design pressure.

An RCCA bank withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of one or more RCCA banks, resulting in a power excursion. Such a transient could be caused by a malfunction of the reactor control rod drive systems or due to operator error. This could occur with the reactor either subcritical, at hot zero power, or at power. The uncontrolled rod (RCCA) withdrawal from subcritical analysis is performed under hot zero power conditions with all three reactor coolant pumps operating. The maximum reactivity insertion rate is conservatively computed by assuming simultaneous withdrawal of two consecutive RCCA banks of maximum combined worth at the maximum speed.

The high reactivity worth postulated by the simultaneous withdrawal of two RCCA banks will cause a large reactivity addition to the system. The neutron flux response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative Doppler coefficient. After the initial power burst, the nuclear power is momentarily reduced and then if the bank withdrawal is not terminated, the nuclear power will increase again, but at a much slower rate. Protective action consists of a trip on nuclear flux, terminating the transient before fuel damage can occur.

Reactor protection and control is available from the following NIS actuations:

1. source range high startup rate rod stop
2. intermediate range high startup rate rod stop
3. intermediate range high startup reactor trip
4. power range high neutron flux -- low setting
5. power range high neutron flux -- mid setting
6. power range high neutron flux -- high setting

For this event, the analysis (the current rod withdrawal from subcritical analysis is discussed in reference 3-2) conservatively assumes reactor trip to be initiated at 118% power by the power range high neutron flux (high setting). Conservatively, no credit is taken for rod stop, the high startup rate reactor trip, or the power range high neutron flux (low setting) reactor

trip in the analysis although these reactor protection NIS functions would be available. The rod stop signal would be generated by high startup rate signal from either the source or intermediate range channels. The rod stop generated from these channels are unaffected by the NIS upgrade. Also, the functionality of the intermediate range high startup reactor trip is not affected by the NIS upgrade. As stated previously, the ability of the NIS power range detectors to measure core power is not affected by the movement of the power range detector located in Thimble 4 to Thimble 3. Thus, reactor protection would continue to be available by the power range high neutron flux reactor trip signals (high or low setting). Therefore, the current safety analysis remains valid, and the conclusions provided in the FSA remain valid.

3.2.2.2 Rod Withdrawal At Power (FSA 7.2)

This event is analyzed to show that the core and reactor coolant system are not adversely affected. This is done by showing that the DNBR limit is met and that the peak pressure does not exceed 110% of design pressure.

A continuous uncontrolled RCCA bank withdrawal at power due to operator action or a malfunction of the reactor instruments will result in an increase in the core heat flux. Immediately after the uncontrolled RCCA withdrawal accident, steam generator heat removal rates will lag behind the core power generation rate until the steam generator pressure reaches the relief or safety valve setpoint. This unbalanced heat removal rate will cause the reactor coolant temperature to rise. Unless this event is terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in departure from nucleate boiling (DNB) and/or fuel centerline melt. In order to avert core damage, the Reactor Protection System is designed to automatically terminate any such transient before the DNB ratio (DNBR) falls below the limit value.

Depending on the initial power level and rate of reactivity insertion, protection against an uncontrolled RCCA withdrawal at power is available from any of the following actuations:

1. power range high neutron flux rod stop
2. power range high neutron flux reactor trip
3. variable low pressure reactor trip
4. high pressurizer pressure reactor trip
5. high pressurizer water level reactor trip

The safety analysis (the current rod withdrawal at power safety analysis is discussed in reference 3) for this event assumes that core protection is provided by the variable low pressure reactor trip and the power range high neutron flux (high setting -- 118% power) reactor trip. Conservatively, no credit is taken in the safety analysis for rod stop actuation nor high pressurizer pressure reactor trip. As stated previously, the ability of the NIS power range detectors to measure core average power is not affected by the movement of the power range detector located in Thimble 4 to Thimble 3. Thus, there is no change to the core protection available from the NIS actuations (power range high neutron flux signals -- rod stop or reactor trip) for an uncontrolled RCCA bank withdrawal at power. Therefore, the current safety analysis remains valid, and the conclusions of the FSA remain valid.

3.2.2.3 Boron Dilution (FSA 7.3)

This transient is analyzed to demonstrate that there is sufficient time for operator action to terminate the dilution prior to a loss of shutdown margin.

Operator action or malfunction of the Chemical and Volume Control System (CVCS) may cause unborated water to be added to the Reactor Coolant System through the Makeup Water Supply System. This addition of makeup water will dilute the boron in the RCS, leading to an uncontrolled reactivity addition. The reactivity addition may lead to loss of shutdown margin and inadvertent criticality if the reactor is subcritical. If the reactor is operating at power, dilution of boron may cause an increase in power level and/or loss of capability to shut the reactor down via a trip of the control rods.

There are no automatic protection systems which will stop the addition of unborated water. Termination of the accident depends on operator action. The

following indications may alert the operator to the occurrence of a malfunction:

1. high source range neutron flux,
2. lo insertion limit alarm,
3. lo-lo insertion limit alarm,
4. variable low pressure reactor trip,
5. CVCS flow deviation alarm.

The performance of the source range neutron flux detectors is not affected by the NIS upgrade. All other indicators, which may alert the operator to the occurrence of a malfunction, are not impacted by the NIS upgrade. Therefore, the conclusions of the FSA remain valid.

3.2.2.4 Startup of an Inactive Loop (FSA 7.4.2)

This transient is analyzed to determine the effect on core integrity. Core integrity is assured by calculating a minimum DNBR above the limit value. Startup of an inactive loop causes the core average temperature to decrease due to the addition of the colder water from the inactive loop. This causes a increase in reactivity due to the increase in moderator density.

This incident need not be addressed due to the Technical Specification restrictions which prohibit operation with a loop out of service for power levels greater than 10 percent. However, a brief discussion of the impact of the NIS upgrade on the FSA analysis is included. The analysis in FSA Section 7.4.2 (reference 3-1) shows that the plant is tripped on power range high neutron flux, mid-setting (100% power). This trip setpoint would be conservative for operation less than 10 percent power since the low setting of the power range high neutron flux, with a nominal value of 25 percent, is activated below 10 percent power. The nuclear power and heat flux transients shown in FSA Section 7.4.2 are conservative. As stated previously, the movement of the power range detector located in Thimble 4 to Thimble 3 does not change the functionality for core power indication.

Thus, the NIS upgrade does not affect the ability of the power range high neutron flux detectors to measure core power, and the conclusions presented in the FSA remain valid.

3.2.2.5 Addition of Excess Feedwater (FSA 7.4.3)

This event is analyzed to show that the core and reactor coolant system are not adversely affected. This is done by showing that the DNBR limit is met and that the peak pressure does not exceed 110% of design pressure.

The addition of excessive feedwater is an excessive heat removal incident which results in a power increase due to moderator feedback. FSA Section 7.4.3 (reference 3-1) presents two cases. The first case assumes that all three feedwater control valves fully open together at full load. The second case assumes the startup of a feedwater pump with one pump already running while at 50 percent power; the control valves are in manual. The results presented in the FSA show that there is no significant reduction in core inlet temperature and there is very little response to this condition by the Reactor Coolant System for both cases. Manual trip of the reactor is assumed to occur following high level alarms on the steam generators. The core never approaches the DNBR safety limit, and the RCS pressure never exceeds 110% of design during this transient.

If no operator action is assumed, core protection is provided by the overpower and variable low pressure reactor trips to prevent violation of the acceptance criteria. The movement of the power range detector located in Thimble 4 to the new location (Thimble 3) does not affect the ability of the power range detectors to monitor core power during an excessive feedwater event or to provide a reactor trip on overpower if necessary. Thus, the conclusions presented in the FSA remain valid.

3.2.2.6 Large Load Increase (FSA 7.4.4)

This event is analyzed to show that the core and reactor coolant system are not adversely affected. This is done by showing that the DNBR limit is met and that the peak pressure does not exceed 110% of design pressure.

An excessive load increase event, in which the steam load exceeds the core power, results in a decrease in reactor coolant system temperature which may lead to a power increase due to moderator feedback. The maximum thermal power level evaluated in the FSA corresponds to the situation with the turbine control valves fully open. This eliminates the possibility of a large load increase above this power level. Therefore, as was shown in FSA Section 7.4.4 (reference 3-1), a step load increase of 30 percent to the maximum achievable steam flow, is not expected to result in a reactor trip since sufficient margin to the reactor protection setpoints (including uncertainties) exists. If required, protection for this event is provided by the overpower and variable low pressure reactor trip functions. As stated previously, the ability of the power range high neutron flux detectors to measure core power is not affected by the movement of the power neutron flux detector from Thimble 4 to Thimble 3. Thus, the conclusions of the FSA remain valid.

3.2.2.7 Control Rod Ejection Accident (FSA 7.6)

This accident is analyzed to ensure that the average fuel pellet enthalpy remains below the limit, that the hot spot clad temperature remains below 2450°F, that the peak reactor coolant pressure is less than a value which would cause stresses to exceed the Faulted Condition stress limit, and that fuel melting is limited to less than 10% at the hot spot.

This incident is the result of the assumed mechanical failure of a control rod mechanism pressure housing such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. The consequence of this mechanical failure is a large, rapid reactivity insertion together with an adverse core power distribution. The current Control Rod Ejection safety analysis is discussed in reference 3-3. Four cases are presented in the safety analysis: beginning-of-life hot full power (HFP), beginning-of-life hot zero power (HZP), end-of-life HFP, and end-of-life HZP. Core protection is provided rapidly by the power range high neutron flux, high setting for the HFP cases, or the power range high neutron flux, low setting for the HZP cases. The safety analysis assumes that the power range high neutron flux reactor trip low setting (35% with uncertainties) is available

for the HZP cases and the power range high neutron flux reactor trip high setting (118% with uncertainties) is available for the HFP cases. Although the RCCA ejection occurs in one core location, it results in a large reactivity addition. Consequently, the core average power excursion is very large (greater than a 40% change in power for all four cases analyzed). Thus, the ability of the NIS to measure core average power and provide core protection during a control rod ejection is not adversely affected by the movement of the power range detector located in Thimble 4 to Thimble 3. Therefore, the current safety analysis is not affected by the NIS upgrade, and the conclusions provided in the FSA remain valid.

3.2.2.8 Loss of Coolant Flow (FSA 8.2)

This accident is analyzed to show that the DNB design basis is met.

The current Loss of Coolant Flow safety analysis is discussed in reference 3-3. Reactor trip is assumed to occur due to low flow. The reactor will trip on the low flow, under voltage, or under frequency reactor trips. No credit is taken in the accident analysis for any protection or control signal generated by the NIS. The current safety analysis is, therefore, unaffected by the NIS upgrade, and the conclusions provided in the FSA remain valid.

3.2.2.9 Steam Line Break Core Response (FSA 8.4)

The inadvertent opening of a steam generator relief or safety valve case (Credible Break) is analyzed to show that the DNB design basis is met. The steam system piping failure cases (Hypothetical Break) are analyzed to show that the core remains intact and in place and that radiation doses do not exceed the limit.

A hypothetical steamline break results in a rapid depressurization of the steam generators which causes a large reactivity insertion to the core via primary cooldown. The acceptance criterion for this incident is that no DNB must occur following a return to power. This limit, however, is highly conservative since a hypothetical steamline break is classified as a Condition

IV event. The credible steamline break, a Condition II event, results in a much slower depressurization of the steam generators and, hence, a slower reactivity insertion. The acceptance criterion for this incident is that the minimum DNBR must remain above the limit value throughout the transient.

For the cases presented in FSA Section 8.4 (reference 3-1) and the most recent analysis (reference 3-3), the reactor is assumed to be at hot zero power and no actuation by the NIS overpower protection is required. These are considered the limiting cases for the steamline break analysis. As stated previously, the movement of the power range detector located in Thimble 4 to Thimble 3 does not affect the ability of the NIS to measure core power and to provide core protection for this event. Thus, the current safety analysis remains valid, and the conclusions provided in the FSA remain valid.

3.2.2.10 Steam Line Break Mass / Energy Release Outside Containment

Mass/energy releases following a steamline rupture are used to determine the temperature profiles for qualification of equipment. The temperature profile is a function of both the steam blowdown and the compartment in which the equipment is located. This analysis (reference 3-5) provides information for use in evaluating the effects of steam generator tube bundle uncovery and the associated superheated steam generation for areas outside containment.

For some of the cases (full power, large break sizes) presented in reference 3-5, credit was taken for reactor trip on power range high neutron flux, high setting (118% power). The movement of the power range detector located in Thimble 4 to Thimble 3 will not affect this reactor trip since the ability of the NIS to measure core power is unaffected. Thus, the analysis presented in reference 3-5 remains valid.

3.2.2.11 Steam Line Break Mass / Energy Release Inside Containment

Mass/energy releases following a steamline rupture inside containment are used to determine the maximum pressure peaks for containment integrity evaluations.

For all of the cases (full power, partial power, break sizes), no credit was taken for reactor trip on power range high neutron flux. The movement of the power range detector located in Thimble 4 to Thimble 3 does not impact this analysis since no credit was taken for NIS overpower protection. Thus, the analysis remains valid.

3.2.2.12 Feedline Break

This accident is analyzed to ensure that the reactor coolant and main steam pressures are maintained below 110% of their design pressures and the core remains in a coolable geometry. No control or protection signals generated by the NIS are assumed in the safety analysis so the upgrade has no effect. The conclusions presented in the current safety analysis (reference 3-6) remain valid.

3.2.2.13 Loss of Normal Feedwater (LONF) / LONF Coincident With Loss of Offsite Power at Reactor Trip

This event is analyzed to show that the core and reactor coolant system are not adversely affected. This is done by showing that the DNBR limit is met, that the peak pressure does not exceed 110% of design pressure, and that the pressurizer does not become water solid. No control or protection signals generated by the NIS are assumed in the safety analysis, so the NIS upgrade has no effect. The current safety analysis (reference 3-6) and the conclusions presented in the current safety analysis remain valid.

3.2.2.14 Loss of Load (FSA 8.6)

This event is analyzed to show that the core and reactor coolant system are not adversely affected. This is done by showing that the DNBR limit is met and that the peak pressure does not exceed 110% of design pressure.

The loss of load in combination with failure of the steam dump system cause an increase in steam generator temperature and pressure. This in turn causes an increase in RCS temperature and pressure. As described in FSA Section 8.6

(reference 3-1), core protection is provided by high pressurizer water level, high pressurizer pressure, or variable low pressure reactor trip. No control or protection signals generated by the NIS are assumed in the safety analysis so the upgrade has no effect. The conclusions presented in the FSA remain valid.

3.2.2.15 Locked Rotor / Pump Shaft Break

This accident is analyzed to ensure that the reactor coolant and main steam pressures are maintained below 110% of their design pressures and the core remains in a coolable geometry.

The Locked Rotor/Shaft Seizure safety analysis is discussed in reference 3-4. The instantaneous seizure of the RCP rotor or break of the shaft cause flow in the affected loop to decrease rapidly. Reactor trip is assumed to occur on low flow or RCP overcurrent. No credit is taken in the accident analysis for any protection or control signal generated by the NIS. The safety analysis is, therefore, unaffected by the NIS upgrade, and the conclusions remain valid.

3.2.3 Non-LOCA Incidents Reanalyzed

3.2.3.1 Dropped Rod (FSA 7.5)

This event is analyzed to show that the core and reactor coolant system are not adversely affected. This is done by showing that the DNBR limit is met and that the peak pressure does not exceed 110% of design pressure.

Dropping of a full-length RCCA occurs when the drive mechanism is deenergized. This would cause a power reduction and an increase in the hot channel factor. If no protective action were to occur, the Reactor Control System would restore the power to the level which existed before the incident. This would lead to a reduced safety margin or possibly DNB, depending upon the magnitude of the resulting hot channel factor.

If an RCCA drops into the core during power operation, it would be detected by either a rod bottom signal, by the power range nuclear instrumentation channels, or both. The rod bottom signal device provides an indication signal for each RCCA. The other indication of a dropped RCCA is obtained by using the power range channel signals. This rod drop detection circuit is actuated upon sensing a negative flux rate greater than the setpoint and is designed such that normal load variations do not cause it to be actuated.

A rod drop signal from any rod position indication channel or from one or more of the four neutron flux average difference signals is designed to initiate the following protective actions: reduction of the turbine load by a preset adjustable amount and blocking of further automatic rod withdrawal. The turbine runback is achieved by acting upon the turbine load limit. The power range rod drop signal must latch until manually reset.

For SCE SONGS 1 the analyses support operation on either nominal Tavg program (full power Tavg = 575.15 degrees F) with 20% steam generator tube plugging or a reduced Tavg program (full power Tavg = 551.5 degrees F) with 15% steam generator tube plugging. The dropped rod analysis presented in FSA Section 7.5 (reference 3-1) assumes the turbine runback portion of the protection system is available. This analysis presented in FSA Section 7.5 supports either of the operating programs listed above. A recent analysis (reference 3-7) supports the disabling of the turbine runback portion of the protection system for SONGS 1 operation on only the reduced Tavg program. Both analyses assumed that the rod withdrawal block is available and actuated.

Although the movement of the power range detector located in Thimble 4 to Thimble 3 does not affect the ability of power range detectors to measure core power, the movement may possibly affect the ability of the power range channels to detect the small localized negative reactivity insertion of a single RCCA drop. The worst single failure in the event of a dropped RCCA (single or multiple) is postulated as the failure of the rod bottom signal to generate the rod withdrawal block and turbine runback for operation on the nominal Tavg program and to generate the rod withdrawal block for operation on the reduced Tavg program. Although the rod bottom signal device provides an

indication signal for each RCCA, the protective action actuated by this signal is not redundant. Consequently, a single failure in the rod bottom signal detection system could prevent all rod bottom signals from actuating rod withdrawal block or turbine runback. However, multiple dropped RCCAs or dropped RCCA bank remains detectable even with the resultant asymmetry of the power range detectors due to the NIS upgrade. This is because multiple dropped RCCAs occur in different quadrants of the core. Thus, at least one NIS power range channel will detect the RCCA drop and actuate protection. Therefore, the analyses presented in FSA Section 7.5 and reference 7 remain applicable for multiple dropped RCCAs and dropped RCCA bank.

However, the asymmetry of the power range detectors produced by the NIS upgrade may hinder the detection of a single dropped RCCA, which is confined to a specific location in the core, by the power range channels. To be conservative, it is assumed that a single dropped RCCA will not be detected by one of the four neutron flux average difference channels to actuate protective action. The failure of the rod bottom signal to generate dropped RCCA protection is taken as the single failure. The dropped RCCA incident was reanalyzed to determine the impact of no protective actuation in the event of a single dropped RCCA for SONGS 1 Cycle 10 operation. As stated above, the protection consists of turbine runback and rod withdrawal block for SONGS 1 operation on the nominal Tavg program and only rod withdrawal block for SONGS 1 operation on the reduced Tavg program.

The reanalysis was performed to bound both operating programs for SONGS 1 during Cycle 10: nominal Tavg program with 20% steam generator tube plugging and reduced Tavg program with 15% steam generator tube plugging. The limiting initial condition to bound both operating programs correspond to the nominal Tavg program with 20% steam generator tube plugging and are listed below:

Power = 103% of 1351 MWt
Flow = 195000 gpm
Inlet Temperature = 554.8 degrees F
Pressurizer Pressure = 2070 psia

An analysis based upon the methodology presented in reference 8 and applied in a plant specific manner to SONGS 1 was performed to determine the impact of the single dropped RCCA with no protective action for operation either on the nominal Tavg program or on the reduced Tavg program for the upcoming Cycle 10 core design. The plant specific analysis showed that the accident (occurring during Cycle 10 operation) results in a minimum DNBR above the safety limit DNBR value of 1.30. Note that, consistent with current reload practice (reference 9), the single dropped rod scenario will be addressed as part of each future reload safety evaluation for SONGS 1.

Thus, the conclusions given in the FSA remain valid for the movement of the power range detector located in Thimble 4 to Thimble 3 for the NIS upgrade.

3.2.4 Conclusions

The NIS upgrade does not affect the ability of the Nuclear Instrumentation System to provide core protection during SONGS 1 non-LOCA transients. Specifically, the movement of the power range detector located in Thimble 4 to Thimble 3 does not adversely impact the ability of the power range channels to measure core power and to provide core protection for the non-LOCA transients. The conclusions presented in the FSA and the current safety analyses remain valid.

3.2.5 Evaluation of Overpower and Dropped Rod Rod Stop Bypasses

3.2.5.1 Introduction

A design change has been requested by SCE to the SONGS 1 replacement NIS which consists of adding a keyed switch to each power range channel to bypass the overpower and dropped rod rod stops when the power range channel is out of service. The following discussion provides an evaluation of the bypass feature with respect to the safety analysis.

Technical Specification 3.5 outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection

system when any one or more of the channels of the reactor trip instrumentation is out of service. Table 3.5-1 of Technical Specification 3.5 provides the total number of channels, channels to trip, minimum channels operable, applicable modes, and action statements for the reactor trip system instrumentation. Placing a power range channel out of service renders the channel inoperable, and three channels remain operable for the power range high neutron flux protection. The minimum channels operable per Table 3.5-1 is three. The proposed applicable action statement for the power range high neutron flux protection is that with the number of operable channels one less than the total number of channels, startup and/or power operation may proceed provided the inoperable channel is placed in the tripped condition within 1 hour.

The power range high neutron flux reactor trip actuation requires two channels to indicate power above the setpoint. Placing the out of service power range channel in the tripped condition still requires another channel to indicate power above the setpoint to generate a reactor trip. However, a dropped rod or an overpower rod stop is actuated when only one out of four power range channels reach the setpoint. Placing the out of service power range channel in the tripped condition actuates the dropped rod and overpower rod stop. The design change of adding a switch to bypass the dropped rod and overpower rod stop when the power range channel is out of service permits continued operation without the rod stop. The remaining three operable channels are available to generate a rod stop when one of the three power range channels reaches the setpoint.

The following evaluation addresses the impact of the design change on the non-LOCA safety analyses in two separate sections:

1. Bypass of the overpower (low, mid, and high settings) rod stops
2. Bypass of the dropped rod rod stop

3.2.5.2 Bypass of the Overpower Rod Stops

An overpower rod stop is actuated when one out of four power range channels indicates power above the overpower rod stop setpoints (low, mid, or high).

The SONGS 1 non-LOCA safety analyses do not take credit for the power range overpower rod stops. The bypass of the overpower rod stop when a power range channel is out of service does not impact the SONGS 1 non-LOCA safety analyses. Therefore, bypass of the overpower rod stop when a power range channel is out of service is acceptable, and the conclusion presented in the FSA and the current safety analyses remain valid.

3.2.5.3 Bypass of the Dropped Rod Rod Stop

A dropped rod rod stop is actuated when one out of four power range channels indicates a negative flux rate greater than the setpoint. The only SONGS 1 non-LOCA safety analysis which takes credit for the dropped rod rod stop is the Dropped Rod Incident analysis (references 3-1 and 3-7).

A rod drop signal from any rod position indication channel or from one or more of the four neutron flux average difference signals is designed to initiate the following protective actions: reduction of the turbine load by a preset adjustable amount and blocking of further automatic rod withdrawal. The turbine runback is achieved by acting upon the turbine load limit. The power range rod drop signal must latch until manually reset.

For SONGS 1 the analyses support operation on either nominal Tavg program (full power Tavg = 575.15 degrees F) with 20% steam generator tube plugging or a reduced Tavg program (full power Tavg = 551.5 degrees F) with 15% steam generator tube plugging. The dropped rod analysis presented in FSA Section 7.5 (reference 3-1) assumes the turbine runback portion of the protection system is available. This analysis presented in FSA Section 7.5 supports either of the operating programs listed above. A recent analysis (reference 3-7) supports the disabling of the turbine runback portion of the protection system for SONGS 1 operation on only the reduced Tavg program. Both analyses assumed that the rod withdrawal block is available and actuated.

Bypassing of a power range dropped rod rod stop when the power range channel is out of service results in three power range channels available to provide the dropped rod protection assumed in the safety analyses.

The evaluation for movement of the power range detector located in Thimble 4 to Thimble 3 presented an analysis which showed that no dropped rod protection was required for a single dropped RCCA for either Unit 1 operating program. Thus, adding a bypass for the dropped rod rod stop when the power range channel is out of service does not impact the safety analysis for a single dropped RCCA.

However, adding a bypass for the dropped rod rod stop when the power range channel is out of service does impact the safety analyses for multiple dropped RCCAs. Bypassing of a power range dropped rod rod stop when the power range channel is out of service results in three power range channels available to provide the dropped rod protection. Without supporting analysis, it is conservatively postulated that the combination of only three power range channels available (when one channel is inoperable) and the asymmetry of the detector locations could prevent the NIS from detecting multiple dropped RCCAs. Although rod bottom signal device provides an indication signal for each RCCA, the protective action actuated by this signal is not redundant. Consequently, a single failure in the rod bottom signal detection system could prevent all rod bottom signals from actuating rod withdrawal block or turbine runback. With a single failure in the rod bottom signal detection system and multiple dropped RCCAs not seen by the NIS, no protective action (turbine runback/rod stop) would occur.

For SONGS 1 operation on the reduced Tavg program, automatic rod withdrawal block is required to provide core protection in the event of multiple dropped RCCAs. The current analysis for the reduced Tavg program supports the disabling of the turbine runback. To support the dropped rod analysis for SONGS 1 operation on the reduced Tavg program, the reactor must be placed in a configuration such that automatic rod withdrawal is prevented before bypassing a power range channel dropped-rod rod stop when the power range channel is out of service. This can be achieved, for example, by placing the reactor in manual rod control or permanently engaging the rod stop. This requirement is reflected in the proposed Technical Specifications.

For SONGS 1 operation on the nominal Tavg program, automatic rod withdrawal block and turbine runback are required to provide core protection in the event of multiple dropped RCCAs. If the reactor is operating above the turbine load cutback power level, a dropped-rod signal runs back the turbine to a preset value and initiates rod withdrawal block. If the reactor is operating below the preset turbine cutback power level, a rod drop signal initiates rod withdrawal block but not turbine runback. The core remains protected, however, since the plant is operating at a power level below the cutback value. To support the dropped-rod analysis for SONGS 1 operation on the nominal Tavg program, the reactor must be placed in a configuration such that automatic rod withdrawal is prevented and the reactor must not operate above the turbine runback power level setpoint (before bypassing a power range channel dropped-rod rod stop) when the power range channel is out of service. Again, this is reflected in the proposed Technical Specification.

3.2.5.4 Conclusions

Adding a keyed switch to each power range channel to bypass the overpower rod stops when the power range channel is out of service to the NIS upgrade is acceptable with respect to the SONGS 1 non-LOCA safety analyses, since the analyses do not take credit for the overpower rod stops. However, adding a keyed switch to each power range channel to bypass the dropped rod rod stop when the power range channel is out of service is acceptable for this NIS upgrade provided certain conditions are met. For SONGS 1 operation on the reduced Tavg program, the reactor must be placed in a configuration such that automatic rod withdrawal is prevented before bypassing a power range channel dropped rod rod stop when the power range channel is out of service. For SONGS 1 operation on the nominal Tavg program, the reactor must be placed in a configuration such that automatic rod withdrawal is prevented and the reactor must not operate above the turbine runback power level setpoint (before bypassing a power range channel dropped rod rod stop) when the power range channel is out of service. These requirements can be reflected as an additional Technical Specification or as additional administrative procedures.

TABLE 3-1

LIST OF NON-LOCA ANALYSES

AND EFFECT OF NIS UPGRADE

Accident	Impact on Results
Uncontrolled Rod Withdrawal from Subcritical	Conclusions provided in FSA and current safety analysis remain valid.
Rod Withdrawal at Power	Conclusions provided in FSA and current safety analysis remain valid.
Boron Dilution	Conclusions provided in FSA remain valid.
Startup of an Inactive Loop	Conclusions provided in FSA remain valid.
Addition of Excess Feedwater	Conclusions provided in FSA remain valid.
Large Load Increase	Conclusions provided in FSA remain valid.
Control Rod Ejection	Conclusions provided in FSA and current safety analysis remain valid.
Loss of Coolant Flow	Conclusions provided in FSA and current safety analysis remain valid.
Steam Line Break Core Response	Conclusions provided in FSA and current safety analysis remain valid.
Steam Line Break Mass/Energy Release Outside Containment	No adverse effect on mass/energy releases.
Steam Line Break Mass/Energy Release Inside Containment	No adverse effect on mass/energy releases.
Feedline Break	Conclusions provided in current safety analysis remain valid.
Loss of Normal Feedwater (LONF)/ LONF coincident with Loss of Offsite Power	Conclusions provided in current safety analysis remain valid.
Loss of Load	Conclusions provided in FSA remain valid.

TABLE 3-1 (cont'd)
LIST OF NON-LOCA ANALYSES
AND EFFECT OF NIS UPGRADE

Accident	Impact on Results
Locked Rotor/Pump Shaft Break	Conclusions provided in current safety analysis remain valid.
Dropped Rod	Asymmetry of power range detectors required reanalysis of single dropped rod. Acceptable results were shown. Conclusion provided in FSA and current safety analyses remain valid. Bypass of a power range channel dropped rod rod stop when a power range channel is out of service required operating restrictions, which are included in the proposed Technical Specifications.

3.3 Loss of Coolant Accident (LOCA) and Steam Generator Tube Rupture (SGTR) Evaluations

3.3.1 Large Break LOCA

The description of the various aspects of the large break LOCA analysis methodology is given in reference 3-10. This document describes the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which ensure compliance with the Interim Acceptance Criteria. The codes are used to assess the core heat transfer geometry and to determine if the core remains amenable to cooling throughout and subsequent to the blowdown, refill, and reflood phases of the large break LOCA.

A review of these codes used for the large break LOCA analysis for SONGS 1 has shown that the upgrade to the NIS, the moving of the power range detector, and the use of a multiple power range rod stop are not explicitly modeled in any of the codes. An examination of the important parameters that would impact the above mentioned transients also demonstrated that the parameters would not be affected by the SCE changes. Therefore, the aforementioned changes would not directly impact the results of the large break LOCA analysis.

The results of the analysis could indirectly be impacted by a change to any input related to allowable power distribution or fuel rod initial conditions necessitated by the SONGS 1 hardware upgrade. However, the Westinghouse Nuclear Fuel Division has confirmed that the NIS upgrade along with changes to the power range detector and the use of a multiple power range or dropped rod rod stops will not affect the allowable power distribution or fuel rod initial conditions.

Based on the above discussions, the changes by SCE will have no significant impact on the large break LOCA analysis results. Thus, the changes will have no significant impact on the SONGS 1 large break margin to the PCT limit of 2300 degrees F.

3.3.2 Small Break LOCA

The Westinghouse small break LOCA analysis consists of a thermal hydraulic Reactor Coolant System (RCS) and a hot rod analysis. For SONGS 1 this analysis was performed with the WFLASH and LOCTA computer codes respectively.

A review of these codes used for the small break LOCA analysis for SONGS 1 has shown that the upgrade to the NIS, the moving of the power range detector, and the use of a multiple power range rod stop are not explicitly modeled in any of the codes. An examination of the important parameters that would impact the above mentioned transients also demonstrated that the parameters would not be affected by the SCE changes. Therefore, the aforementioned changes would not directly impact the results of the small break LOCA analysis.

The results of the analysis could indirectly be impacted by a change to any input related to allowable power distribution or fuel rod initial conditions necessitated by the SONGS 1 hardware upgrade. However, the Westinghouse Nuclear Fuel Division has confirmed that the NIS upgrade along with changes to the power range detector and the use of a multiple power range or dropped rod rod stops will not affect the allowable power distribution or fuel rod initial conditions.

Based on the above discussions, the changes by SCE will have no significant impact on the small break LOCA analysis results. Thus, the changes will have no significant impact on the SONGS 1 small break margin to the PCT limit of 2300 degrees F.

3.3.3 Containment Integrity

The NIS is not a factor in performing Containment Integrity Analyses, and is not credited in performing a function to mitigate the consequences of the accident. Hence, an upgrade to the NIS hardware for SONGS 1 would not have an effect on the containment integrity analyses.

3.3.4 Hot Leg Switchover to Prevent Potential Boron Precipitation

Post-LOCA hot leg recirculation switchover time is determined for inclusion in emergency procedures to ensure no boron precipitation in the reactor vessel. This time is dependent on power level, and the RCS and RWST water volumes and boron concentrations. Since the upgraded NIS, the moving of the power range detector, and the use of a multiple power range or dropped rod rod stops will not affect the power level or the maximum boron concentrations or volumes assumed for the RCS and RWST, there is no impact on the post-LOCA hot leg switchover time.

3.3.5 Rod Ejection Mass and Energy Release

To analyze a rod ejection accident, a small break LOCA is performed with a break in the upper head the size of a control rod drive shaft in order to determine the primary coolant mass released to the containment through the break and the steam released from the steam generator safety valves. This information is then used to compute the radiological consequences of a rod ejection accident. Therefore, the discussion of the effects of the NIS upgrade, the moving of the power range detector, and the use of a multiple power rod stop for the small break LOCA is applicable to the rod ejection accident. It can be concluded then that there will be no effect on the rod ejection mass and energy releases due to the NIS upgrade.

3.3.6 Blowdown Reactor Vessel and Loop Forcing Functions

The effects of a postulated LOCA on the integrity of the reactor vessel internals and reactor coolant loops are not part of the design basis analysis in the SONGS 1 FSA.

3.3.7 Post-LOCA Long-Term Core Cooling and Subcriticality Requirements

Reference 3-11 presents the Westinghouse licensing commitment to keep the reactor core subcritical on the boron provided by the ECCS with all control rods out

(ARO) following a hypothetical large break LOCA. Reference 3-12 recently provided clarification to the utilities on the Westinghouse licensing position for Westinghouse designed PWR's. This licensing commitment is verified by Westinghouse each cycle using design calculations which are independent of the detailed design of the NIS and will not be adversely affected by the NIS upgrade.

3.3.8 Steam Generator Tube Rupture

The SGTR accident is analyzed to assess the magnitude of radiological releases to the atmosphere and subsequently the offsite dose expected from such an accident. The upgrade to the SONGS 1 NIS will not affect the RCS and secondary operating parameters nor should it change the fuel failure assumption for the SGTR analysis. Since these important parameters which control the thermal-hydraulic phenomena and subsequently the radiological consequences of a SGTR event are not impacted, the conclusions of the SONGS 1 SGTR analysis remain valid.

3.3.9 Conclusions

It may be concluded that the assumptions and results of the safety analyses for the loss of coolant and the steam generator tube rupture accidents remain valid for SONGS 1, as a result of the NIS upgrade described in section 2.0 of this document.

Table 3-2 contains the list of analyses within the Westinghouse LOCA/SGTR scope and the effect on the results due to the NIS upgrade.

Table 3-2
LIST OF LOCA/SGTR ANALYSES
AND EFFECT OF NIS UPGRADE

Accident	Impact on Results
Large Break LOCA	No adverse effect on the FSAR PCT calculations. Compliance with the Interim Acceptance Criteria maintained.
Small Break LOCA	No adverse effect on the FSAR PCT calculations. Compliance with the Interim Acceptance Criteria maintained.
Rod Ejection Accident	No adverse effect on mass releases.
Containment Integrity	No adverse effect on peak containment pressure or temperature.
Steam Generator Tube	No adverse effect on mass releases Rupture or offsite radiation doses.
Blowdown Reactor Vessel and Loop Forces	No adverse effect on the LOCA hydraulic forcing functions.
Post-LOCA Longterm Core Cooling	No adverse effect on results.
Hot Leg Switchover to	No adverse effect on the Prevent Potential Boron post-LOCA hot leg switchover Precipitation time.

3.4 References

3-1. Docket Number 50-206, "San Onofre Nuclear Generating Station, Unit 1, Part 2, Final Safety Analysis."

3-2. Skaritka, J., Editor, "Reload Safety Evaluation - San Onofre Unit 1, Cycle 7," August 1978.

3-3. Skaritka, J., Editor, "Reload Safety Evaluation - San Onofre Nuclear Generating Station Unit 1, Cycle 8, Revision 2," April 1981.

3-4. Gergos, B. W., Editor, "Reload Safety Evaluation - San Onofre Nuclear Generating Station Unit 1, Cycle 9," March 1986.

3-5. Rinkacs, W. J., "SCE SONGS Unit 1 Steamline Break Outside Containment Mass/Energy Release Analysis," WCAP-11294, (Westinghouse Proprietary Class 2), September 1986.

3-6a. "Engineered Safety Features Single Failure Analysis," Letter from M. O. Medford (SCE) to USNRC, November 20, 1987.

3-6b. "Requests for Additional Information," Letter from M. O. Medford (SCE) to G. E. Lear (NRC), May 1, 1986.

3-7. "San Onofre Unit 1 (SCE) Disabling of Turbine Runback", SCE-86-557, Letter from L. E. Elder (W) to J. L. Rainsberry (SCE), May 30, 1986.

3-8. Morita, T., et. al., "Dropped Rod Methodology for Negative Flux Rate Trip Plants," WCAP-10298-A (Westinghouse Proprietary Class 3), June 1983.

3-9. Risher, D. H., et. al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9273-A (Westinghouse Proprietary Class 3), July 1985.

References (Continued)

3-10. WCAP-7422-L (Westinghouse Proprietary Class 2), "Topical Report-Westinghouse PWR Core Behavior Following A Loss Of Coolant Accident," January 1971.

3-11. WCAP-8339 (Westinghouse Non-Proprietary Class 3), "Westinghouse Emergency Core Cooling System Evaluation Model - Summary," June 1974.

3-12. Westinghouse Technical Bulletin NSID-TB-86-08, "Post-LOCA Long-Term Cooling: Boron Requirements," October 31, 1986.

4.0 INSTRUMENTATION AND CONTROLS SYSTEMS SAFETY EVALUATION

4.1 Introduction

Westinghouse has supplied hardware which upgrades the Nuclear Instrumentation System (NIS) for Southern California Edison Company (SCE) at SONGS 1. This NIS upgrade incorporates a design that is similar to that supplied to current nuclear generating plants. The primary differences in the SONGS 1 NIS upgrade are found in the intermediate range channel, as described previously in section 2.2.2. Incorporated into the intermediate range channel is the capability for post accident monitoring. This safety evaluation addresses the adequacy of the Westinghouse NIS hardware design relative to current regulations for nuclear plant instrumentation and control systems.

4.2 Basis

The NIS upgrade supplied by Westinghouse has been designed to conform to current regulatory design requirements regarding instrumentation and control (I&C) systems; in particular, IEEE-279-1971 (reference 4-4). In order to assess the adequacy of the system's design, the Westinghouse-supplied part of the NIS was evaluated against the functional performance and reliability requirements of section 4 of IEEE-279. These criteria establish minimum requirements for the functional performance and reliability of safety related systems for nuclear generating stations. The interface criteria for the Westinghouse supplied equipment is provided in Appendix D (SONGS 1 Functional Requirements section entitled, "Requirements For Associated Equipment" (sections 1.3.7.2 and 2.3.7.2 of the SONGS 1 Functional Requirements document)).

The following discussion addresses each of the criteria in section 4 of IEEE-279 for the Westinghouse supplied equipment.

4.2.1 General Functional Requirement

Adequate functional performance of safety related I&C systems is achieved by constructing a system that performs the requisite protective actions necessary for reactor protection, and that contains the appropriate design features to ensure the functional capability and operational readiness (reliability) of the system. The NIS automatically initiates the protective actions described in sections 2.3 and 2.4 of this document, whenever a monitored condition reaches a preset level. The formal safety analyses performed for SONGS 1 demonstrate that these actions satisfy the requirements for adequate reactor protection. The functional capability and reliability of the NIS has been assured by designing the system to conform to IEEE-279 as detailed above. The balance of this evaluation will address this conformance, such that evidence of an acceptable design is presented.

4.2.2 Single Failure Criterion

Having demonstrated, by formal safety analysis, that the protective functions provided by the NIS are adequate for reactor protection, the reliability of the system is addressed by designing a system that is capable of sustaining a single failure within the system without causing the system to suffer a loss of its functional capability.

In general, the NIS is designed to provide either two or four redundant instrumentation channels for each protective function, and two logic trains for the actuation of protective actions. These redundant channels and trains have been designed with due regard for all considerations which make the system hardened against a single failure (e.g., electrically isolated and physically separated) such that any single failure within a channel or train will not prevent protective action at the NIS level when required. Loss of input power, the most likely mode of failure, to a channel or logic train will result in a signal calling for a reactor trip.

4.2.3 Quality of Components and Modules

The system is constructed of high quality components consistent with low failure rates. These components are similar to those used in currently licensed operating plants and are subject to the provisions of Westinghouse quality assurance and design control policies.

4.2.4 Equipment Qualification & Channel Integrity

In general, Westinghouse supplied safety related equipment has been qualified by test, or a combination of test and analysis in order to reduce the potential for common mode failures due to environmental and seismic effects. Each piece of Class 1E equipment supplied for the NIS upgrade will eventually have been subjected to this rigorous acceptance program, and have documentation to support qualification of the equipment to perform its intended function during specified environmental and seismic events. Currently not all of the hardware to be supplied as part of the NIS upgrade has been qualified.

WCAP-8587 (reference 4-1) is the methodology employed by Westinghouse for qualification of safety-related electrical equipment. Equipment subjected to this program is tested in accordance with IEEE-323-1974 (reference 4-2) and IEEE Standard 344-1975 (reference 4-3). IEEE-323 specifies the sequence for worst case condition testing, i.e., seismic testing followed by environmental testing. Performing the proper sequence of testing demonstrates that there is no degradation in equipment performance for either seismic or environmental events. Since there is no degradation in equipment performance for either event, there is no concern for the case where coincident seismic and environmental events would occur. (Coincidental seismic and environmental events are considered only in the case of non-seismic grade piping.)

The environmental and seismic qualifications of the supplied NIS equipment is discussed below. The equipment title and Westinghouse Equipment Qualification Data Package (EQDP) reference number is specified.

Low Noise Source Range Preamplifier (ESE-47B)

The Low Noise Source Range Preamplifier is qualified environmentally for operation during normal and abnormal outside-containment excursions and seismically for operability before and after a design basis earthquake. (For details of environmental and seismic qualification, see ESE-47B.)

Source Range Drawer (ESE-47C) (Additional Qualification to be Provided Later)

Presently, the Source Range Drawer is equipped with a mechanical key switch that permits bypass of the high startup rate rod stop when the source range channel is out of service. Qualification of the mechanical key switch has not been completed. This keyed switch must be qualified for seismic and environmental conditions before it (and the source range drawer) is considered qualified.

Except for the mechanical key switch, the Source Range Drawer is qualified environmentally for operation during normal and abnormal outside containment excursions and seismically for operability before and after a design basis earthquake. (For details of environmental and seismic qualification (with the exception of the keyed switch), see ESE-47C.)

Power Range Drawer (ESE-10A)

Presently, the Power Range Drawer is equipped with a mechanical key switch that permits bypass of the overpower and dropped rod rod stops when the power range channel is out of service. Qualification of the mechanical key switch has not been completed. This keyed switch must be qualified for seismic and environmental conditions before it (and the power range drawer) is considered qualified.

Except for the mechanical key switch, the power range drawer is qualified environmentally for operation during normal and abnormal outside containment excursions, and seismically for operability before, during and after a design basis earthquake. (For details of environmental and seismic qualification (with the exception of the keyed switch) see ESE-10A.)

Quadaxial Cable Nickel/Quartz Cable, Triaxial Connector Plug and Jack and Connector Seal Assembly (ESE-52A) (Additional Qualification to be Provided Later)

The above items have been qualified environmentally for operation during normal and abnormal inside-containment excursions, and seismically for operability before and after a design basis earthquake. (For details of environmental and seismic qualification, see ESE-52A.) However, qualification of the listed items has not been completed. In addition to the above qualification, the listed items must be qualified for the effects of containment spray chemical attack due to low pH before the equipment is considered qualified for HELB and one year post-accident. (Additional qualification documentation is to be supplied later.)

Crimp-on Connector (ESE-8C)

The Crimp-on Connector is qualified environmentally for operation during normal and abnormal inside-containment excursions. For SONGS 1, the crimp-on connector is not required to perform during or after a HELB. The connector is seismically qualified for operability before and after a design basis earthquake. (For details of environmental and seismic qualification, see ESE-8C.)

Power Range Detector, SS Mineral Insulated Triaxial Cable with Connector and Field Connector (ESE-8D)

The above items are qualified environmentally for operation during normal and abnormal inside containment excursions and seismically for operability before, during and after a design basis earthquake. For SONGS 1, the above power range items are not required to perform during or after a HELB inside containment. (For details of environmental and seismic qualification, see ESE-8D.)

Wide Range Preamplifier -- Reg Guide 1.97 NFMS (ESE-52B Addendum 1)

The wide range preamplifier is qualified environmentally for operation during normal and abnormal outside containment excursions and seismically for operability before, during and after a design basis earthquake. (For details of environmental and seismic qualification, see ESE-52B.)

Intermediate Range Drawer (Supplement 2-E47C Addendum 2, Rev 0) (Additional Qualification to be Provided Later)

Presently, the Intermediate Range Drawer is equipped with a mechanical key switch that permits bypass of the high start-up rate reactor trip and high start-up rate rod stop when the intermediate range channel is out of service. Qualification of the mechanical key switch has not been completed. This keyed switch must be qualified for seismic and environmental conditions before it (and the intermediate range drawer) is considered qualified.

Except for the mechanical key switch, the intermediate range drawer is qualified for operation during normal and abnormal outside containment excursions, and seismically for operability before and after a design basis earthquake. (For details of environmental and seismic qualification (with the exception of the keyed switch), see Supplement 2-E47C Addendum 2, Rev 0.)

Following a Verification and Validation (V&V) program for the wide range amplifier (which is included in the intermediate range drawer), electronically programmable read only memory (EPROM) integrated circuits were burned-in which reflect the verified software. Installation of the EPROMs with the verified software provides an additional level of confidence that the microprocessor based system will meet its functional requirements in a highly reliable manner.

Source Range Detector, Quartz Insulated SS-Clad Triaxial Cable and Connector (ESE-47E)

The above items are qualified environmentally for operation during normal and abnormal inside containment excursions and seismically for operability before

and after a design basis earthquake. The source range detectors are not required to be qualified to perform during or after a HELB. (For details of environmental and seismic qualification, see ESE-47E.)

Two-section Fission Chamber (Later)

Qualification of the two-section fission chamber has not been completed. The two-section fission chamber must be qualified for the effects of containment spray chemical attack on the fission chamber due to low ph before the equipment is considered qualified.

Penetration (Later)

Qualification of the penetration has not been completed. The penetration must be qualified for the effects of containment spray chemical attack due to low ph before the equipment is considered qualified.

Coincident Logic Cabinet (ESE-67A) (Additional Qualification to be Provided Later)

Qualification of the coincident logic cabinet has not been completed with relays mounted without sockets. Additional seismic review is required, due to the addition of relay sockets and spacers. The logic cabinet (with the socket-mounted relays) must be qualified for seismic conditions before it is considered qualified.

Triaxial Cable (Later)

Review of the Rockbestos qualification document for the triaxial cable has not been completed. The Rockbestos qualification document must be reviewed and approved by Westinghouse, as it pertains to this specific plant, before it is considered qualified.

The following is a summary of intended NIS equipment qualifications:

Environment -- Normal/Abnormal (Inside and Outside Containment)

All of the supplied NIS equipment (whether inside or outside containment) is environmentally qualified for operation during normal and abnormal conditions.

Seismic -- Outside Containment

All of the outside containment equipment is to be seismically qualified for operability before and after a design basis earthquake. In addition, the Reg Guide 1.97 wide range preamplifier (ESE-52B Addendum 1), the intermediate range drawer (Supplement 2-E47C Addendum 2, Rev 0) and the power range drawer (ESE-10A) are also seismically qualified for operability during a design basis earthquake.

Seismic -- Inside Containment

All of the inside containment neutron flux detectors are seismically qualified for operability before and after a design basis earthquake. The intermediate range two-section fission chambers and the uncompensated ion chamber power range detectors are seismically qualified for operability during a design basis earthquake.

HELB -- Inside Containment

The intermediate range neutron flux detectors, Nickel/Quartz triaxial cable, quadaxial cable, penetrations, connectors, and seal assemblies are required for operation during and after a HELB inside containment. The power range and source range detectors are not required for operation during or after a HELB inside containment.

Systems Interaction

For SONGS 1, the intermediate range channels are required to remain operational after a seismic event and during and after a HELB inside containment (See sections 3.2.2.9 and 3.2.2.11). The source range and power

range channel outside containment equipment is required to remain operational after a seismic event. The source range and power range detectors are not required to remain operational during or after a HELB inside containment. Since environmental effects may cause common mode failure mechanisms of the source range and power range detectors, interaction of the source range and power range channels with the intermediate range channels must be addressed.

The source range detectors are qualified by test to demonstrate that the detectors will function before, during and after normal/abnormal environmental conditions and after a seismic event. The power range detectors are qualified for operation during normal/abnormal environmental conditions and seismically for operability before, during and after a design basis earthquake. For SONGS 1, neither the source range detectors nor the power range detectors are required to perform during or after a HELB. For a HELB, the possible source range and power range detector failures are short circuit, open circuit or interconnection of the low voltage elements with the high voltage elements. These source range and power range detector failures will not adversely impact the intermediate range channels because the high voltage power supplies for both the source range and power range detectors are current limited and are not exposed to the HELB.

In summary, since both the source range and power range detector power supplies are current limited, there is no possible failure of the source range or power range detectors that can adversely impact the intermediate range channel.

4.2.5 Independence

Channel independence is carried throughout the system, extending from the neutron sensors (and associated cabling) in containment, through the containment penetrations, to the signal conditioning equipment and associated output wiring. Independence of redundant channels is achieved by incorporating the principles of both physical and electrical separation. For example, redundant instrument channels are separated by locating modules in different cabinets. Each redundant channel set is energized from a separate AC power source.

4.2.6 Control and Protection System Interaction

The NIS is designed to be independent of all control systems to which nuclear flux signals are sent. In certain applications, the control signals and other nonprotective functions are derived from individual nuclear protective channels of the NIS through isolation amplifiers. The isolation amplifiers are classified as part of the protection system and are located in the NIS cabinets. Nonprotective functions include those signals used for remote flux indication and computer monitoring. The isolation amplifiers are designed such that a short circuit, open circuit, or the application of 120 VAC or ±150 VDC on the isolated output portion of the circuit (i.e., the nonprotective side of the circuit) will not affect the input (protective) side of the circuit. This has been confirmed by proof of design testing. By design, the signals obtained through the isolation amplifiers are never returned to the protective racks.

Isolation between safety related and non-safety related signals in the coincidentor logic cabinet is achieved through the use of seismically and environmentally qualified relays. Coil to contact separation provides isolation between channel and train-oriented wiring and between safety and non-safety wiring.

Physical separation of the wiring within the logic cabinet is maintained by routing safety and non-safety wiring separately. Also, metal barriers within the cabinet provide physical separation between channel and train wiring and safety related and non-safety related terminals.

The Rod Control System is the only control system that receives input signals from the NIS. Rod movement is initiated from a difference signal between Tavg and Tref; rod speed is also determined by a temperature signal which is compensated using power range nuclear flux signals and pressurizer pressure. Therefore, failure of the nuclear flux signal alone will not cause the rods to move in or out without a specified Tavg - Tref difference signal. Since channel failure alone cannot cause the rods to move in or out without other parameter requirements being meet, there is no control and protection interaction issue with the NIS.

4.2.7 Derivation of System Inputs

All NIS inputs are derived from neutron flux detector signals which are a direct measure of neutron flux.

4.2.8 Capability for Sensor Checks

The operational availability of each NIS input sensor during reactor operation can be accomplished by cross checking between channels that bear a known relationship to each other and that have readouts available.

4.2.9 Capability for Testing

The NIS is capable of being tested during power operation. The power range channels of the NIS are tested by superimposing a test signal on the actual detector signal being received by the channel at the time of testing. Since the power range overpower trip logic is two out of four, bypass of this reactor trip function is not provided, and the output of the bistable is not placed in a tripped condition prior to testing.

An NIS channel which can cause a reactor trip from a one out of two protection logic (intermediate range) is provided with a bypass function which prevents the initiation of a reactor trip from that particular channel during the short period that it is undergoing test.

4.2.10 Channel Bypass or Removal From Operation

The NIS is designed to permit periodic testing during reactor power operation without initiating a protective action unless a trip condition actually exists. During testing, the remaining active parts of the NIS will continue to meet the single failure criterion for a reactor trip subject to the provisions of this safety evaluation as stated earlier, and the exception for "one out of two" logic as allowed by IEEE-279-1971.

4.2.11 Operating Bypasses

Permissive P-7 automatically blocks (bypasses) the low flow, low pressure and turbine reactor trips when reactor power is below 10% power. Additionally, it blocks the startup rate reactor trip and rod stop when reactor power is greater than 10%. When permissive conditions are no longer met and these protective actions are required to be active for safe plant operations, P-7 automatically reinstates the various reactor trips provided by the NIS.

Permissive P-8 automatically blocks (bypasses) the reactor trip for loss of flow in any one loop when reactor power is below 50%. When this permissive condition is no longer met (reactor power above 50%), P-8 automatically reinstates the reactor trip for a loss of flow in any one loop.

4.2.12 Indication of Bypass

Indication is provided in the control room if some part of the system has been administratively bypassed or taken out of service. Control room indications are provided when the following protective functions are bypassed: source range high startup rate rod stop, intermediate range high startup rate trip and rod stop and power range rod stop and dropped rod rod stop.

4.2.13 Access to Means of Bypassing

The NIS design requires administrative control for manually bypassing channels or protective functions.

4.2.14 Multiple Setpoints

The Westinghouse supplied equipment for monitoring neutron flux provides multiple setpoints. Control over these setpoints is provided by the customer.

4.2.15 Completion of Protective Action

The NIS is designed such that, once initiated, a protective action goes to completion. Return to normal operation requires action by the operator.

4.2.16 Manual Initiation

No manual initiation of protective action is provided by the NIS; however, the requirement for manual actuation is satisfied by the manual reactor trip function of the reactor protection system.

4.2.17 Access

The NIS design requires the administrative control to access all setpoint adjustments, module calibration adjustments, test points and the means for manually bypassing channels or protective functions.

4.2.18 Identification of Protective Actions

Channel level protective actions are identified, as discussed in sections 2.2.1, 2.2.2, and 2.2.3 of this document.

4.2.19 Information Readout

The NIS provides output indicators (or signals) which gives the operator complete information pertinent to all neutron flux signals. Any reactor trip signal will actuate an annunciator. Such protective actions are indicated and identified down to the channel level.

Annunciators are also used to alert the operator to deviations from normal operating conditions. For example, actuation of any rod stop or trip of any channel will actuate an annunciator.

4.2.20 System Repair

The NIS is designed to facilitate the recognition, location, replacement, repair or adjustment of malfunctioning components or modules.

4.3 Conclusion

Based on conformance to IEEE-279-1971, the Nuclear Instrumentation System upgrade designed and supplied by Westinghouse has demonstrated that it is capable of performing its designated safety-related functions while exposed to applicable normal, abnormal, test, accident and post-accident environmental conditions. Therefore, implementation of this Nuclear Instrumentation System upgrade will not adversely impact plant operations at SONGS 1 provided that the interface criteria are met (see sections 1.3.7.2 and 2.3.7.2 of the Functional Requirements, Appendix D) and qualification for the following Westinghouse-supplied equipment meets the criteria for Equipment Qualification and Channel Integrity.

- Source, intermediate, and power range drawer mechanical key switch (seismic and environmental)
- Quadaxial cable (low pH only)
- Two-section fission chamber (low pH only)
- Penetration (low pH only)
- Coincidentor logic cabinet (seismic and environmental)
- Triaxial cable (seismic and environmental)

4.4 References

Equipment supplied by Westinghouse for the NIS upgrade has been qualified in accordance with the following WCAP and IEEE Standards:

4-1. WCAP-8587, Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment.

4-2. IEEE-323-1974, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

4-3. IEEE-344-1975, IEEE Recommended Practices for Seismic Qualification for Class 1E Equipment for Nuclear Power Generating Stations.

4-4. IEEE-279-1971, IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations.

4-5. IEEE-317-1983, IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations.

FUNCTIONAL REQUIREMENTS

SONGS 1 Functional Requirements (Appendix D of this document), section entitled, "Requirements For Associated Equipment," provides the interface criteria.

5.0 TECHNICAL SPECIFICATIONS REVIEW

5.1 Introduction

The following summarizes the changes to SONGS 1 Technical Specifications required by the NIS upgrade. The marked-up pages corresponding to these descriptions are provided in Appendix A.

5.2 Technical Specification Changes

Technical Specification 2.1 Reactor Core

This specification contains requirements which place limits on combinations of power, pressure, and temperature. Item 3 of Table 2.1 is the Nuclear Overpower setpoint. Item 3 references a footnote which requires that the nuclear overpower trip on all channels be reduced one percent for each percent that core power distribution asymmetry exceeds 10%. Asymmetry is defined in plant procedures as the difference between the highest and lowest reading power range percent power meters.

The relocation of one power range excore detector from Thimble 4 to Thimble 3 alters the radial separation of the power range detectors from 90 degrees between each so that the detector in Thimble 3 is 45 degrees from the adjacent detector in one direction and 135 degrees from the adjacent detector in the opposite direction. This relocation potentially changes what the detectors will sense given an asymmetric power distribution and thus the resulting value for asymmetry.

An evaluation of asymmetric power distribution has shown that with the relocated power range instruments, certain cases with an asymmetry of greater than 10%, which now requires technical specification action, would show an asymmetry of less than 10% but greater than 5%. Accordingly, the technical specification allowance of up to 10% asymmetry is reduced to allow up to 5% asymmetry. The asymmetry limit in the Basis is also changed to 5%.

6.0 CONCLUSIONS

As stated in section 1.0, this document provides descriptions and details regarding evaluations performed to support a license amendment request to install the NIS upgrade at SONGS 1. It describes the features and functions of the NIS, as they compare with the current system and with systems installed in other plants. It provides safety evaluations for impact of the changes to current FSA safety analyses, i.e., LOCA, non-LOCA, and SGTR analyses, as well as for adequacy of the system design relative to regulatory criteria for nuclear plant I&C systems.

Technical specifications changes as a result of the NIS upgrade are addressed in section 5.0, and markups of current technical specifications to reflect the changes are provided in Appendix A.

As a result of this safety review of the Westinghouse-supplied equipment, as described in the previous sections, Westinghouse has determined that the upgraded NIS is capable of performing its designated safety-related functions. Therefore, implementation of the upgraded NIS will not adversely impact plant operations at SONGS 1 provided that the system meets the criteria and conditions specified in the preceeding sections. It has been demonstrated that the conclusions presented in the FSA and the current safety analyses remain valid.

APPENDIX A

to

Safety Review Report

for

NIS Upgrade

Mark-Ups of Technical Specification Pages

2.1 REACTOR CORE - Limiting Combination of Power, Pressure, and Temperature

APPLICABILITY: Applies to reactor power, system pressure, coolant temperature, and flow during operation of the plant.

OBJECTIVE: To maintain the integrity of the reactor coolant system and to prevent the release of excessive amounts of fission product activity to the coolant.

SPECIFICATION: Safety Limits

(1) The reactor coolant system pressure shall not exceed 2735 psig with fuel assemblies in the reactor.

(2) The combination of reactor power and coolant temperature shall not exceed the locus of points established for the RCS pressure in Figure 2.1.1. If the actual power and temperature is above the locus of points for the appropriate RCS pressure, the safety limit is exceeded. | 60 6/8/81

Maximum Safety System Settings

The maximum safety system trip settings shall be as stated in Table 2.1

TABLE 2.1

		Three Reactor Coolant Pumps Operating	
*1.	Pressurizer High Level	≤ 20.8 ft. above bottom of pressurizer when steam/feedflow mismatch trip is not credited, or ≤ 27.3 ft. above bottom of pressurizer when steam/feedflow mismatch trip is credited	97 4/7/86
2.	Pressurizer Pressure: High	≤ 2220 psig	49 7/19/79
**3.	Nuclear Overpower	≤ 109% of indicated full power	
***4.	Variable Low Pressure	≥ 26.15 (0.894 ΔT+T avg.) - 14341	60 6/8/81
***5.	Coolant Flow	≥ 85% of indicated full loop flow	49 7/19/79

* Credit can be taken for the steam/feedflow mismatch trip when this system is modified such that a single failure will not prevent the system from performing its safety function. | 97 4/7/86

** The nuclear overpower trip is based upon a symmetrical power distribution. If an asymmetric power distribution greater than ~~10%~~ 5% should occur, the nuclear overpower trip on all channels shall be reduced one percent for each percent above ~~10%~~ 5%

***May be bypassed at power levels below 10% of full power.

Revised: 4/28/87
Typo Revised: 5/6/87

Maximum Safety System Settings

1. Pressurizer High Level and High Pressure

In the event of loss of load, the temperature and pressure of the Reactor Coolant System would increase since there would be a large and rapid reduction in the heat extracted from the Reactor Coolant System through the steam generators. The maximum settings of the pressurizer high level trip and the pressurizer high pressure trip are established to maintain the DNB ratio above 1.30 and to prevent the loss of the cushioning effect of the steam volume in the pressurizer (resulting in a solid hydraulic system) during a loss-of-load transient.(3)(4)

In the event that steam/feedflow mismatch trip cannot be credited due to single failure considerations, the pressurizer high level trip is provided. In order to meet acceptance criteria for the Loss of Main Feedwater and Feedline Break transients, the pressurizer high level trip must be set at 20.8 ft. (50%) or less. 97 4/7/86

2. Variable Low Pressure, Loss of Flow, and Nuclear Overpower Trips

These settings are established to accommodate the most severe transients upon which the design is based, e.g., loss of coolant flow, rod withdrawal at power, inadvertent boron dilution and large load increase without exceeding the safety limits. The settings have been derived in consideration of instrument errors and response times of all necessary equipment. Thus, these settings should prevent the release of any significant quantities of fission products to the coolant as a result of transients.(3)(4)(5)

In order to prevent significant fuel damage in the event of increased peaking factors due to an asymmetric power distribution in the core, the nuclear overpower trip setting on all channels is reduced by one percent for each percent that the asymmetry in power distribution exceeds 18%. This provision should maintain the DNB ratio above a value of 1.30 throughout design transients mentioned above. 5%

The response of the plant to a reduction in coolant flow while the reactor is at substantial power is a corresponding increase in reactor coolant temperature. If the increase in temperature is large enough, DNB could occur, following loss of flow.

The low flow signal is set high enough to actuate a trip in time to prevent excessively high temperatures and low enough to reflect that a loss of flow conditions exists. Since coolant loop flow is either full on or full off, any loss of flow would mean a reduction of the initial flow (100%) to zero.(3)(6)

Revised 4/28/87
Typo Revised 5/6/87

APPENDIX B

to

Safety Review Report

for

NIS Upgrade

Drawings Referenced in Section 2.0

(To Be Supplied Later)

APPENDIX C

to

Safety Review Report

for

NIS Upgrade

Listing of

Equipment Qualification Data Packages (EQDPs)

Referenced in Section 4.0

Appendix C

Equipment Qualification Documentation for Westinghouse-Supplied Equipment

In order to support qualification of the class 1E equipment for the NIS upgrade, Westinghouse provides documentation which describes the qualification program. The qualification program (WCAP-8587) demonstrates that the mechanical and electrical portions of the NIS will perform their safety-related functions when exposed to the specified environmental and seismic events.

The following is a listing of NIS equipment, identification number (I.D.#), equipment location (inside containment IC or outside containment OC) and the Westinghouse Equipment Qualification Data Package (EQDP) reference number.

EQUIPMENT	ID NO.	LOCATION	EQDP
Low Noise S/R Preamp	NM-31A	OC	ESE-47B
Low Noise S/R Preamp	NM-32A	OC	ESE-47B
Source Range Drawer	N-31	OC	ESE-47C (Note 4)
Source Range Drawer	N-32	OC	ESE-47C (Note 4)
Power Range Drawer	N-41	OC	ESE-10A (Note 4)
Power Range Drawer	N-42	OC	ESE-10A (Note 4)
Power Range Drawer	N-43	OC	ESE-10A (Note 4)
Power Range Drawer	N-44	OC	ESE-10A (Note 4)
Quadaxial Cable	A/E Scope	IC	ESE-52A (Note 1)
Crimp-on Connector	A/E Scope	IC	ESE-8C

Power Range Detector	NE-43A	IC	ESE-8D (Note 6)
Power Range Detector	NE-43B	IC	ESE-8D (Note 6)
Power Range Detector	NE-44A	IC	ESE-8D (Note 6)
Power Range Detector	NE-44B	IC	ESE-8D (Note 6)
Wide (I/R) Range Preamp (Reg Guide 1.97 NFMS)	NM-35F	OC	ESE-52B Addendum #1
Wide (I/R) Range Preamp (Reg Guide 1.97 NFMS)	NM-36F	OC	ESE-52B Addendum #1

EQUIPMENT	ID NO.	LOCATION	EQDP
Rockbestos Cable	A/E Scope	IC/OC	Note 2
I/R Drawer	N-35	OC	Supplement 2-E47C Add 2,Rev 0 (Note 4)
I/R Drawer	N-36	OC	Supplement 2-E47C Add 2,Rev 0 (Note 4)
Source Range Detector	NE-31	IC	ESE-47E
Source Range Detector	NE-32	IC	ESE-47E
Two-section Detector	NE-41A	IC	Note 3

Two-section Detector	NE-41B	IC	Note 3
Two-section Detector	NE-42A	IC	Note 3
Two-section Detector	NE-42B	IC	Note 3
Penetration	A/E Scope	IC	Later
Penetration	A/E Scope	IC	Later
Penetration	A/E Scope	IC	Later
Penetration	A/E Scope	IC	Later
Coincident Logic Cabinet	Later	OC	ESE-67A Note 5
Coincident Logic Cabinet	Later	OC	ESE-67A Note 5
Indicator: PAM	NRI-1200-3	OC	ESE-14
Indicator: PAM	NRI-1200-4	OC	ESE-14
Indicator: PAM	NLI-1200-3	OC	ESE-14
Indicator: PAM	NLI-1200-4	OC	ESE-14
Indicator	NRI-1200-1	OC	ESE-14
Indicator	NRI-1200-2	OC	ESE-14
Indicator	NLI-1204C	OC	ESE-14
Indicator	NWI-1204C	OC	ESE-14

Indicator	NLI-1200-1	OC	ESE-14
Indicator	NLI-1200-2	OC	ESE-14
Indicator	NLI-1200-5	OC	ESE-14
Indicator	NLI-1200-6	OC	ESE-14
Indicator	NLI-1200-7	OC	ESE-14
Indicator	NLI-1200-8	OC	ESE-14

Notes:

1. In addition to the quadaxial cable, this document qualifies the following inside containment intermediate range components: connector (additional qualification to be provided later), seal assembly, and quartz/nickel triaxial cable.

2. Qualification documentation to be supplied by Westinghouse later.

3. In addition to the two-section detectors, this document will provide reference to the qualification document for the following inside containment power range components: mineral insulated triaxial cable with connector (Ref.: ESE-8A).

4. Additional keyed switch qualification to be provided later.

5. Additional seismic qualification to be provided later.

6. The power range detectors are qualified for operation five minutes into a HELB provided that the installation procedures listed in ESE-8D are followed. HELB qualification of the power range detectors is not a requirement for SONGS 1, but is included here for informational purposes only.

APPENDIX D
to
Safety Review Report
for
NIS Upgrade

Functional Requirements

FUNCTIONAL REQUIREMENTS

1.0 NUCLEAR INSTRUMENTATION SYSTEM (SOURCE AND INTERMEDIATE RANGE)

System Description (For Information Only):

The Nuclear Instrumentation System (NIS) is composed of three subsystems, the source range instrumentation channels and the intermediate range instrumentation channels form the first two overlapping steps in the nuclear protection system involving the total NIS; the power range provides the third portion of the system. The source range and intermediate range instrumentation system are primarily used to provide reactor protection during startup by supplying a high startup rate rod stop and a high startup rate reactor trip.

Each subsystem (Source and Intermediate Range) has two separate channels and functions with one out of two logic. The source range instrumentation provides a high startup rate rod stop, an audible count rate, and visual display of count rate and startup rate. The intermediate range system provides an intermediate range high startup rate reactor trip, high startup rate rod stop, and displays reactor power or count rate, and startup rate. The intermediate range system is also qualified as a neutron flux Post Accident Monitoring System (PAMS).

1.1 APPLICABLE CRITERIA & STANDARDS

The following criteria apply to this system. (For titles and subject of each document, refer to the listing of criteria in Table III to this set of functional requirements documents.)

1.1.1 NRC General Design Criteria (GDC) (7/7/74):
GDC: 1, 2, 3, 4, 10, 12, 13, 15, 19, 20, 21, 22, 23, 24, 29

1.1.2 Institute of Electrical & Electronics Engineers (IEEE) Standards:
IEEE Std. 279-1971, 338-1975, 379-1972, 384-1974, 323-1974, 344-1975

1.1.3 American National Standards Institute (ANSI) Standards:
ANSI N18.2-1973 and ANSI N18.2a-1975, ANSI 18.8

1.14 Westinghouse Nuclear Safety Position Paper (NSPP):
NSPP 3.1-2, 3.1-3, 3A-1.22, 3A-1.47, 3A-1.53, 3A-1.75, 3A-1.105, 3A-1.118

1.2 ENVIRONMENTAL REQUIREMENTS

Table I supplies generic plant normal operating condition parameters for various areas within a typical plant. The operating condition parameters specified include the following: temperature, relative humidity and pressure. This data is based on available A/E interface information.

The table also provides the operating condition parameters for abnormal operating conditions. The loss of heating, ventilation and air conditioning is considered an abnormal event. Abnormal conditions apply only to plants which do not have Class 1E ventilation and/or air conditioning systems.

Table II supplies generic plant design basis operating condition parameters for various areas within a typical plant for the following postulated events: double ended main steamline rupture, small steamline rupture, operating basis earthquake, safe shutdown earthquake, double ended loss of coolant accident, small loss of coolant accident, and main feedline rupture.

1.2.1 Plant Design Normal and Abnormal Operating Conditions

1.2.1.1 Temperature

The minimum and maximum expected temperatures for plant normal and abnormal operating conditions are specified.

1.2.1.2 Relative Humidity

The minimum and maximum expected relative humidity for plant normal and abnormal operating conditions are specified.

1.2.1.3 Pressure

The minimum and maximum expected pressure for plant normal and abnormal operating conditions are specified.

1.2.1.4 Radiation

The maximum expected radiation environment for plant normal and abnormal operating conditions is specified in WCAP-8587, Supplement 1.

1.2.1.5 Chemical Environment

No zone within the plant is subjected to a chemical environment during normal and abnormal operating conditions.

2.2.2 Plant Operating Conditions for Design Basis Events

The Source and Intermediate Range Protection System must be qualified for the following events: operating basis earthquake and the safe shutdown earthquake.

2.2.2.1 Vibration

The vibration due to seismic activity is specified for each zone as a function of the applicable design basis event.

2.2.3 Time Duration of Requirements Following Initiation of Design Basis Events

2.2.3.1 Environmental Considerations

Not applicable

2.2.3.2 Seismic Considerations

The equipment must be qualified for a time period consistent with that specified in WCAP-8587.

1.3 DESIGN REQUIREMENTS

1.3.1 Accuracy

1.3.1.1 Source Range

Reproducibility shall be within ±5% of the linear full scale analog voltage. (This includes all error contributions). Reproducibility shall apply assuming constant reactor coolant system temperature and pressure conditions. This accuracy requirement applies to both indication and recording.

1.3.1.2 Intermediate Range

Reproducibility shall be within ±5% of the linear full scale analog voltage. (This includes all error contributions). In addition, the reproducibility shall be within ±1% of the linear full scale analog voltage in the range of 10 to 100% of full power. Reproducibility shall apply assuming constant reactor coolant temperature and pressure conditions. This accuracy requirement applies to indication, recording and trip functions.

1.3.2 Range

Source Range	6 decades of neutron flux (10^0 to 10^6 counts/second); corresponds to 0 to full scale analog voltage.
Intermediate Range	9 decades of neutron flux (10^{-7} to 200% power); corresponds to 0 to full scale analog voltage and overlapping all of the source range and all of the power range.
Intermediate Range reactor power	10^{-7} to 200% power
Intermediate Range Count rate	0.1 to 10^6 counts/second
Source Range Count Rate	1 to 10^6 counts/second
Startup Rate (Source and Intermediate)	-0.5 to 5.0 decades/minute

1.3.3 Time Response

The trip signals must be generated within the following time periods under all operating conditions. These time delays include the NIS sensor response time. The sensor response time is based on 500 microseconds, but can vary from this value as long as the total time delays mentioned below are met.

The High Startup Rate Reactor Trip and Rod Stops must be actuated within 3.0 seconds for intermediate range signals above 10^{-3}% power and source range signals above 8000 counts/second. The High Startup Rate Reactor Trip and Rod Stops must be actuated within 10.0 seconds for intermediate range signals below 10^{-3}% power and source range signals below 8000 counts/second. This time delay is defined as the time required for a signal to reach the reactor trip breakers or rod control mechanism (which ever is applicable) following a step input of the sensor output, from 10% below to 10% above the trip setpoint (based on a sensor delay of 500 microseconds).

1.3.4 Noise Levels

The peak to peak noise shall be no more than 1% of the linear full scale analog voltage for the intermediate range channels. The source range noise level should be such that the count rate immediately following core loading is discernible. Spikes due to starting or stopping equipment should not cause the source range signal to vary by more than 0.5 decade when in the region of overlap between the source and intermediate ranges. The above noise requirements should not include any process signal noise, e.g. fluctuations in neutron flux, but should include all noise generated from detecting the signal onward.

1.3.5 Controller Transfer Functions

Not applicable.

1.3.6 Requirements for Test and Calibration

All protection channels should be supplied with sufficient redundancy to provide the capability for channel calibration and test at power. The capability for channel calibration and test is required only in the range in which the protection channel function is active.

In the case of 1/N logic a bypass must be provided to prevent a reactor trip during test.

1.3.7 Requirements for Associated Equipment

1.3.7.1 The nuclear startup protection channels should be designed so that upon loss of electrical power to any channel, the output of that channel shall be a trip signal.

1.3.7.2 The following interface criteria must be met in order to ensure that the Westinghouse supplied portion of the Nuclear Instrumentation System is in compliance with IEEE-279-1971.

IEEE-279-1971 Section 4.4 Equipment Qualification
IEEE-279-1971 Section 4.5 Channel Integrity
IEEE-279-1971 Section 4.6 Channel Independence
IEEE-279-1971 Section 4.7 Control and Protection Interaction

FUNCTIONAL REQUIREMENTS

2.0 NUCLEAR INSTRUMENTATION SYSTEM (POWER RANGE)

System Description (For Information Only)

The Nuclear Power Range System provides the third overlapping step in the Nuclear Protection System involving the NIS (Nuclear Instrumentation System); the source and intermediate range systems providing the first and second portions of the system. The major functions of the Power Range channels are: 1) Provide various at-power reactor trips (high, mid and low range overpower); and, 2) Input to various operational permissives. Other functions of the system are: provide power range low power, mid power and overpower rod stops and dropped rod/rod stop. These functions are accomplished through the use of four two-section power range detectors and associated logic and indications.

2.1 APPLICABLE CRITERIA & STANDARDS

The following criteria apply to this system. (For titles and subject of each document, refer to the listing of criteria in Table III to this set of functional requirements documents.)

2.1.1 NRC General Design Criteria (GDC) (7/7/74):
GDC 1, 2, 3, 4, 10, 12, 13, 15, 19, 20, 21, 22, 23, 24, 25, 29

2.1.2 Institute of Electrical & Electronics Engineers (IEEE) Standards:
IEEE Std. 279-1971, 323-1974, 338-1975, 379-1972, 384-1974, 344-1975

2.1.3 American National Standards Institute (ANSI) Standards: ANSI N18.2-1973, and ANSI N18.2a-1975, ANSI 18.8

2.1.4 Westinghouse Nuclear Safety Position Papers (NSPP):
NSPP: 3.1-2, 3.1-3, 3.10-1, 3A-1.22, 3A-1.29, 3A-1.47, 3A-1.53, 3A-1.75, 3A-1.89, 3a-1.100, 3A-1.105, 3A-1.118, 7.2-2

2.2 ENVIRONMENTAL REQUIREMENTS

Table I supplies generic plant normal operating condition parameters for various areas within a typical plant. The operating condition parameters specified include the following: temperature, relative humidity and pressure. This data is based on available A/E interface information.

The table also provides the operating condition parameters for abnormal operating conditions. The loss of heating, ventilation and air conditioning is considered an abnormal event. Abnormal conditions apply only to plants which do not have Class 1E ventilation and/or air conditioning systems.

Table II supplies generic plant design basis operating condition parameters for various areas within a typical plant for the following postulated events: double ended main steamline rupture, small steamline rupture, operating basis earthquake, safe shutdown earthquake, double ended loss of coolant accident, small loss of coolant accident, and main feedline rupture.

2.2.1 Plant Design Normal and Abnormal Operating Conditions

2.2.1.1 Temperature

The minimum and maximum expected temperatures for plant normal and abnormal operating conditions are specified.

2.2.1.2 Relative Humidity

The minimum and maximum expected relative humidity for plant normal and abnormal operating conditions are specified.

2.2.1.3 Pressure

The minimum and maximum expected pressure for plant normal and abnormal operating conditions are specified.

2.2.1.4 Radiation

The maximum expected radiation environment for plant normal and abnormal operating conditions is specified in WCAP-8587, Supplement 1.

2.2.1.5 Chemical Environment

No zone within the plant is subjected to a chemical environment during normal and abnormal operating conditions.

2.2.2 Plant Operating Conditions for Design Basis Events

The Nuclear Power Range Protection System must be qualified for the following events: operating basis earthquake and the safe shutdown earthquake.

2.2.2.1 Vibration

The vibration due to seismic activity is specified for each zone as a function of the applicable design basis event.

2.2.3 Time Duration of Requirements Following Initiation of Design Basis Events

2.2.3.1 Environmental Considerations

Not applicable

2.2.3.2 Seismic Considerations

The equipment must be qualified for a time period consistent with that specified in WCAP-8587.

2.3 DESIGN REQUIREMENTS

2.3.1 Accuracy

Linearity and reproducibility of the power range (high, mid and low) neutron flux signals should be within ±1% of full power. (This includes all error contributions.) Reproducibility should apply assuming constant reactor coolant system temperature and pressure conditions.

System indicators on the control board should have an accuracy of at least ±2% of full scale and a readability of at least ±1% of full scale.

2.3.2 Range

Power Range Flux	1 to 120% of Full Power
Power Range Axial Offset	-20% to + 20%
τ (Power Range Rod Drop)	OFF, 1 to 30 seconds

Equipment should be capable of providing a signal for recording overpower excursions to 200% of full power.

2.3.3 Time Response

The trip signals must be generated within the following time periods under all operating conditions. These time delays include the NIS sensor response time. The sensor response time is based on 500 microseconds, but can vary from this value as long as the total time delays mentioned below are met.

The Power Range (high, mid and low range) Neutron Flux Reactor Trip must be actuated within 0.2 second. This time delay is defined as the time required for a signal to reach the reactor trip breakers following a step input of the sensor output, from 5% below to 5% above the trip setpoint (based on a sensor delay of no greater than 500 microseconds). All automatic interlock circuits must be actuated within 0.2 second. This time delay is defined as the time required following a step input of the sensor output from 5% below (above) to 5% above (below) the setpoint until an output signal is obtained (based on a sensor delay of no greater than 500 microseconds). All manual interlock circuits must take action upon their respective circuits within 1.0 second.

2.3.4 Noise Levels

The peak-to-peak noise should be limited to 0.5% of output span for any noise occurring in a frequency range which could affect downstream modules or systems. The noise limitation does not apply to process signal noise, e.g., fluctuations in applicable process variables, but should apply to all noise generated from detecting the signal onward. Where applicable, the requirement should be met with all lead, lag, and filter time constants set to OFF, and module gains set to 1.

2.3.5 Controller Transfer Functions

The neutron flux channels for the power range rod drop protection should have the following transfer function:

$$\frac{\tau s}{1+\tau s}$$

2.3.6 Requirements for Test and Calibration

All protection channels should be supplied with sufficient redundancy to provide the capability for channel calibration and test at power.

Removal of one trip circuit should be accomplished by placing that circuit in the tripped mode.

Testing can be achieved by superimposing test signals on detector signals.

Provisions should be made in the control room to calibrate the power range channels to agree with Nuclear Steam Supply System power as determined from calorimetric measurements.

In the case of 1/N logic, a bypass must be provided to prevent undesirable protection system action during a channel test.

2.3.7 Requirements for Associated Equipment

2.3.7.1 The reactor protection channels should be designed so that upon loss of electrical power, the resultant output of a channel is a trip signal.

2.3.7.2 The following interface criteria must be met in order to ensure that the Westinghouse supplied portion of the Nuclear Instrumentation System is in compliance with IEEE-279-1971.

IEEE-279-1971 Section 4.4 Equipment Qualification
IEEE-279-1971 Section 4.5 Channel Integrity
IEEE-279-1971 Section 4.6 Channel Independence
IEEE-279-1971 Section 4.7 Control and Protection Interaction

TABLE I

NORMAL AND ABNORMAL OPERATING ENVIRONMENTS

General Area	Zone Description	Zone Code	Typical Areas	Range	Normal Operation Temp (°F)	Normal Operation RH (%)	Normal Operation Pres (psig)	Abnormal Operation* Time Limit	Abnormal Operation* Temp (°F)	Abnormal Operation* RH (%)	Abnormal Operation* Pres (psig)
In Containment	Inaccessible	IC/I	Inside Sec. Shield	Max.	135	70	+0.3	8 hours	150	95	Atmos.
				Min.	65	20	-0.1		50	0	Atmos.
	Accessible	IC/O	Outside Sec. Shield	Max.	120	70	+0.3	∞	120	95	Atmos.
				Min.	65	20	-0.1		50	0	Atmos.
Out of Containment	Air-Conditioned	OC/AC	Control room, Aux. Equip Room	Max.	80	95	Atmos.	12 hrs.	[82 to 120]	[95 to 35]	Atmos.
				Min.	60	30	Atmos.		40	0	Atmos.
	Ventilated	OC/V	Aux. building, Safeguards etc.	Max.	104	70	Atmos.	12 hrs.	[82 to 120]	[95 to 35]	Atmos.
				Min.	60	20	Atmos.		40	0	Atmos.
	Non-Ventilated	OC/NV	Turbine-Hall	Max.	104	70	Atmos.	∞	[82 to 120]	[95 to 35]	Atmos.
				Min.	60	20	Atmos.		40	0	Atmos.

*Abnormal parameters are applicable only for areas not controlled by Class 1E ventillation and air conditioning systems.

TABLE II

DESIGN BASIS OPERATING ENVIRONMENTS

General Area	Parameter	DBE FLB/SLB	DBE LOCA	DBE Seismic	POST DBE FLB/SLB	POST DBE LOCA	POST DBE Seismic
In Containment							
	Temperature (^{o}F)	Figure 1	Figure 2	Ambient Conditions	Figure 1	Figure 2	Ambient Conditions
	Pressure (psig)	Figure 1	Figure 2		Figure 1	Figure 2	
	Humidity (%RH)	100	100		100	100	
	Radiation (R)	See Post DBE	See Post DBE		$3.9 \times 10^{4}\gamma$ $6.4 \times 10^{5}\beta$	$4.1 \times 10^{7}\gamma$ $9 \times 10^{8}\beta$	
	Chemicals	Figure 1	Figure 2		Figure 1	Figure 2	
	Vibration	None	None		None	None	
	Acceleration (g)	None	None		None	None	
Out of Containment							
	Temperature (^{o}F)	Figure 1	Ambient Conditions	Ambient Conditions	Figure 1	Ambient Conditions	Ambient Conditions
	Pressure (psig)	Figure 1			Figure 1		
	Humidity (%RH)	100			100		
	Radiation (R)	$<10^{4}\gamma$ $<10^{5}\beta$			$3.9 \times 10^{4}\gamma$ $6.4 \times 10^{5}\beta$		
	Chemicals	None			None		
	Vibration	None			None		
	Acceleration (g)	None			None		

Figure 1. Environmental Design Conditions
-- Main Steam Line Break and Feedline Break --

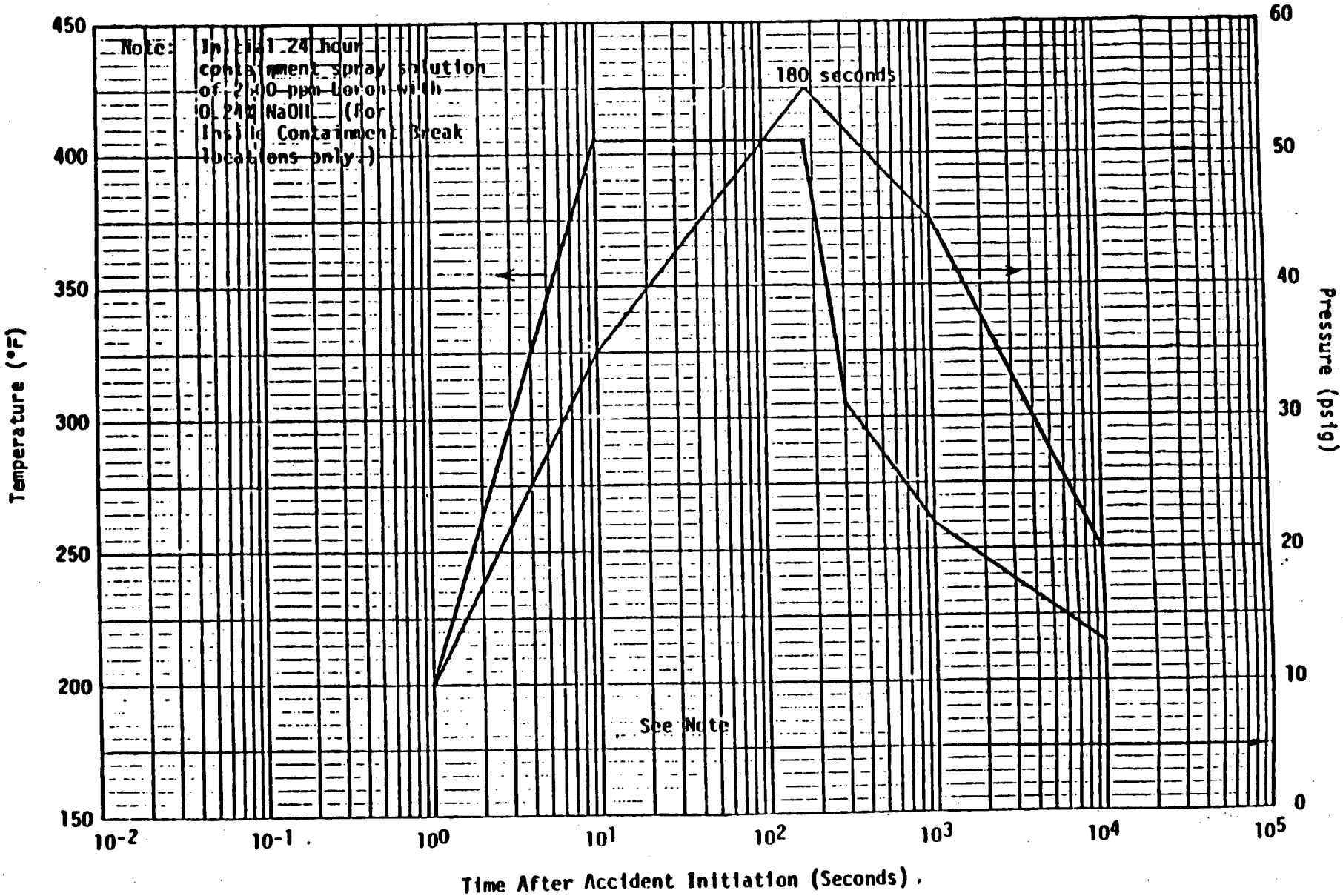


Figure 2. Containment Environmental Conditions
-- LOCA --

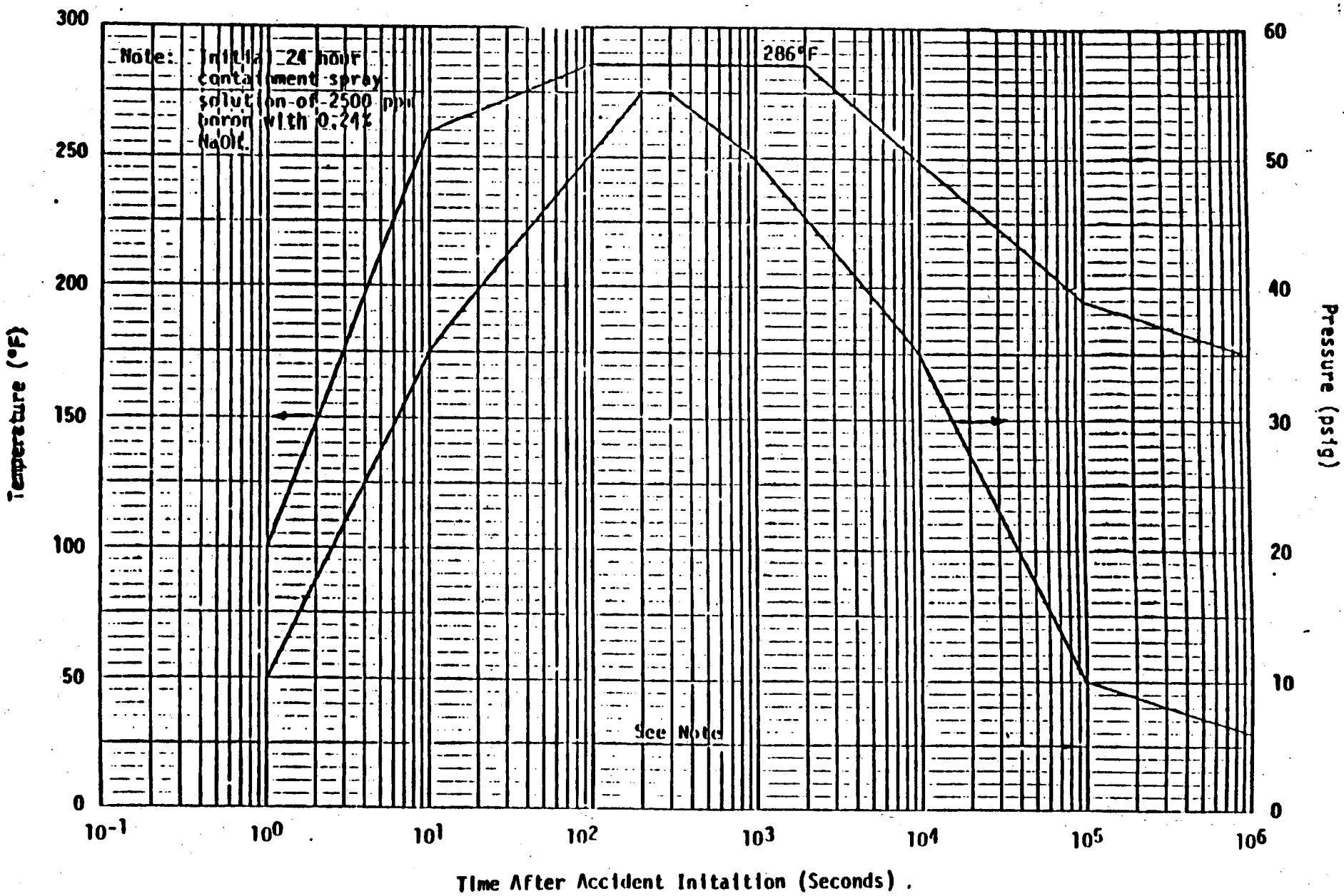


Table III

LISTING OF APPLICABLE CRITERIA

CRITERIA	TITLE
1. General Design Criteria (GDC), Appendix A to 10 CFR Part 50	General Design Criteria for Nuclear Power Plants
GDC 1	Quality Standards and Records
GDC 2	Design Bases for Protection Against Natural Phenomena
GDC 3	Fire Protection
GDC 4	Environmental and Missile Design Bases
GDC 10	Reactor Design
GDC 12	Suppression of Reactor Power Oscillations
GDC 13	Instrumentation and Control
GDC 15	Reactor Coolant System Design
GDC 19	Control Room
GDC 20	Protection System Functions
GDC 21	Protection System Reliability and Testability
GDC 22	Protection System Independence

GDC 23	Protection System Failure Modes
GDC 24	Separation of Protection and Control Systems
GDC 25	Protection System Requirements for Reactivity Control Malfunctions
GDC 29	Protection Against Anticipated Operational Occurrences

2. Institute of Electrical and Electronics Engineers (IEEE) Standards:

IEEE Std. 279-1971 (ANSI N42.7-1972)	Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE Std. 323-1974	IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
IEEE Std. 338-1975	Criteria for the Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems
IEEE Std. 344-1975	IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
IEEE Std. 379-1972 (ANSI N41.2)	Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems
IEEE Std. 384-1974 (ANSI N41.14)	Criteria for Separation of Class 1E Equipment and Circuits

3. American National Standards Institute (ANSI) Standards:

ANSI N18.2-1973 and N18.2a-1975	Nuclear Safety Criteria for the Design of Pressurized Water Reactor Plants
ANSI N18.8-1973	Criteria for Preparation of Design Bases for Systems that Perform Protective Functions in Nuclear Power Generating Stations

4. Westinghouse NES Nuclear Safety Position Papers

3.1.2	Common Mode Failures (1/8/75)
3.1-3	Single Failure Criteria (1/8/75)
3.10-1	Seismic Testing of Instrumentation and Electrical Equipment (1/8/75)
3A-1.22	Regulatory Guide 1.22 - Periodic Testing of Protection System Actuation Functions (1/10/75)
3A-1.29	Regulatory Guide 1.29 - Seismic Design Classification (1/8/75)
3A-1.47	Regulatory Guide 1.47 - Bypassed and Inoperable Status Indication for Nuclear Power Plants (1/10/75)
3A-1.53	Regulatory Guide 1.53 - Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (1/10/75)

3A-1.75	Regulatory Guide 1.75 - Physical Independence of Electric Systems (5/19/75)
3A-1.89	Regulatory Guide 1.89 Qualifications of Class 1E Equipment for Nuclear Power Plants (10/6/75)
3A-1.100	Regulatory Guide - Seismic Qualification of Electric Equipment for Nuclear Power Plants.
3A-1.105	Regulatory Guide - Instrument Spans and Setpoints
3A-1.118	Regulatory Guide - Periodic Testing of Electric Power and Protection Systems
7.2-2	Operating with a Loop Out of Service (N-1 Loop Operation) (1/9/75)

NIS INSTALLATION DESIGN

The SONGS 1 NIS installation design provides the interface requirements to install the Westinghouse (W) furnished equipment in the plant and the electrical systems to interconnect the various individual pieces of W NIS equipment and interface these with existing plant systems. The description of the W furnished portions of the upgraded NIS system are provided in the W Safety Review Report (SRR) for NIS SONGS 1. In addition, the SRR evaluates how the W system conforms to appropriate industry and regulatory standards and demonstrates its capability for performing the required safety related functions.

The NIS installation design provides for the physical installation of the W furnished equipment as follows: electrical and electronics cabinets for the NIS, coincidentors, and preamplifiers; four electrical penetrations, source range, intermediate range and power range excore neutron flux detectors, quadaxial and triaxial electrical cable and connectors. The NIS installation design is in accordance with the seismic and environmental qualification requirements of the W equipment.

The NIS installation design retains the following Main Control Room equipment in the upgraded NIS system: Delta flux recorder, Events recorder, Flux Feedback normal/defeat switch, NIS recorder, Plant Annunciator windows and axial offset calculator.

In addition to the W equipment described above, the NIS installation design replaces additional equipment as follows: In the main control room the twelve neutron flux power level and start up rate indicators, the two axial offset indicators, the controls for the audio count rate speaker and the mode selector switch are replaced. At the remote shutdown panel and the neutron flux indicator is also replaced.

The NIS installation design removes certain equipment which is no longer required because the required functions are performed by other equipment in the upgraded NIS. The removed equipment consists of the main control room delta flux monitoring channel drawer and the remote shutdown panel log power channel drawer.

The electrical installation design includes raceway and cable systems to provide vital bus electrical power to the upgraded NIS equipment and to interconnect the W furnished neutron flux detectors, containment electrical penetration assemblies, preamplifiers, NIS electronics processing cabinets and coincidentor logic cabinets. The electrical design provides the interface with existing plant systems such as the Reactor Control and Protection equipment, main control room indication, controls, annunciation, recorders, Onsite Technical Support Center (OTSC) Computer, and remote shutdown panel. The electrical design also provides new lighting for the NIS and coincidentor cabinets location.

The structural installation design provides for the proper mounting, fastening and structural support systems to correctly install each piece of W NIS equipment in accordance with the W seismic qualification for each equipment. The design also includes the necessary supports for the new electrical raceways.

The mechanical installation design includes the provisions for the physical installation of the electrical penetrations weld neck flanges and test features and for the installation of BISCO seals to maintain fire barrier rating in accordance with fire protection requirements.

The installation design of the upgraded NIS has been performed, as interface design conditions permit, in accordance with the applicable codes, standards and regulations shown on the list provided as part of this attachment. This list is taken from the San Onofre Nuclear Generating Station, Unit 1 Retrofit General Design Criteria Manual and the Nuclear Instrumentation System (Upgrade) Design Criteria.

0896P:LAB

1.5 NRC Regulations

1.5.1 Code of Federal Regulations

Title 10, Chapter I-Nuclear Regulatory Commission. Specifically the following Parts:

Part 20 Standards for Protection Against Radiation

Part 21 Reporting of Defects and Noncompliance

Part 50 Domestic Licensing of Production and Utilization Facilities, including all Appendices

Part 73 Physical Protection of Plants and Materials

Part 100 Reactor Site Criteria

1.5.2 NRC Regulatory Guides

The following NRC Regulatory Guides which are identified by (*) or (**) shall be complied with as noted. The other listed Regulatory guides shall be complied with to the extent applicable as determined by commitments made to the NRC. When NRC Regulatory Guides not listed below are applicable to a specific work package they shall be identified in that Work Package Design Criteria (a Project Engineer's memorandum is not adequate in this case).

Guide No.	Rev/Date	Title
1.4	2 6/74	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors
1.7	2 11/78	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident
1.11	- 3/71	Instrument Lines Penetrating Primary Reactor Containment

Guide No.	Rev/Date	Title
1.17**	- 10/71	Protection of Nuclear Power Plants Against Industrial Sabotage
1.21	1 6/74	Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants
1.26**	2 6/75	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants
1.28**	- 6/72	Quality Assurance Program Requirements (Design and Construction)
1.29*	1 8/73	Seismic Design Classification
1.30**	- 8/72	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment
1.32	2 2/77	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants
1.33**	2 2/78	Quality Assurance Program Requirements (Operation)
1.37**	- 3/73	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants
1.38**	2 5/77	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants
1.39**	2 9/77	Housekeeping Requirements for Water-Cooled Nuclear Power Plants
1.40	- 3/73	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants

* No specific commitment exists. Revision 1 to be used on a case-by-case basis as outlined in TQAM Chapter 8-C.

**Full compliance commitment per TQAM, Appendix II.

Guide No.	Rev/Date	Title
1.48	- 5/73	Design Limits and Loading Combinations for Seismic Category I Fluid System Components
1.52	2 3/78	Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants
1.53	- 6/73	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems
1.54**	- 6/73	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
1.58**	1 9/80	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel
1.60	1 12/73	Design Response Spectra for Seismic Design of Nuclear Power Plants
1.61	- 10/73	Damping Values for Seismic Design of Nuclear Power Plants
1.62	- 10/73	Manual Initiation of Protective Actions
1.63	2 7/78	Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants
1.64**	2 6/76	Quality Assurance Requirements for the Design of Nuclear Power Plants
1.67	- 10/73	Installation of Overpressure Protection Devices
1.69	- 12/73	Concrete Radiation Shields for Nuclear Power Plants
1.73	- 1/74	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

**Full compliance commitment per TQAM, Appendix II.

Guide No.	Rev/Date	Title
1.74**	- 2/74	Quality Assurance Terms and Definitions
1.75	2 9/78	Physical Independence of Electric Systems
1.80	- 6/74	Preoperational Testing of Instrument Air Systems
1.88**	2 10/76	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records
1.89	1 6/84	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants (Task EE 042-2)
1.92	1 2/76	Combining Modal Responses and Spatial Components in Seismic Response Analysis
1.94**	1 4/76	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Steel During the Construction Phase of Nuclear Power Plants
1.97	1 8/77	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident
1.100	1 8/77	Seismic Qualification of Electric Equipment for Nuclear Power Plants
1.101	2 10/81	Emergency Planning and Preparedness for Nuclear Power Reactors
1.105	1 11/76	Instrument Setpoints
1.116**	0 6/76	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems
1.118	2 6/78	Periodic Testing of Electric Power and Protection Systems
1.122	1 2/78	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components

**Full compliance commitment per TQAM, Appendix II.

Guide No.	Rev/Date		Title
1.123**	1	7/77	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
1.131	-	8/77	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants
1.141	-	4/78	Containment Isolation Provisions for Fluid Systems
1.144**	1	9/80	Auditing of Quality Assurance Programs for Nuclear Power Plants
1.146**	-	8/80	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants
8.8**	3	6/78	Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable

**Full compliance commitment per TQAM, Appendix II.

TABLE 1.4.4-1 DESIGN CODES AND STANDARDS

NOTE: This Table includes codes and standards of general applicability to SONGS 1. The specific edition and date of the code or standard applicable to a Work Package shall be specified in either the Work Package Design Criteria or the Project Engineer's Design Criteria Memorandum.

	AIR CONDITIONING AND REFRIGERATION INSTITUTE
ARI 210	Unitary Air Conditioning Equipment
ARI 240	Unitary Heat Pump Equipment
ARI 410	Standard for Forced Circulation, Air Cooling and Air Heating
	AIR MOVING AND CONDITIONING ASSOCIATION
AMCA 99	Standards Handbook (Fan Design)
AMCA 210	Test Code for Air Moving Devices
AMCA 211	Certified Ratings Program for Air Moving Devices
AMCA 300	Test Code for Sound Ratings
	AMERICAN CONCRETE INSTITUTE
ACI	Manual of Concrete Practice, Parts 1, 2, 3, 4 and 5
ACI 301	Specifications for Structural Concrete for Buildings
ACI 318	Building Code Requirements for Reinforced Concrete
ACI 349	Code Requirements for Nuclear Safety-Related Concrete Structures
ACI 531	Building Code Requirements for Concrete Masonry Structures
ACI 531.1	Specification for Concrete Masonry Construction
	AMERICAN INSTITUTE OF STEEL CONSTRUCTION
AISC	Manual of Steel Construction
	AMERICAN IRON AND STEEL INSTITUTE
AISI	Specification for the Design of Light Gage Cold-Formed Steel Structural Members, with Commentary and Supplement

AMERICAN NUCLEAR SOCIETY

ANS 51.1	Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants.
ANS 51.10	Auxiliary Feedwater System for Pressurized Water Reactors

AMERICAN NATIONAL STANDARDS INSTITUTE

ANSI A11.1	Standard Practice for Industrial Lighting
ANSI A58.1	Building Code Requirements for Minimum Design Loads in Buildings and Other Structures
ANSI A85.1	Standard Practice for Protective Lighting
ANSI B16.5	Steel Pipe Flanges, Flanged Valves, and Fittings
ANSI B16.9	Factory-Made Wrought Steel Buttwelding Fittings
ANSI B16.10	Face-to-Face and End-to-End Dimensions of Ferrous Valves
ANSI B16.11	Forged Steel Fittings, Socket Welding and Threaded
ANSI B16.25	Buttwelding Ends
ANSI B16.34	Steel Buttwelding End Valves
ANSI B30.2	Overhead and Gantry Cranes
ANSI B31.1	Power Piping
ANSI C37	Power Switchgear
ANSI C50.2	Alternating-Current Induction Motors, Induction Machines in General, and Universal Motors
ANSI C50.10	General Requirements for Synchronous Machines
ANSI C50.12	Salient Pole Synchronous Generators and Condensers
ANSI C50.13	Requirements for Cylindrical-Rotor Synchronous Generators
ANSI C57	Transformers, Regulators and Reactors
ANSI N14.6	Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials
ANSI N18.5	Earthquake Instrumentation Criteria for Nuclear Power Plants

ANSI N18.17	Industrial Security for Nuclear Power Plants
ANSI N45.2	Quality Assurance Program Requirements for Nuclear Power Plants
ANSI N101.2	Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities
ANSI N509	Nuclear Power Plant Air Cleaning Units and Components
ANSI N510	Testing of Nuclear Air-Cleaning Systems
ANSI N658	Single Failure Criteria for PWR Fluid Systems (ANS 51.7)
ANSI S1.2	Method for the Physical Measurement of Sound

AMERICAN SOCIETY OF CIVIL ENGINEERS

ASCE Paper No. 3269	Wind Forces on Structures

AMERICAN SOCIETY OF HEATING REFRIGERATING AND AIR-CONDITIONING ENGINEERS

ASHRAE	Guide and Data Book - Systems
	Guide and Data Book - Applications
	Guide and Data Book - Fundamentals
	Guide and Data Book - Equipment
	Guide - USAEC Health and Safety Bulletin No. 212

AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE

SEC II	Material Specifications
SEC III	Nuclear Power Plant Components
SEC V	Nondestructive Examination
SEC VIII	Pressure Vessels
SEC IX	Welding and Brazing Qualifications
SEC XI	Rules for Inservice Inspection

AMERICAN SOCIETY FOR TESTING AND MATERIALS

ASTM A36	Structural Steel
ASTM A213	Seamless Ferritic and Austenitic Alloy-Steel Boiler, Superheater, and Heat-Exchanger Tubes
ASTM A269	Stainless Steel Tubing
ASTM A307	Carbon Steel Externally and Internally Threaded Standard Fasteners
ASTM A312	Seamless and Welded Austenitic Stainless Steel Pipe
ASTM A325	High-Strength Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers
ASTM A376	Seamless Austenitic Steel Pipe for High Temperature Central-Station Service
ASTM A395	Ferritic Ductile Iron Pressure Retaining Castings for Use at Elevated Temperature (Replaces ASTM A445)
ASTM A449	Quenched and Tempered Steel Bolts and Studs
ASTM A488	Qualification of Procedures and Personnel for the Welding of Steel Castings
ASTM A514	High-Yield-Strength Quenched and Tempered Alloy Steel Plate, Suitable for Welding
ASTM A615	Deformed and Plain Billet-Steel Bars for Concrete Reinforcement
ASTM A668	Steel Forgoings, Carbon and Alloy, for General Industrial Use
ASTM D975	Diesel Fuel Oils
ASTM D1557	Moisture-Density Relations of Soils Using 10-lb (4.5 kg) Rammer and 18-in. (457 mm) Drop

AMERICAN WELDING SOCIETY

AWS A2.4	Symbols for Welding and Nondestructive Testing (including brazing)
AWS B3.0	Welding Procedure and Performance Qualification
AWS D1.1	Structural Welding Code - Steel

AWS D1.3	Structural Welding Code - Sheet Steel
AWS D1.4	Structural Welding Code - Reinforcing Steel

DIESEL ENGINE MANUFACTURERS ASSOCIATION

DEMS	Standard Practices for Low and Medium Speed Stationary Diesel and Gas Engines

HYDRAULIC INSTITUTE

HI	Hydraulic Institute Standards

ILLUMINATING ENGINEERING SOCIETY

IES	Lighting Handbook

INSTITUTE OF ELECTRICAL AND ELECTRONIC ENGINEERS

IEEE 80	Guide for Safety in Alternating-Current Substation Grounding
IEEE 115	Test Procedures for Synchronous Machines
IEEE 142	Recommended Practice for Grounding of Industrial and Commercial Power Systems
IEEE 143	Application Guide for: Ground-Fault Neutralizers; Grounding of Synchronous Generator Systems; Neutral Grounding of Transmission Systems
IEEE 144	Guide for Evaluating the Effect of Solar Radiation on Outdoor Metal-Clad Switchgear
IEEE 279	Criteria for Protection Systems for Nuclear Power Generating Stations (ANSI N12.7)
IEEE 288	Guide for Induction Motor Protection
IEEE 308	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE 317	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations
IEEE 323	Qualifying Class 1E Equipment for Nuclear Power Generating Stations
IEEE 334	Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations

IEEE 336	Installation, Inspection, and Testing Requirements for Class 1E Instrumentation and Electric Equipment at Nuclear Power Generating Stations
IEEE 338	Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems
IEEE 344	Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
IEEE 352	Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE 379	Application of the Single-Failure Criterion to Nuclear Power Generating Station Class 1E Systems
IEEE 380	Definitions of Terms Used in IEEE Standards on Nuclear Power Generating Stations
IEEE 382	Trial-Use Guide for Type Tests of Class I Electric Valve Operators for Nuclear Power Stations.
IEEE 382	Qualifications of Safety-Related Valve Actuators
IEEE 383	Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations
IEEE 384	Standard Criteria for Independence of Class 1E Equipment and Circuits
IEEE 387	Standard Criteria for Diesel-Generator Units Applied As Standby Power Supplies for Nuclear Power Generating Stations
IEEE 420	Trial-Use Guide for Class 1E Control Switchboards for Nuclear Power Generating Stations
IEEE 450	Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations
IEEE 484	Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations
IEEE 485	Sizing Large Lead Storage Batteries for Generating Stations and Substations
IEEE 497	Standard Criteria for Post Accident Monitoring Instrumentation for Nuclear Power Generating Stations

IEEE 498	Standard Requirements for the Calibration and Control of Measuring and Test Equipment Used in the Construction and Maintenance of Nuclear Power Generating Stations
IEEE 500	Guide to the Collection and Presentation of Electrical, Electronic, and Sensing Component Reliability Data for Nuclear Power Generating Stations
IEEE 577	Reliability Analysis in the Design and Operation of Safety Systems for Nuclear Power Generating Stations
IEEE 603	Standard Criteria for Safety Systems for Nuclear Power Generating Stations
IEEE 649	Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Station
IEEE 650	Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations

INSULATED CABLE ENGINEERS ASSOCIATION

ICEA P-46	Power Cable Ampacities (AIEE S-135-1)
ICEA P-54	Ampacities in Open-Top Cable Trays (NEMA WC51)
ICEA S-19	Rubber-insulated Wire and Cable for the Transmission and Distribution of Electrical Energy (NEMA WC3)
ICEA S-61	Thermoplastic - Insulated Wire and Cable for the Transmission and Distribution of Electrical Energy (NEMA WC5)
ICEA S-66	Cross-Linked-Thermosetting-Polyethylene Insulated Wire and Cable for the Transmission and Distribution of Electrical Energy (NEMA WC7)
ICEA S-68	Ethylene-Propylene-Rubber-Insulated Wire and Cable for the Transmission and Distribution of Electrical Energy (NEMA WC8)

INTERNATIONAL ASSOCIATION OF PLUMBING AND MECHANICAL OFFICALS

IAPMO	Uniform Plumbing Code

INTERNATIONAL CONFERENCE OF BUILDING OFFICALS

ICBO Reports	Research Committee Reports
UBC	Uniform Building Code
UBC STDS	Uniform Building Code Standards

MANUFACTURERS STANDARDIZATION SOCIETY

MSS SP-6	Standard Finishes for Contact Faces of Pipe Flanges and Connecting - End Flanges of Valves and Fittings
MSS SP-61	Hydrostatic Testing of Steel Valves
MSS SP-84	Steel Valves - Socket Welding and Threaded Ends

MASONRY INSTITUTE OF AMERICA

MIA	Masonry Code and Specification

NATIONAL ASSOCIATION OF CORROSION ENGINEERS

NACE RP-01	Control of External Corrosion on Submerged Metallic Piping Systems

NATIONAL ELECTRICAL MANUFACTURERS ASSOCIATION

NEMA AB1	Molded Case Circuit Breakers
NEMA E12	Instrument Transformers
NEMA ICS	Industrial Controls and Systems
NEMA IS1.1	Enclosures for Industrial and Control Systems
NEMA IS5	Resistance Welding Control
NEMA MG1	Motors and Generators
NEMA PB1	Panelboards
NEMA SG3	Low-Voltage Power Circuit Breakers
NEMA SG4	Alternating-Current High-Voltage Breakers
NEMA SG5	Power Switchgear Assemblies
NEMA SG5	Power Switching Equipment
NEMA TR1	Transformers, Regulators and Reactors
NEMA VE1	Cable Tray Systems

NATIONAL FIRE PROTECTION ASSOCIATION

NFPA	National Fire Codes
NFPA 70	National Electrical Code

NUCLEAR MUTUAL LIMITED

NML	Property Loss Prevention Standards for Nuclear Generating Stations

PIPE FABRICATION INSTITUTE

PFS TB1	Pressure-Temperature Ratings of Seamless Pipe Used in Power Plant Piping Systems

SHEET METAL AND AIR-CONDITIONING CONTRACTORS NATIONAL ASSOCIATION

SMACNA	Standard for Low Velocity Systems
	Low-Velocity Duct Construction Standards
	High-Pressure Duct Construction Standards

TUBULAR EXCHANGERS MANUFACTURERS ASSOCIATION

TEMA	Class R Mechanical Standards

UNDERWRITERS LABORATORIES

UL1	Flexible Metal Conduit
UL3	Flexible Nonmetallic Tubing
UL4	Armored Cable
UL6	Rigid Metal Conduit
UL38	Manually Actuated Signaling Boxes for Use with Fire-Protection Signaling Systems
UL50	Cabinets and Boxes
UL83	Thermoplastic - Insulated Wire
UL96A	Installation Requirements - - Master Labeled Lightning Protection Systems
UL168	Smoke Detectors, Photoelectric Type for Fire Protective Signaling Systems
UL193	Alarm Valves for Fire-Protection Service
UL393	Indicating Pressure Gauges for Fire-Protection Service

UL514	Outlet Boxes and Fittings
UL521	Fire-Detection Thermostats
UL789	Indicator Posts for Fire-Protection Service
UL797	Electrical Metallic Tubing
UL873	Temperature - Indicating and - Regulatory Equipment
UL924	Emergency Electric Lighting Equipment