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SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards marked-up responses to NRC 801230 questions re instrumentation & controls sys.Responses will be incorporated into FSAR amend within 4 months.

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Docket No. 50-397

August 28, 1981
G02-81-255
NS-L-02-CDT-81-044

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington D.C. 20555

Dear Mr. Schwencer:

Subject: SUPPLY SYSTEM NUCLEAR PROJECT NO. 2
RESPONSES TO INSTRUMENTATION AND
CONTROLS SYSTEMS BRANCH QUESTIONS

Reference: Letter, RL Tedesco to RL Ferguson, "Additional WNP-2 Questions (ICSB)", dated December 30, 1980.

Enclosed are sixty (60) copies of responses to the Instrumentation and Controls System Branch (ICSB) Questions transmitted to the Supply System by the referenced letter. These responses will be incorporated into the FSAR in an amendment within four months.

Very truly yours,

G D Bouchey

G. D. Bouchey
Director, Nuclear Safety

GDB/CDT/ldm

Enclosure

cc: WS Chin - BPA
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WNP-2 Files

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Q. 031.114
(7.0)

Revise the following text, tables, and figures of the FSAR to correct omissions, errors, and discrepancies:

a. The reference section (i.e., 7.2.1.1.(3)) cited in Section 7.2.1.2 does not address the concerns of this section.

b. Change the reference in the second paragraph of Section 7.3.1.1.1.2.3 from 7.3-7 to 7.3-5.

c. Change the reference to the heating, ventilating, and air conditioning (HVAC) control logic diagram in Section 7.3.1.1.7 from Figure 7.3-18 to Figure 7.3-14.

d. Revise Figure 7.4-2a to show the automatic transfer of the reactor core injection coolant (RCIC) pump suction to the suppression pool when the condensate storage tank inventory is low.

e. Complete Table 7.5-1 by providing relevant specifications for the containment instrument air line pressure and the primary containment radiation instruments.

f. The second sentence of the second paragraph in Section 7.6.1.4.1(b) is meaningless without a statement regarding the flux level associate with the reading. Correct this deficiency.

Response:

In response to the question, the following changes have been made.*

a. Amendment 10 omitted numbering and titling the last three paragraphs in Section 7.2.1.1.b.11 as 7.2.1.1.(c) RPS Power Sources. This amendment corrects this oversight and correctly numbers the reference sited in 7.2.1.2(f).

b. The reference sited in Section 7.3.1.1.2(b) has been changed from 7.3.7 to 7.3.5.

c. The reference in 7.3.1.1.7 has been corrected from 7.3-14 to 7.3.-18.

d. Figure 7.4-2b (attached) is revised to show the automatic trasfer of the RCIC suction. Figure 7.4-2a, however, is correct as shown.

e. Table 7.5-1 has been revised in response to this question.*

f. Section 7.6.1.4.1 (b) has been revised in response to this question*

*Draft FSAR page changes attached.

which allows the control rod drive hydraulic system valve lineup to be restored to normal before the control room operator can reset the RPS logic.

(c) RPS Power Sources.

The RPS receives power from two high inertia a-c motor generator sets. A flywheel provides high inertia sufficient to maintain voltage and frequency within 5% of rated values for at least 15 seconds following a total loss of power to the drive motor. The drive motor supplies are backed up by divisional diesel generator supplies which prevent automatic scram from a loss of offsite power.

Alternate power is available to each RPS bus and is manually switched to the bus as necessary for maintenance of the RPS M-G sets. The alternate power switch is interlocked to prevent simultaneous feeding of both buses from the same source. The switch also prevents paralleling of a motor-generator set with the alternate supply.

The RPS is designed to utilize a fail-safe logic and actuation scheme. Therefore, the power supplied by the RPS M-G sets to hold RPS components energized is expendable and considered non-safety-related. The M-G sets are not Quality Class I or Seismic Class I. However, to assure that overvoltage or underfrequency does not damage safety related components within the RPS, redundant Class IE bus monitoring devices are provided to trip the RPS bus should voltage and frequency exceed predetermined limits.

7.2.1.2 Design Basis

The RPS is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Chapter 15, "Accident Analysis," identifies and evaluates events that jeopardize the fuel barrier and reactor coolant pressure boundary. The methods of assessing barrier damage and radioactive material releases, along with the methods by which abnormal events are identified, are presented in that chapter.

The following variables are monitored in order to provide protective actions to the RPS indicating the need for reactor scram.

a. Variables Monitored to Provide Protective Actions.

1. Neutron Monitoring System Trip

2. Reactor Vessel System High Pressure

f. Range of Transient, Steady State, and Environmental Conditions

The reactor protection system (RPS) 120 Vac power is provided by high inertia M-G sets. Voltage regulation is designed to respond to a step load change of 50% of rated load with an output voltage change of not more than 15%. The flywheel on each M-G set provides stored energy to maintain voltage and frequency within ±5%, for 15 seconds, preventing momentary switchyard transients from causing a scram. RPS relays and contactors will operate without failure within the range of -15% to +10% of rated voltage. Refer also to the discussion in ~~7.2.1.1(3)~~ 7.2.1.1(c).

Environmental conditions for proper operation of the RPS components are discussed in Table 3.11-1 for inside the containment and Table 3.11-2 for outside the containment.

g. Malfunctions, Accidents, and Other Unusual Events Which Could Cause Damage to Safety Systems

Unusual events are defined as malfunctions, accidents, and others which could cause damage to safety systems. Chapter 15 and Appendix 15A, "Accident Analysis", describe the following credible accidents and events; floods, storms, tornados, earthquakes, fires, LOCA, pipe break outside containment, feedwater line break, and missiles. Each of these events is discussed below for the RPS.

All components essential to the operation of the RPS are designed, fabricated, and mounted to Class IE standards. However even though the sensors initiating reactor scram which monitor turbine stop valve position and turbine governor valve fast closure are designed and purchased to Quality Class I; Seismic Class I, they are physically mounted on equipment which is not Seismic Class I/Quality Class I, and located in the turbine generator building which is not Siesmic Class I.

For this reason other diverse variables (reactor pressure and neutron flux trips) may be relied upon for reactor scram if components in the turbine generator building fail.

1. Floods

The buildings containing RPS components have been designed to meet the PMF (Probable Maximum Flood) at the site location. This ensures that the buildings will remain water tight under PMF including wind generated wave action and wave runup. For a discussion of internal flooding protection refer to 3.4.1.4.1.2, 3.4.1.5.2, and 3.6.

To prevent inadvertent actuation of the ADS two channels of logic for each ADS trip system (A & B) are used. Both channels must be activated to actuate an ADS trip system. Refer to Figure 7.3-7 for a schematic representation of the ADS initiation logic. 7.3-5

Each channel contains a single input from a drywell high pressure sensor. In addition, one channel includes two differential pressure sensor inputs monitoring reactor vessel low water level (Trip Level 3 and Trip Level 1). The second low water level trip (Trip Level 3) provides confirmation of a reactor vessel low water level condition. The other channel, in addition to drywell high pressure, includes a single reactor vessel low water level (Trip Level 1) input.

To assure that adequate makeup water is available after the vessel has been depressurized each logic channel includes a pump discharge pressure permissive signal indicating LPCI or LPCS system available for vessel water makeup. Any one of the three LPCI pumps or the LPCS pump is sufficient to permit automatic depressurization.

After receipt of the initiation signals and after a delay provided by timers, each of the two solenoid pilot air valves are energized. This allows pneumatic pressure from the accumulator to act on the air cylinder operator. Each ADS trip system timer can be reset manually to delay system initiation. If reactor vessel water level is restored by HPCS prior to the end of the time delay, ADS initiation will be prevented.

The ADS trip system A actuates the "A" solenoid pilot valve on each ADS relief valve. Similarly, the ADS trip system B actuates the "B" solenoid pilot valve on each ADS relief valve. Actuation of either solenoid pilot valve causes the ADS valve to open to provide depressurization.

Once initiated the ADS logic seals-in and can be reset by the control room operator only when either drywell pressure or vessel water level return to normal.

Two control switches (one for each trip system solenoid) are located in the main control room for each safety/relief valve associated with the ADS. Each switch controls one of the two solenoid pilot valves.

TABLE 7.5-1 (Continued)

Design Criteria	Type Readout	Number of Channels	Range	Accuracy	Location
SSW System Pump Discharge Line Pressure	Meter	2 1	0-300 psig 0-100 psig	± 1% FS ± 1% FS	CR CR
Suppression Pool Water Level	Recorder	2	-25"-0-+25"	± 1%.	CR
Main Control Room Temperature	Meter	2	50-100°F	± 2%	CR
SGTS Flow Rate	Meter	4	0-6000 CFM	± 3%	CR
CAC System Flow Rate	Meter	4	0-300 CFM	± 3%	CR
Primary Containment Hydrogen	Recorder	2	0-~~10%~~ 30% H_2	± 1%	CR
Suppression Pool Water Temperature	Recorder	2	50-400°F	± 1%	CR
Primary Containment High Level Radiation Detectors	Recorder	4	1 to 10^7 R/hr	N/A	CR
Reactor Bldg. Vent Exhaust Monitors	Recorder	4	10^{-2} to 10^2 mR/hr	N/A	CR.
Primary Containment Atmosphere Rad. Monitors.	Recorder	4	1 to 10^6 cpm	N/A	CR
Control Room Fresh Air Intake Rad Monitors.	Recordor	4	10^1 to 10^6 cpm.	N/A	CR.
CONT. INSTR. AIR N_2 Header Press.	Meter	2	0-150 psig	±2% FS	CR.

Each detector assembly consists of a fission chamber attached to a low-loss, quartz-fiber-insulated transmission cable. When coupled to the signal conditioning equipment, the detector produces a reading of full scale on the SR most sensitive ranges (8KHZ TO 16KHZ FLUX LEVEL BANDWIDTH). The detector cable is connected underneath the reactor vessel to a triple-shielded cable that is connected to the preamplifier.

The preamplifier converts current pulses to voltage pulses, modifies the voltage signal, and provides impedance matching. The preamplifier output signal is then sent to the IRM signal conditioning electronics (see Figure 7.6-8).

Each IRM channel input signal from the preamplifier can be amplified and attenuated. IRM preamplification is selected by a remote range switch that provides 10 ranges of increasing attenuation (The first 6 called low range and the last 4 called high range). As the neutron flux of the reactor core increases the signal from the fission chamber is attenuated to keep the input signal to the inverter in the same range. The output signal, which is proportional to neutron flux at the detector, is amplified and supplied to a locally mounted meter, a remote meter and recorder.

The IRM Scram Trip Functions are discussed in 7.2.1.1.B. The IRM trips are shown in Table 7.6-1.

The IRM range switches must be up-ranged or down-ranged to follow increases and decreases in power within the range of the IRM to prevent either a scram or a rod block. The IRM detectors must be inserted into the core whenever these channels are needed, and withdrawn from the core, when permitted, to prevent unnecessary burnup.

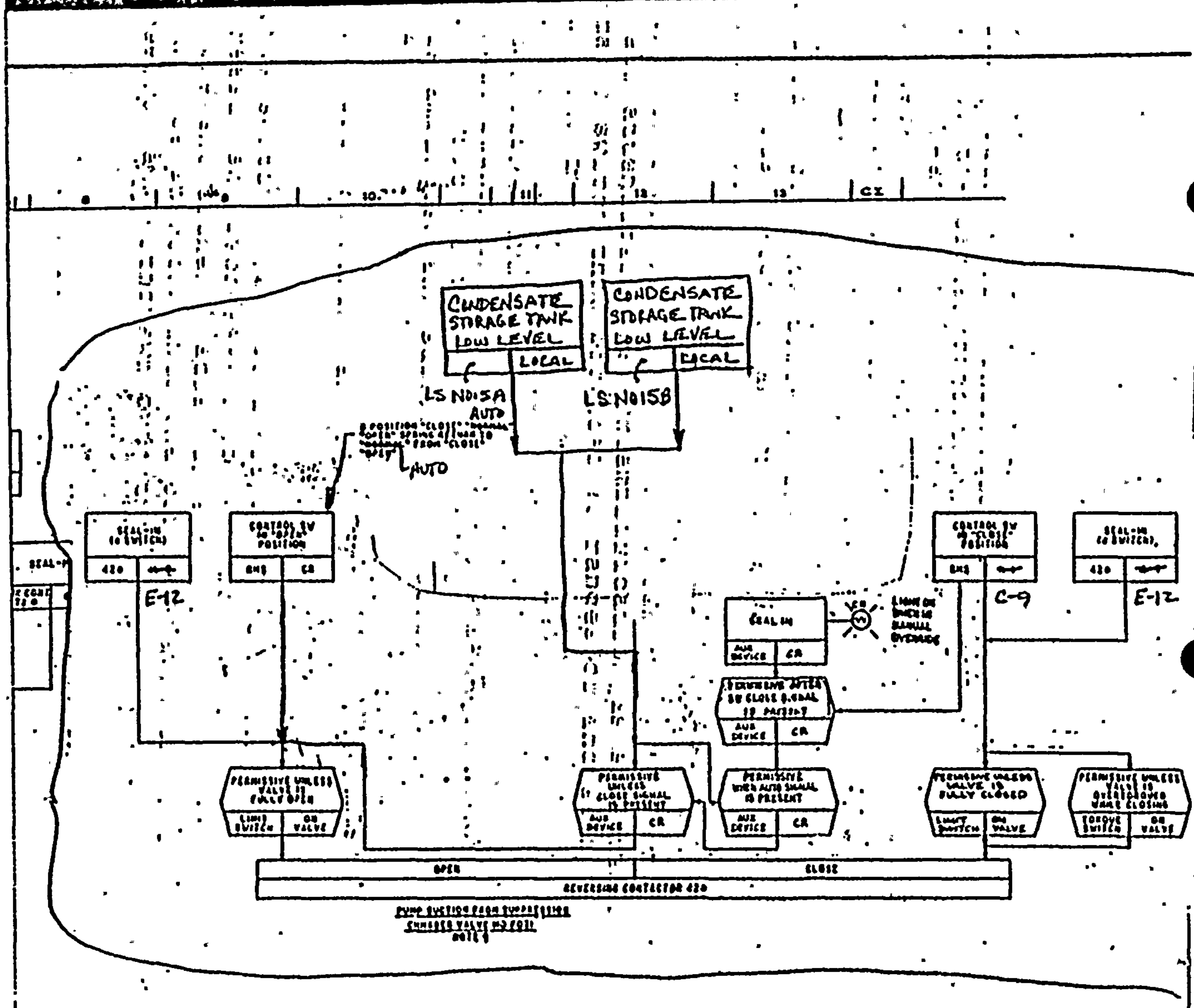
7.6.1.4.2 Local Power Range Monitor (LPRM)

a. LPRM Function

The LPRM's provide localized neutron flux detection over the full power range for input to the APRM.

b. LPRM Operation

The LPRM includes 43 detector strings having detectors located at different axial heights in the core; each detector string contains four fission chambers. Figure 7.6-3 shows the LPRM detector radial layout scheme.



CONDENSATE STORAGE TANK LOW LEVEL
LOCAL
CONDENSATE STORAGE TANK LOW LEVEL
LOCAL
LS N015A
AUTO
LS N015B
AUTO
SEAL-IN
CONTROL SW
E-12
C-9
E-12
SEAL IN
AUX DEVICE
CR
PERMISSIVE UNLESS VALVE IS FULLY OPEN
LIMIT SWITCH
ON VALVE
PERMISSIVE WHEN AUTO SIGNAL IS PRESENT
PERMISSIVE UNLESS VALVE IS FULLY CLOSED
OPEN
CLOSE
REVERSING CONTACTOR 42O

14-2b

Q. 031.115
(7.2)

The WNP-2 SER issued at the CP stage of our review in September 1971 acknowledges your commitment to include a recirculation pump trip (RPT) on receipt of a signal indicating high reactor pressure. This trip is intended to mitigate the effects of a failure to scram. Provide the details of your proposed RPT design; indentify and justify any exceptions to the requirements of the reactor protection system (RPS).

Response:

A recirculation pump trip on high reactor pressure is provided to mitigate the effects of failure to scram (ATWS condition). This ATWS RPT is designed to be non-safety related.

A modification of Appendix H to the FSAR provides a description of the equipment and function.*

As stated in the insert to the text, the ATWS RPT does not interact with the RPS, nor are any of the RPS requirements addressed by this function.

*Draft FSAR page change attached.

These functional requirements are satisfied by an elbow flow element where the pressure from the inside to the outside of the elbow is proportional to flow. Consequently, the flow nozzle in the BWR/4 design was replaced by an elbow pressure tap flow element in the recirculation pump suction line for the BWR/5 and /6 design.

H.1.2.8 Recirculation Pump Trip (RPT)

The recirculation pumps are tripped for many reasons, among which are low NPSH, some transients and electrical faults such as short circuits. Only one trip function is currently required to be safety grade, and that function is given the name RPT. The purpose of RPT is to mitigate the thermal consequences of the turbine trip and generator trip transients by tripping the recirculation pumps early in the event, producing rapid pump flow coastdown and additional core voiding, which results in a core reactivity reduction. This system is linked to the reactor protection system (RPS) such that both a scram and a pump trip occur when the turbine stop valves start to close and when turbine governor valve fast closure occurs. Both scram and RPT are bypassed at low thermal power levels.

Since only one power source is available to a BWR/4 pump motor, RPT trips the pumps completely off. The BWR/5 activates the 25% speed source (the low-frequency M-G set) when the pump has coasted down to that speed.

Insert attached

H.1.2.9 Core Flow Measurement

The core flow measurement system is unchanged from the BWR/4 design. For BWR/5 and /6, as an operating convenience, individual jet pump pressure drop signals are fed to the process computer to calibrate the system and obtain the jet pump integrity surveillance data required by the technical specifications.

H.1.2.10 Recirculation System Operation

Due to the changes described in Subsections H.1.2.2 and H.1.2.4, the startup and operation of the BWR/5 recirculation system is significantly changed from previous systems. As a result, new control interlocks were necessary to prevent significant transients, equipment damage, or unnecessary scrams. Electrical interlocks were installed between the LFMG set and the normal power supply to prevent damage to the LFMG set and on the flow control valve to prevent cavitation damage. These interlocks also protect against flow-increase transients when starting the system or transferring to the normal power supply.

Q. 031.116
(T7.3-3)
(T7.3-5)
(T7.3-7)
(T7.3-9)

The use of level switches with a range of -150 inches/0/+60 inches to initiate the automatic depressurization system (ACS), the low pressure core spray (LPCS) system and the low pressure coolant injection (LPCI) system with a setpoint of -149 inches as shown in Tables 7.3-3, 7.3-5, and 7.3-7 of the FSAR, respectively, if not a conservative design feature. A similar situation exists for the differential pressure switch on the RCIC turbine steam line where the range is given as -200 inches/0/+200 inches and the high flow trip point is indicated in Table 7.3-9 to be +198 inches. Provide justification for using these instruments whose extreme range is barely above the trip point or the setpoint. Justify the use of these ranges in these applications. Discuss the accuracy of the trip settings and how they are affected by long-term drift and by normal environmental conditions and those occuring during and after postulated accidents.

Response:

The values provided in Chapter 7 are for information only since setpoints are not finialized until the Technical specifications are completed. The actual setpoints for the parameters contained in Tables 7.3-3, 7.3-5, 7.3-7 and 7.3-9 will be shown in the Chapter 16 Technical Specifications. Notes to this effect follow each of the Tables in question.

The values shown on the tables in Chapter 7 will either be updated or deleted and reference made to the Technical Specifications once these specifications are complete.

When the Technical Specifications are issued, the LPCI/LPDS/ADS reactor vessel low water level trip set point will be approximately 181 inches.

These set points are derived through application of setpoint margins as delineated in the BWR Standard Technical Specifications taking into account instrumentation drift, loop accuracy, calibration errors, etc.

Q. 031.117
(7.3)

General Electric and other nuclear steam supply system (NSSS) vendors have reported that post-accident temperature conditions can affect reactor vessel water level instrumentation.

a. Describe the liquid level measuring systems within the WNP-2 containment which are used to initiate safety-related actions or are used to provide post-accident monitoring information. Provide a description of the type of reference leg used; i.e., an open column or a sealed reference leg.

b. Provide an evaluation of the effect of post-accident ambient temperatures on the indicated level which will relate the change in indicated level to the actual water level. This evaluation must include all other sources of error, including the effects on the water level measurements caused by varying fluid pressure and flashing of water in the reference leg to scram.

c. Provide an analysis of the effect that the potential level measurement errors in both the WNP-2 control and protection systems, discussed in Item (b.) above, can have with respect to the validity of the assumptions used in your analyses of plant transients and postulated accidents. Your response should include a review of all safety and control setpoints derived from level signals to verify that the setpoints will initiate the safety-related action required by the plant safety analyses. This review should encompass the anticipated range of ambient temperatures that may be encountered by the safety-related instrumentation, including those temperatures which could occur during and after postulated accidents. If this analysis demonstrates that the level measurement errors which could occur in the WNP-2 facility are greater than those assumed in your safety analysis, indicate the corrective action to be taken. The corrective actions which you should consider include design changes that could be made to ensure that the effects of varying containment temperatures are automatically accounted for. These measures may include setpoint changes as an acceptable corrective action for the short term. However, some form of temperature compensation or modification to eliminate or reduce temperature errors should be investigated as a long-term solution.

d. Indicate any required revisions to your emergency procedures to include specific information obtained from the review and evaluation of Items (a.), (b.) and (c.) above to ensure that the reactor operators are instructed on the potential for, and magnitude of, erroneous level signals. Provide a copy of tables, curves, or correction factors that would be applied to post-accident monitoring systems which will be used by plant operators.

Response:

An analysis by General Electric addressing water level instrumentation accuracy is presently underway. The final results of the analysis are expected to be submitted to NRC by October 1, 1981.

Q. 031.119
(7.3.1)
(6.2.6)

In Section 7.3.1.1.8 of the FSAR, you refer to Section 6.2.5 for a complete description of the WNP-2 containment atmosphere control system instruments and controls. However, Section 6.2.5.2.2 contains a reference, for part of the relevant information, to Section 7.6.2.13.8 which is not obsolete. Provide the missing information and/or correct references in Section 6.2.5.2.2.

Response:

The reference sited in Section 6.2.5.2.2 has been changed from 7.6.1.13.8 to 7.5.1.5.5.*

*Draft FSAR page change attached.

In the long term, hydrogen mixing may be provided by the containment recirculation and head area return fans (see 9.4.11). The redundant head area return fans will exhaust any potential hydrogen concentration from the head area to the upper drywell area. Redundant recirculation fans in the drywell area will provide proper mixing. Hydrogen generated within the wetwell is diluted by essentially hydrogen free effluent gas from the containment atmosphere control system recombiners. The mixture is automatically directed back to the drywell through vacuum breaker valves located high in the wetwell when wetwell pressure exceeds drywell pressure by 0.15 to 0.35 psi. The drywell suctions for the containment atmosphere control system are located in the upper drywell area as shown in Figures 6.2-32 and 6.2-33.

6.2.5.2.2 Hydrogen Concentration Monitoring System

The concentration of hydrogen is the limiting parameter. The hydrogen concentration is continuously monitored during normal operation and following the postulated LOCA, and dispalyed in the control room. If not already initiated, hydrogen mixing is initiated at a concentration of 2% by volume (see 6.2.5.2.1 above). When the H_2 concentration approaches 3.5% by volume a visual and audible alarm initiates in the control room. The hydrogen recombiner is started manually from teh main control room to limit the hydrogen concentration to less than 4% by volume. The operation of the hydrogen recombiners is independetn of the operation of the hydrogen concentration monitoring system.

The accuracy of the combustible gas analyzers, number and location of sampling points, and instrumentation are discussed in ~~7.6.1.13.8~~ 7.5.1.5.5 7.5.1.5.5

Shop tests are performed to calibrate and verify instrument accuracy against known gas composition.

Two redundant hydrogen concentration monitoring systems are provided. A single failure does not interrupt the gas analysis or alarm annunciation. Provisions are made for electrical and physical divisional separation.

6.2.5.2.3 Hydrogen Recombiner System

The concentration of hydrogen in the primary containment (drywell and suppression chamber), following a postulated loss-of-coolant accident, is controlled by the hydrogen recombiner system. Each of the two redundant recombiners has a hydrogen recombining capability that meets the criteria of Regulatory Guide 1.7 (Rev. 1 dated September 1976). The

Q. 031.120
(7.3.1)
(T7.3-28)

In Section 7.3.1.1.3 of the FSAR, provide justification for the data given in Talbe 7.3-28 which shows that zero channels indicating the main steam leakage control (MSLC) system header pressure and high flow in the MSLC are required to complete the protective function of the MSLC system.

Response:

Table 7.3-28 is incorrect as shown. The table has been amended to clarify entries and remove typographical errors.*

*Draft FSAR page change attached.

TABLE 7.3-28

CHANNELS REQUIRED FOR PROTECTIVE FUNCTION COMPLETION FOR MAIN STEAM LINE LEAKAGE CONTROL SYSTEM

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Reactor Pressure Low ~~(MSLC-PS-20)~~ INBOARD SYSTEM (MSLC-PS-8A-D)	~~5~~ 4 (1/STM LINE)	4 (1/STM LINE)
MSLC Header Press. Low (MSLC-PS-25) OUTBOARD SYSTEM	1	~~8~~ 1
MSLC Header Press. Low (MSLC-PS-70A-D) INBOARD SYSTEM	4 (1/STM LINE)	4 ~~8~~ (1/STM LINE)
MSLC High Flow INBOARD SYSTEM (MSLC-FS-3A-D)	4 (1/STM LINE)	4 ~~8~~ (1/STM LINE)
REACTOR PRESSURE LOW OUTBOARD SYSTEM (MSLC-PS-20)	1	1

Q. 031.121
(7.3.1)

Indicate in Section 7.3.1.1.11 of the FSAR, how the control room operator knows how many of the nitrogen bottles have been consumed in the event that the Class I portion of the WNP-2 containment instrument air system is in operation.

Response:

Section 7.3.1.1.11 has been expanded to clarify the method of nitrogen supply inventory control. It should be noted, that the Class I portion of the WNP-2 CIA system has been disigned with a 30 day in-place pneumatic supply and the capability for remote (outside Reactor Building) connection of an auxiliary N_2 supply. There is no direct control room indication regarding availability of N_2 bottles; however, the the operator can monitor the 30-day supply by either the counters described in the revised section 7.3.1.1.11, or as a back-up to the counters, a low header pressure alarm. The low pressure header alarm provides an indication of when the Class I N_2 supply has been nearly exhausted. See also Section 9.3.1.2.2.*,**

*Draft FSAR page changes attached.

**Remote charging connections and bottle counter not yet shown on drawings.

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Logic Diag.). Instrument specifications are listed in Tables 7.3-15 and 7.3-16. Plant Layout Drawings and Electrical Schematics are identified in 1.7. Operator Information Displays are shown in Figure 3.2-18 (HVAC React. Bldg. Flow Diag.) and Figure 7.3-19 (SGTS Control Logic Diag.).

The differential pressure is monitored by eight redundant differential pressure transmitters, four in Division 1 and four in Division 2, which measure the differential pressure from the exterior of four sides of the reactor building to the fuel pool area. The signal indicating the least differential pressure from the four differential pressure transmitters in one division is selected and is used to control the position of a damper of that division in the normal reactor building exhaust fan units or upon the initiation of the standby gas treatment system by containment isolation signals high drywell pressure, low reactor water level, or reactor building exhaust high radiation, the reactor building ventilation and pressure control system then controls reactor building pressure by controlling the standby gas treatment system fan units. (See 6.5.1).

7.3.1.1.11. Containment Instrument Air (CIA) System

a. CIA System Function

The purpose of the containment instrument air system is to provide uninterruptable divisional instrument air to essential ADS valve accumulators inside primary containment and non-safety-related instrument air supplies to other valves inside containment as shown in Figure 3.2-21. The system consists of a safety-related portion, which is comprised of two divisions, and a non-safety-related portion. During normal operation, the non-safety-related portion of the system provides control air to ADS accumulators and other valves.

In the event of failure of the non-safety-related portions of the system, which is indicated by low header pressure and detected by two redundant pressure switches, the safety-related portion automatically maintains header pressure from two nitrogen bottle sources. The non-safety-related portion of the system is isolated from the safety-related portion upon detection of failure of the non-safety-related portion. See 9.3.1.2.2.

INSERT A: (attached)

b. CIA System Operation

Schematic Arrangements of system mechanical equipment is shown in Figure 3.2-21 (CIA Flow Diag.). CIA component control logic is shown in Figure 7.3-20 (CIA Control Logic Diag.). Instrument specifications are listed in Tables 7.3-13 and

FSAR REVISION

This system consists of two 100% capacity air compressors, associated coolers, a twin tower air dryer, filters, an air receiver, valves, and piping of a leak tight design. In addition, two nitrogen gas bottle banks and associated piping are provided as a backup to the compressor supplied air for seven of the main steam relief valves which perform the ADS function.

The compressors located in the reactor building take suction from the building atmosphere through intake filter-silencers which are 98% efficient in filtration of particles as fine as five microns. The air is then discharged through an aftercooler, a prefilter, a dryer, an afterfilter and air receiver to deliver dry, clean, pressurized air to the pneumatic control systems of the following valves inside the primary containment vessel:

a. Four main steam isolation valves and their accumulators,

b. Eighteen main steam safety/relief valves and their accumulators.

The two independent nitrogen bottle bank subsystems are provided to deliver pressurized nitrogen to seven of the safety/relief valves and accumulators. These seven valves perform the ADS function, if required, during postulated LOCA conditions. These nitrogen banks ~~ensure~~ provide a 30-day supply of nitrogen for the ADS function. ~~during isolation of the compressor loop.~~ INSERT B ~~One bank of 15 bottles supplies~~ ING ~~nitrogen for three main steam safety/relief valves and accumulators, while~~ and ~~second the other bank of 19 bottles supplies~~ to the remaining ~~four main steam safety/relief valves and accumulators (see Figure 9.3-2)~~.

The nitrogen bottles are located in the railroad lock of the reactor building to facilitate access. Under normal operating conditions, the controlled leakage boundary of the Reactor Building is maintained above the railroad lock so access is available to the bottles for recharging if required. The bottles are standard, commercially available units pressurized to 2490 psig. Each bottle has a capacity of 257 SCF. However, the bottles are mounted in accordance with Seismic Category I, Quality Class I requirements. The required quantity of bottles for each bank was conservatively based on providing a 30-day supply to the ADS valves to satisfy the long term post-LOCA demand based on the following:

Insert A

Insert to Page 7.3-23:

The local stepping controller used for sequential nitrogen bottle opening is equipped with a local wheel index counter. A remote counter display is also located near the auxiliary N_2 supply connections outside of the Reactor Building, (see Section 9.3.1.2.2). The number of steps cycled is equal to the number of bottles provided. Administrative controls will be applied to monitor the wheel index counters position (which corresponds to the number of bottles consumed) and advise the control room operator of the current inventory. In addition, a low header pressure alarm is provided to alert the operator to the loss of the ADS pneumatic supply.

Insert B Page 9.3-4

In addition, two remote nitrogen connections are provided (one for each bank) to allow supplementing of the existing 30-day supply to ensure the ADS function is operable indefinitely during isolation of the compressor loop. The remote connections are located outside of the Reactor Building to ensure access during post accident conditions. The in-place nitrogen supply consists of one bank of 15 bottles supplying nitrogen for three main steam safety/relief valves and accumulators, and a second bank of 19 bottles supplying the remaining four main steam safety/relief valves and accumulators (see Figure 9.3-2).

Q. 031.122
(7.0)

Correct the discrepancy between the relay tabulation for the contacts on relay K15 as shown on GE Drawing No. 807E1735C, sheet 1A, and the actuating relay contacts on valve E51-F008, shown on GE Drawing No. 807E173TC, sheet 6.

Response:

Relay K15 contacts 304 and 506 are incorrectly shown on GE Drawing No. 807E173TC, sheet 1A, to interlock with valve E51-F064 control circuit. This drawing will be revised to show valve E51-F008 instead of E51-F064. Sheet 6 of the above drawing is correct as is.

Q. 031.123
(7.6.1)
(F7.6-1a)

Differential pressure switches E31-N007A and E31-N013A are shown on GE Drawing 807E173TC, sheet 2, as though each were one diagram with multiple contacts. However, as indicated on the contact labels, GE Drawing No. 807E173TC, sheet 3, shows two switches, both labeled E31-N007B, and two switches labeled E31-N013B and E31-N013D. Figure 7.6-1a of the FSAR shows four differential pressure switches for this system labeled dPIS N))7A, dPIS N007B, dPIS N013A, and dPIS N013B. Correct this discrepency.

Response:

Differential pressure switches (dPIS) E31-N007A and E31-N013A on GE Drawing 807E173TC, Sheet 2, are correct as shown. On Drawing 807E173TC, Sheet 3, the two switches, labeled E31-N007B, are correct and the two switches labeled E31-N013B and E31-N013D should both be labeled E31-N013B. The elementary has been corrected. The four dPIS N007A, dPIS N007B, dPIS N013B on Fig. 7.6-1a of FSAR are correct as shown.

Q. 031.124
(7.6.1)
(F7.6-1a)

Your discussion of the WNP-2 leak detection system in Section 7.6 of the FSAR is incomplete. In its present form, it is written almost exclusively on the basis of detecting leaks through their effect on air temperature, differential temperature or sump flow. Provide an expanded discussion, including the design basis, for the other leak detection methods incorporated into the WNP-2 design; e.g., the fission products monitoring system and the flow monitor on the drywell air cooler condensate line which are both shown in Figures 7.6-1a and 7.6-1b of the FSAR. Indicate the appropriate set-points for these systems in Table 7.6-7 and the minimum number of required channels in Table 7.6-8. Include these systems in the tabulation contained in Section 7.6.1.8.

Response:

There are no separte monitoring devices provided to monitor the flow from the drywell cooler condensate drains. All condensate from these coolers is drained to the Drywell Floor Drain Sump. Since this is the only condensate flowing into this sump, sump flow is equal to condensate flow from the drywell coolers. Flow from this sump is closely monitored as (attached) is revised to delete the condensate flow monitor. The desing basis for this system is provided in Section 7.6.2.4.

The following subsection will be added to Section 7.6.1.3:

7. Drywell Atmosphere Radiation Monitoring System.

The drywell atmosphere is continuously monitored for gaseous and particulate radioactivity by redundant sampling systems. In each system the sample is drawn into the sample system by its each system the sample is drawn into the sample system by its vacuum pump. Flow control is provided to insure proper sample flow. The sample flow path is from the sample point inside the primary containment, through the inlet isolation valve to the particulate monitor chamber. Here the sample is passed through a moving filter tape while allowing the noble gases to pass

through. The filter then moves across the face of a scintillation detector for analysis.

The gaseous sample, after removal of any particulate matter as described above, passes into a volume chamber where a second scintillation detector checks activity. Any activity at this point will be due to noble gases.

The sample gas then proceeds through the flow control device, vacuum pump, return line isolation valve and is discharged back into the primary containment.

Tables 7.6-7 and 7.6-8, and Section 7.6.1.8 are revised to indicate other leak detection methods.*

*Draft revised FSAR page changes are attached.

In addition, wall-mounted level switches are provided in each sump room to detect ECCS passive failures and provide annunciation in the main control room.

INSERT "A" (ATTACHED)

~~7.6.1.4 Neutron Monitoring System (NMS) - Instrumentation and Controls~~

~~The safety-related portions of the neutron monitoring system are as follows:~~

~~1. Intermediate Range Monitor (IRM)~~

~~2. Local Power Range Monitor (LPRM)~~

~~3. Average Power Range Monitor (APRM)~~

~~a. Neutron Monitoring System Function~~

~~The neutron monitoring system instrumentation and controls are designed to monitor reactor power (neutron flux) from startup through full power operation.~~

~~b. Neutron Monitoring System Operation~~

~~The neutron monitoring system uses incore detectors, either fixed (LPRM) or removable (IRM), to determine neutron flux levels.~~

~~NMS will initiate a scram when predetermined limits are exceeded and provide operator information during and after accident conditions.~~

~~The NMS component control logic is shown in Figure 7.6-6.~~

~~7.6.1.4.1 Intermediate Range Monitor (IRM)~~

~~a. IRM Function~~

~~The IRM monitors neutron flux from the upper portion of the SRM range to the lower portion of the power range (APRM) as shown in Figure 7.6-22.~~

~~b. IRM Operation~~

~~The IRM has eight channels, each of which includes one detector that can be positioned in the core by remote control. Refer to Figures 7.6-5 and 7.6-2. The detectors are inserted into the core for a reactor startup and are withdrawn after the reactor mode selector switch is placed in the RUN position.~~

INSERT "A"

~~FSAR~~ ~~REVISION~~
alter
Insert ~~for~~ Section 7.6.1.4

7. Drywell Atmosphere Radiation Monitoring System

The drywell atmosphere is continuously monitored for gaseous and particulate radioactivity by redundant sampling systems. In each system the sample is drawn into the sample system by its vacuum pump. Flow control is provided to insure proper sample flow. The sample flow path is from the sample point inside the primary containment, through the inlet isolation valve to the particulate monitor chamber. Here the sample is passed through a moving filter tape where the particulate matter is deposited on the tape while allowing the noble gases to pass through.

The filter then moves across the face of a scintillation detector for analysis. After removal of any particulate matter as described above, the gaseous sample passes into a volume chamber where a second scintillation detector checks activity. Any activity at this point will be due to noble gases.

The sample gas then proceeds through the flow control device, vacuum pump, return line isolation valve and is discharged back into the primary containment.

~~In addition, wall-mounted level switches are provided in each sump room to detect ECCS passive failures and provide annunciation in the main control room.~~

7.6.1.4 Neutron Monitoring System (NMS) - Instrumentation and Controls

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The neutron monitoring system instrumentation and controls are designed to monitor reactor power (neutron flux) from startup through full power operation.

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The neutron monitoring system uses incore detectors, either fixed (LPRM) or removable (IRM), to determine neutron flux levels.

NMS will initiate a scram when predetermined limits are exceeded and provide operator information during and after accident conditions.

The NMS component control logic is shown in Figure 7.6-6.

7.6.1.4.1 Intermediate Range Monitor (IRM)

a. IRM Function

The IRM monitors neutron flux from the upper portion of the SRM range to the lower portion of the power range (APRM) as shown in Figure 7.6-22.

b. IRM Operation

The IRM has eight channels, each of which includes one detector that can be positioned in the core by remote control. Refer to Figures 7.6-5 and 7.6-2. The detectors are inserted into the core for a reactor startup and are withdrawn after the reactor mode selector switch is placed in the RUN position.

d) RCIC steam line pressure

e) RHR area temperatures - differential and ambient

f) RHR shutdown cooling suction flow

g) RWCU area temperatures - differential and ambient

h) RWCU differential flow

i) Identified and Unidentified leakage from the drywell floor and equipment drain sumps.

J) Drywell Atmosphere Radiation Monitor

3. Neutron Monitoring System

a) IRM neutron flux

b) APRM neutron flux

4. Spent Fuel Pool Cooling and Cleanup System

a) Fuel Pool Temperature

5. Suppression Pool ~~and Chamber~~ Temperature Monitoring System

~~a) Suppression Pool and Chamber Temperature~~

The plant conditions which require protective action involving the safety-related systems discussed in 7.6 are described in Chapter 15 and Appendix 15A.

b. Location and Minimum Number of Sensors

See Chapter 16 for the minimum number of sensors required to monitor safety-related variables. The IRM and LPRM detectors are the only sensors which have spatial dependence.

c. Prudent Operational Limits

Prudent operational limits for each safety-related variable trip setting are selected to be far enough above or below normal operating levels so that a spurious safety system initiation is avoided. It is then verified by analysis that the release of radioactive materials, following postulated gross failures of the fuel or nuclear system process barrier, is kept within acceptable bounds.

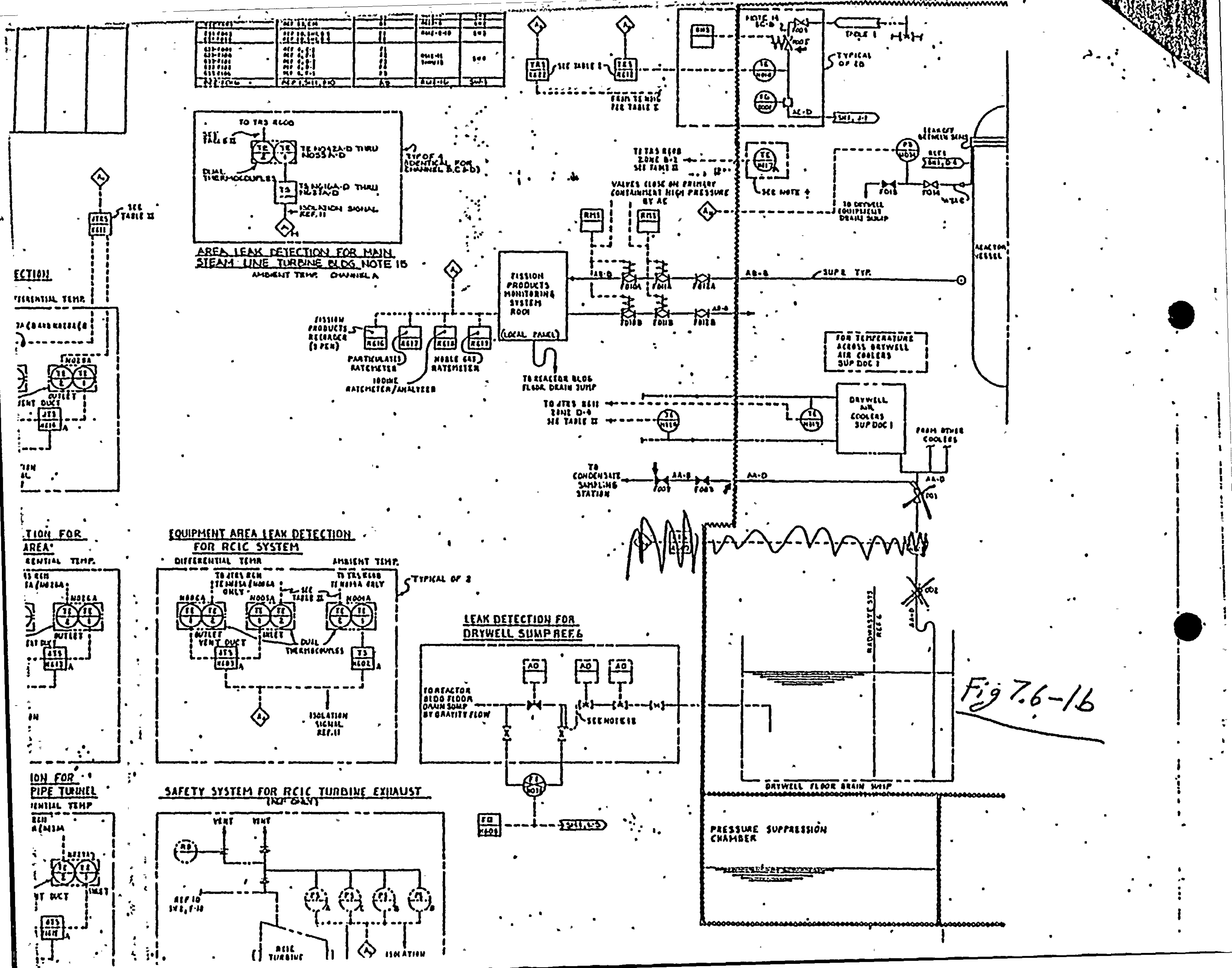
TABLE 7.6-7 (Continued)

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Main Steam Line Steam High Flow	Diff. Press. Switch (E31-N008A-D thru E31-N011A-D)	-15 to 0 to 150 psid	104 psid	-	-	-
RWCU Diff. Flow High	Diff. Flow Switch (E31-N605A,B)	-	57 gpm	-	-	-
React. Bldg. Equip. ~~Drain~~ Sump Level High DRAIN	Level Switch (EDR-LS-N014)	-	-	-	-	-
React. Bldg. Floor Drain Sumps Level High	Level Switch (FDR-LS-N006A, B) (FDR-LS-N005A, B)	-	-	-	-	-
WELL ~~Containment~~ Atmos. Radioactive Particulate Mon.	Radiation Detector (RAD-RE-12A, B)	$10-10^7$ cpm	To be determined after startup.	—	—	—
~~CTMT~~ Atmos. Rad. Gaseous Monitor	Radiation Detector (RAD-RE-12A, B)	$10-10^7$ cpm	To be determined after startup	—	—	—

TABLE 7.6-8 (continued)

INSTRUMENT CHANNEL	CHANNELS PROVIDED	MINIMUM CHANNELS REQUIRED
Main Steam Line Pipe Routing Area in Turbine Gen. Bldg. Ambient Temp. High	48	2
Main Steam Line Steam High Flow	16	2
RWCU Diff. Flow High	2	1
Reactor Bldg. Equip. Drain Sump Level High	1	1
Reactor Bldg. Floor Drain Sumps Level High	4	1
~~Containment~~ Atmosphere Radioactive Particulate Monitors	2	1
~~Containment~~ Atmosphere Radioactive Gaseous Monitors.	2	1

DRYWELL



AREA LEAK DETECTION FOR MAIN STEAM LINE TURBINE BLDG NOTE 15
AMBIENT TEMP. CHANNEL A
DUAL THERMOCOUPLES
ISOLATION SIGNAL REF.11
FISSION PRODUCTS MONITORING SYSTEM R001
(LOCAL PANEL)
FISSION PRODUCTS RECORDER (3 PEN)
PARTICULATES RATEMETER
NOBLE GAS RATEMETER
IODINE RATEMETER/ANALYZER
TO REACTOR BLDG FLOOR DRAIN SUMP
VALVES CLOSE ON PRIMARY CONTAINMENT HIGH PRESSURE BY AC
SEE NOTE 4
TO DRYWELL EQUIPMENT DRAIN SUMP
REACTOR VESSEL
FOR TEMPERATURE ACROSS DRYWELL AIR COOLERS SUP DOC 1
DRYWELL AIR COOLERS SUP DOC 1
FROM OTHER COOLERS
TO CONDENSATE SAMPLING STATION
EQUIPMENT AREA LEAK DETECTION FOR RCIC SYSTEM
DIFFERENTIAL TEMP.
AMBIENT TEMP.
TYPICAL OF 2
OUTLET VENT DUCT
INLET
DUAL THERMOCOUPLES
ISOLATION SIGNAL REF.11
LEAK DETECTION FOR DRYWELL SUMP REF.6
TO REACTOR BLDG FLOOR DRAIN SUMP BY GRAVITY FLOW
SEE NOTE 18
DRYWELL FLOOR DRAIN SUMP
PRESSURE SUPPRESSION CHAMBER
SAFETY SYSTEM FOR RCIC TURBINE EXHAUST
VENT
RCIC TURBINE
ISOLATION

Fig 7.6-1b

Q. 031.125
(7.6.1)
(F3.2-8)

The suppression Chamber temperature is listed in Section 7.6.1.8 of the FSAR as one of the "Variables Monitored to Provide Protection Action". However, the measuring devices and their associated instrumentation are not discussed in Section 7.6.1.7 nor are they shown in Figure 3.2.8. Revise the text and figures to provide this information.

Response:

The reference to chamber temperature monitoring has been deleted in Section 7.6.1.7. Although WNP-2 will have provisions for monitoring this parameter, it is not a variable monitored to provide protective action.*

*Draft FSAR revised page change attached.

(031,125)

d) RCIC steam line pressure

e) RHR area temperatures - differential and ambient

f) RHR shutdown cooling suction flow

g) RWCU area temperatures - differential and ambient

h) RWCU differential flow

i) Identified and Unidentified leakage from the drywell floor and equipment drain sumps.

3. Neutron Monitoring System

a) IRM neutron flux

b) APRM neutron flux

4. Spent Fuel Pool Cooling and Cleanup System

a) Fuel Pool Temperature

5. Suppression Pool ~~and Chamber~~ Temperature Monitoring System

a) ~~Suppression Pool and Chamber Temperature~~

The plant conditions which require protective action involving the safety-related systems discussed in 7.6 are described in Chapter 15 and Appendix 15A.

b. Location and Minimum Number of Sensors

See Chapter 16 for the minimum number of sensors required to monitor safety-related variables. The IRM and LPRM detectors are the only sensors which have spatial dependence.

c. Prudent Operational Limits

Prudent operational limits for each safety-related variable trip setting are selected to be far enough above or below normal operating levels so that a spurious safety system initiation is avoided. It is then verified by analysis that the release of radioactive materials, following postulated gross failures of the fuel or nuclear system process barrier, is kept within acceptable bounds.

Q. 031.126
(7.7.1.13)
(F7.7-3a)
(F7.7-3c)

The discussion of the refueling interlocks in Section 7.6.1.1 is incomplete as follows:

a) Section 7.7.1.13 describes the "all rods in" circuit as "two channel". Is there a separate reed switch at each position for each of these channels, or are both channels activated from the same switch?

b) Even though refueling operations are the means by which the core reactivity is restored, no mention is made of any interlocks related to the monitoring of core reactivity and no reference is made to the mechanisms used to ensure that the fuel used is of the proper enrichment. Justify the omission of a flux related interlock for the motion of the refueling hoists, similar to the one used for generating the rod withdrawl block shown in Figure 7.7-3c.

Response:

a) Section 7.7.1.13 is incorrect in describing the "all rods in" circuit as "two channel". The text for pages 7.7-44 and 7.7-45 has been revised.*

b) None of the refueling interlocks monitor core reactivity. The interlocks are designed to prevent the combination of refueling floor activities and rod motions which could allow a reactivity transient to occur. Plant refueling procedures control the location and placement of bundles of varying enrichment.

A flux related interlock for the motion of the refueling grapple is not necessary for the following reasons:

1) Refueling procedures require the control rods be fully inserted before fuel is loaded into a control cell. This assures Keff < 1 will be maintained.

2) As discussed in 7.7.1.7, during refueling, the SRM subsystem of the NMS monitors the very low level neutrom flux in the subcritical core and provides a scram ini-

tiating function if limits are exceeded.

*Draft FSAR page changes attached.

b. LPDS Operation

The loose parts detection sensors are mounted on the exterior of the primary coolant system and located at natural collection points where loose parts will most likely impact. During startup and normal plant operation the LPDS is on line to provide visual and audio information to the operator. Also, this data is recorded on magnetic tape for detailed analysis at a later date and to provide a startup base line signature of various pumps and valves to be used for comparative analysis. LPDS schematic arrangements are shown in Figure 7.7-15 (LPDS P&ID).

7.7.1.13 Refueling Interlocks - Instrumentation and Controls

a. Refueling Interlocks Function

The purpose of the refueling interlocks is to restrict the movement of control rods and the operation of refueling equipment. This reinforces operational procedures that prevent the reactor from becoming critical during refueling operations.

b. Refueling Interlocks Operation

The refueling interlocks circuitry senses the condition of the refueling equipment and the control rods to prevent the movement of the refueling equipment or withdrawal of control rods (rod block). Redundant circuitry is provided to sense the following conditions:

1. All rods inserted
2. Refueling platform positioned near or over the core
3. Refueling platform hoists fuel-loaded (fuel grapple, frame-mounted hoist, trolley-mounted hoist)
4. Service platform hoist fuel-loaded, and
5. Reactor Mode Switch in "Refuel" position.

The indicated conditions are combined in logic circuits to satisfy all restrictions on refueling equipment operations and control rod movement (Figure 7.7-3). ~~A two channel circuit indicates that all rods are in.~~ The rod-in condition for each rod is established by the closure of a magnetically operated reed switch in the rod position indicator probe. The rod-in

Loss of "all rods in" signal will remove grapple control power, if the refueling platform is over the core.

switch must be closed for each rod before the all-rods-in signal is generated. ~~Both channels must indicate "all-rods-in" to allow refueling equipment to be used.~~

During refueling operations, no more than one control rod is permitted to be withdrawn; this is enforced by a redundant logic circuit that uses the all-rods-in signal and a rod selection signal from the reactor manual control system to prevent the selection of a second rod for movement with any other rod not fully inserted. Control rod withdrawal is prevented by comparison between the A and B portions of the RMCS for rod position with a subsequent rod withdrawal block if necessary. The simultaneous selection of two control rods is prevented by the interconnection arrangement of the select pushbuttons. With the mode switch in the REFUEL position, the circuitry prevents the withdrawal of more than one control rod and the movement of the loaded refueling platform over the core with any control rod withdrawn.

Operation of refueling equipment is prevented by interrupting the power supply to the equipment. The refueling platform is provided with two mechanical switches attached to the platform, which are tripped open by a long, stationary ramp mounted adjacent to the platform rail. The switches open before the platform or any of its hoists are physically located over the reactor vessel to indicate the approach of the platform toward its position over the core.

Load cell readout is provided for all hoists. Indicators display given hoist loads directly to the operator. Load sensing is by hydraulic load cells that use demineralized water as the operating fluid. Associated interlock and load functions are performed by pressure switches that sense the pressure generated by the hydraulic load cells.

The three hoists on the refueling platform and the hoist on the service platform are provided with switches that open when the hoists are fuel loaded. The switches open at a load weight that is lighter than that of a single fuel assembly. This indicates when fuel is loaded on any hoist.

A bypass for the service platform hoist load interlock is provided. When the service platform is no longer needed, its power plug is removed. This de-energizes the power supply to the hoist. The platform can then be moved away from the core.

De-energizing the hoist power supply opens the hoist load switches and gives a false indication that the hoist is loaded. This indication prevents control rod withdrawal with the mode switch in the STARTUP or REFUEL positions. A bypass

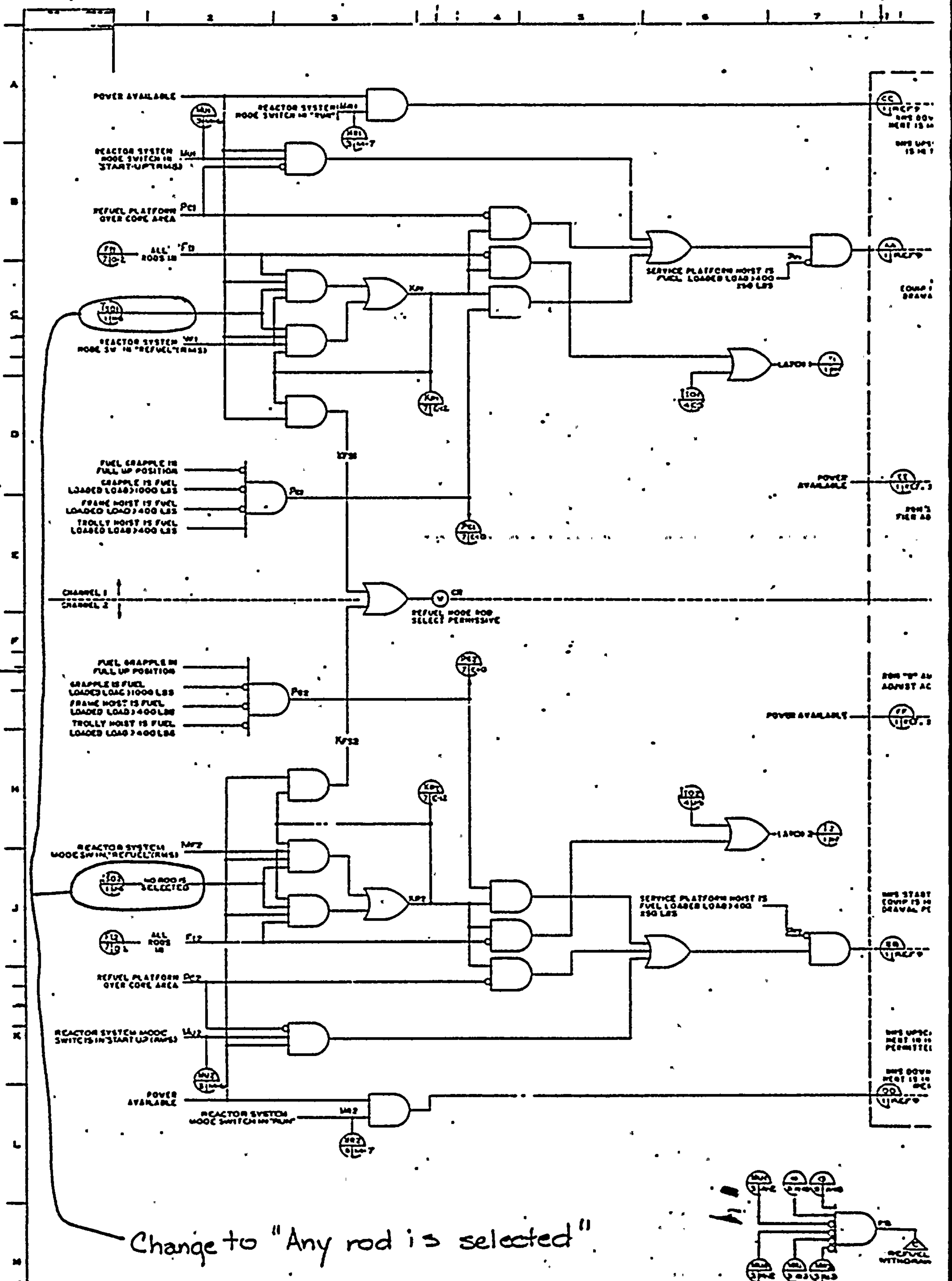
Q. 031.127
(F7.7-3c)

The title for signal Iso_2 at location J1 and for signal Iso_1 at location C1 (shown in Figure 7.7-3c) should be "any rod selected," rather than "no rod selected." Correct this discrepancy.

Response:

The WNP-2 FSAR Figure 7.7-3c has been changed.*

*Draft FSAR page change attached.



Q31.127-2

FIGURE 7.7-3C

Q. 031.129
(7.4.1.4)

The discussion of the remote shutdown capability procedure is incomplete. Describe how the operator determines the proper positions of the control switches on the remote transfer panel before making the transfer from "normal" to "emergency." Analyze the effect on plant safety of operating any of the transfer switches with its associated control switch in the "wrong" position.

Response:

Access to and operation of the remote shudown panel, including proer switch positioning, are administratively and procedurally controlled. A configuration checklist instructs the operator of the proper "standby" position of all switches on the remote panel. Because of the procedural controls invoked, all control switches will be in their "proper" position prior to operating any of the transfer switches. Additionally, procedures followed by the operator verify and maintain proper swithc lineup as he initiates action from the panel.

The Remote Shutdown System has been provided for a specific event during normal plant conditions. There is no design requirement for this system to consider the effects of single failures, including operator error, on plant safety after the occurrence of the specific event - loss of control room.

Q. 031.130
(7.5)

Classify the "Bypassed and Inoperable Status Indication" panel as a safety-related display instrument; describe and discuss this panel in Section 7.5 of the FSAR. Provide an analysis of the conformance of this panel to the applicable criteria (i.e., either the General Design Criteria of 10 CFR 50, regulatory guides, or IEEE Standards). In particular, we are concerned that the panel and its associated connections to the various safety-related systems will satisfy the required criteria for separation. Additionally, this panel and its inter-connections are not shown in the electical, instrument, and control drawings referenced in Section 1.7 of the FSAR. Provide the information requested above and provide information in Section 1.7 on how you comply with the guidelines in Regulatory Guide 1.47.

Response:

WNP-2 is in the process of completing the design implementation for Regulatory Guide 1.47. When the design is complete, a drawing set will be forwarded which detail all electrical interconnections. Representative drawing sets are expected to be ready for submittal before or during December 1981. There will be no "Bypassed and Inoperable Status Indication Panel" per se. Safety system bypasses and inoperable conditions will be indentified at the system level by annunciator to indicate the condition which has caused the safety related system to be out of service. The indications are located with the associated system controls and are thus divisionlized to match that of the associated system division thus preserving separation requirements.

Section 7.1.2.4e and Appendix C.2 and C.3, "Conformance to Regulatory Guide 1.47 (1973)", have been modified in accordance with the prediding paragraph.*

As compliance with Regulatory Guide 1.47 is described in FSAR Section 7.1, it need not be repeated in Section 7.5. Additionally, since the implementation of the Guide requirements produces a design which actually becomes part of each individual safety system, no additional analysis (other than that provided already for each system in FSAR Chapter 7.0) is required.

*Draft FSAR page changes attached.

REVISION

c. Conformance to Regulatory Guide 1.30 (8/11/72)

The quality assurance requirements of IEEE 336-1971 (see discussion above) are applicable during the plant design and construction phases and will also be implemented as an operational QA program during plant operation in response to Regulatory Guide 1.30. The specific requirements of Regulatory Guide 1.30 are met as discussed in Chapter 17.

d. Conformance to Regulatory Guide 1.40 (3/16/73)

There are no safety-related continuous duty motors installed inside the primary containment.

e. Conformance to Regulatory Guide 1.47 (1973)

Each safety-related system described in 7.2, 7.3, 7.4, and 7.6 is provided with an automatically or operator initiated system level bypass and inoperability annunciator.

The system level annunciators are ~~grouped together and~~ located ~~on a panel near the control room operator's console.~~ WITH THE ASSOCIATED SYSTEM CONTROLS AND INDICATIONS ON MAIN CONTROL ROOM PANELS. In addition to system level annunciation, component and channel level annunciators are provided on other panels either in the control room near system controls or locally near affected equipment, to indicate the cause of the system bypass or inoperability.

A switch is provided for manual actuation of each system level annunciator to allow display of those bypass or inoperable conditions which are not automatically indicated.

Typically, the following bypasses or inoperabilities cause actuation of system level (and component level) annunciation for the affected system:

1. Pump motor breaker not in operate position;
2. Loss of pump motor control power;
3. Loss of motor operated valve control power/motive power;
4. Logic power failure;
5. Logic in test;
6. ~~System lineup improper;~~ POSITION OF REMOTE MANUAL VALVES WHICH DO NOT RECEIVE AUTOMATIC ALIGNMENT SIGNALS;

Regulatory Guide 1.47, Rev. 0, May 1973

Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems.

Regulatory Guide Intent:

This guide describes an acceptable method of complying with the requirements of IEEE Standard 279-1971 and Appendix B to 10CFR50.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analyis, design and/or equipment utilized on this facility is in compliance with the intent of the regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

~~The system of bypass indication is designed to satisfy~~ the requirement of IEEE 279-1971, Paragraph 4.13 and Regulatory Guide 1.47 and is discussed for each safety-related system, as applicable, under 7.1. The design of the bypass indication system allows testing during normal operation and is used to supplement administrative procedures by providing indications of safety systems status.

The bypass indication system is designed and installed in a manner which precludes the possibility of adverse effects on the plant safety system. The bypass indication system is electrically isolated from the protection circuits such that the failure or bypass of a protective function is not a credible consequence of failures in the bypass indication system and the bypass indication system cannot reduce the independence between redundant safety systems.

Capability exists on a limited basis for status indication ~~at the safety-related equipment subsystem level~~

See attached insert

~~General Compliance or Alternate Approach Assessment: (Cont'd.)~~

~~within the NSSS scope of supply.~~

~~See C.3.0 for further discussion of compliance.~~

Specific Evaluation Reference:

Refer to 7.1.2.4.

Similar Application Reference:

Similar application was utilized on Zimmer and LaSalle.

~~This guide describes an acceptable method of complying with the requirements of IEEE Standard 279-1971 and Appendix B to 10CFR50.~~

~~Application Assessment:~~

~~Assessed capability in design.~~

~~Compliance or Alternate Approach Statement:~~

~~Identified NSSS scope of supply analyis, design and/or equipment utilized on this facility is in compliance with the intent of the regulatory guide through the incorporation of the alternate approach cited.~~

~~General Compliance or Alternate Approach Assessment:~~

Insert

Each safety-related system described in 7.2, 7.3, 7.4, and 7.6 is provided with an automatically or operator initiated system level bypass and inoperability annunciator.

The system level annunciators ~~are grouped together and~~ located ~~on a panel near the control room operator's console~~ with the associated system controls and indications on main control room panels.

In addition to system level annunciation, component and channel level annunciators are provided on other panels either in the control room near system controls or locally near affected equipment, to indicate the cause of the system bypass or inoperability.

A switch is provided for manual actuation of each system level annunciator to allow display of those bypass or inoperable conditions which are not automatically indicated.

Typically, the following bypasses or inoperabilities cause actuation of system level (and component level) annunciation for the affected system:

1. Pump motor breaker not in operate position;
2. Loss of pump motor control power;
3. Loss of motor operated valve control power/motive power;
4. Logic power failure;
5. Logic in test;
6. ~~System lineup improper;~~ Position of remote manual valves which do not receive automatic alignment signals;
7. Bypass or test switches actuated.

Auxiliary supporting system inoperability or bypass resulting in the loss of other safety-related systems will cause actuation of system level annunciators for the auxiliary supporting system as well as those safety-related systems affected.

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Regulatory Guide 1.47, Rev. 0, May 1973

Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

Compliance or Alternate Approach Statement:

WNP-2 complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

Each safety-related system described in 7.2, 7.3, 7.4, and 7.6 is provided with an automatically or operator initiated system level bypass and inoperability annunciator.

The system level annunciators are ~~grouped together and~~ located ~~on a panel near the control room operator's console.~~ with the associated system controls and indications on main control room panels.

In addition to system level annunciation, component and channel level annunciators are provided on other panels either in the control room near system controls or locally near affected equipment, to indicate the cause of the system bypass or inoperability.

A switch is provided for manual actuation of each system level annunciator to allow display of those bypass or inoperable conditions which are not automatically indicated.

Typically, the following bypasses or inoperabilities cause actuation of system level (and component level) annunciation for the affected system:

1. Pump motor breaker not in operate position;
2. Loss of pump motor control power;
3. Loss of motor operated valve control power/motive power;
4. Logic power failure;
5. Logic in test;
6. ~~System lineup improper;~~ Position of remote manual valves which do not receive automatic alignment signals;
7. Bypass ~~or~~ test switches actuated.

Auxiliary supporting system inoperability or bypass resulting in the loss of other safety-related systems will cause actuation of system level annunciators for the auxiliary supporting system as well as those safety-related systems affected.

Q. 031.131
(7.6)
(7.7)

It has been our position in previous reviews that the refueling interlocks are safety-related since the purpose of the interlocks is to prevent a reactivity accident during refueling operations. Accordingly, provide in Section 7.6 of the FSAR, a discussion of this system with respect to its safety-related function. Provide analyses of its conformance to the applicable General Design Criteria, Regualatory Guides, and IEEE Standards.

Response:

This question has been answered previously in response to Question 211.085. The refueling interlocks are non-safety-related equipment. This position has been accepted by the NRC in the Susquehanna SER.[1] The WNP-2 desing for refueling interlocks is vertually identical to that of Susquehanna. In addition, FSAR Section 7.7.1.13(b) has been revised to discuss the Safety-related backup logic to the refueling interlocks.*

*Draft FSAR page change attached.

Reference:

1. U.S. Nuclear Regulatory Commission NUREG-0776, "Safety Evaluation Report Related to the Operation of Susquehanna Steam Electric Station, Units 1 and 2", Dockets 50-387 and 50-388, April 1981, Section 7.7.2.

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switch must be closed for each rod before the all-rods-in signal is generated. Both channels must indicate "all-rods-in" to allow refueling equipment to be used.

During refueling operations, no more than one control rod is permitted to be withdrawn; this is enforced by a redundant logic circuit that uses the all-rods-in signal and a rod selection signal from the reactor manual control system to prevent the selection of a second rod for movement with any other rod not fully inserted. Control rod withdrawal is prevented by comparison between the A and B portions of the RMCS for rod position with a subsequent rod withdrawal block if necessary. The simultaneous selection of two control rods is prevented by the interconnection arrangement of the select pushbuttons. With the mode switch in the REFUEL position, the circuitry prevents the withdrawal of more than one control rod and the movement of the loaded refueling platform over the core with any control rod withdrawn.

INSERT

Operation of refueling equipment is prevented by interrupting the power supply to the equipment. The refueling platform is provided with two mechanical switches attached to the platform, which are tripped open by a long, stationary ramp mounted adjacent to the platform rail. The switches open before the platform or any of its hoists are physically located over the reactor vessel to indicate the approach of the platform toward its position over the core.

Load cell readout is provided for all hoists. Indicators display given hoist loads directly to the operator. Load sensing is by hydraulic load cells that use demineralized water as the operating fluid. Associated interlock and load functions are performed by pressure switches that sense the pressure generated by the hydraulic load cells.

The three hoists on the refueling platform and the hoist on the service platform are provided with switches that open when the hoists are fuel loaded. The switches open at a load weight that is lighter than that of a single fuel assembly. This indicates when fuel is loaded on any hoist.

A bypass for the service platform hoist load interlock is provided. When the service platform is no longer needed, its power plug is removed. This de-energizes the power supply to the hoist. The platform can then be moved away from the core.

De-energizing the hoist power supply opens the hoist load switches and gives a false indication that the hoist is loaded. This indication prevents control rod withdrawal with the mode switch in the STARTUP or REFUEL positions. A bypass

INSERT

IN ADDITION TO THE (AND AS A SAFETY RELATED BACKUP) ROD BLOCK INTERLOCKS PROVIDED BY THE NON-SAFETY RELATED REACTOR MANUAL CONTROL SYSTEM, THE SOURCE RANGE MONITORS PROVIDE A ROD SCRAM SIGNAL DURING REFUELING WHEN NEUTRON FLUX EXCEEDS A PRESET FLUX LEVEL. THE SCRAM SIGNAL WILL PROVIDE A CONTROL ROOM ALARM AND WILL INSERT ANY CONTROL ROD THAT IS WITHDRAWN. THIS LOGIC IS REMOVED FROM THE SCRAM CIRCUITRY FOLLOWING COMPLETION OF REFUELING OPERATIONS.

Q. 031.132
(7.6.1)

Provide justification in Section 7.6.2.1 of the FSAR for your statement which implies that each of the shutdown cooling suction valves, the head spray valve, and the discharge valve has "redundant and diverse interlocks to prevent the valves from being opened when the primary system pressure is above the subsystem design pressure." Identify the redundant to the appropriate drawing numbers. Additionally, identify the diverse interlocks for each valve.

Response:

Section 7.6.1.2b requires clarification. The last two paragraphs have been replaced with:*

> "The shutdown cooling suction valves, injection valve (i.e., discharge valve), and the head spray valve are provided with interlocks which prevent the valves from being opened when the primary system pressure is above the subsystem design pressure. The interlocks for the valves inside the primary containment are redundant and diverse from the interlocks to the valves outside the primary containment to assure that at least one of two series valves will always isolate when required. Diversity is provided by selecting pressure sensors from two different manufacturers."

The redundant and diverse pressure switches are B35-PS No. 18A and B shown in zones H12 and G5 of FSAR Figure 3.2-3, and also in zone J5 of FSAR Figure 7.3-14C.

*Draft FSAR page changes attached.

The following high pressure/low pressure interlock equipment is provided:

Interlocked Process Line	Type	Valve	Parameter Sensed	Purpose
RHRS Shut-down ~~Supply~~ Suction	MO MO	F009 F008	Reactor pressure	Prevents valve opening until reactor pressure is below system design pressure
RHRS Shut-down ~~Return~~ Injection	MO	F053	Reactor Pressure	Prevents valve opening until reactor pressure is below system design pressure
RHRS Head Spray	MO	F023	Reactor Pressure	Prevents valve opening until reactor pressure is below system design pressure
~~LPCI~~	~~MO~~	~~F042~~	~~Differential pressure across valve~~	~~Does not allow valve to open until differential pressure is low~~
~~LPCS~~	~~MO~~	~~F005~~	~~Differential pressure across valve~~	~~Does not allow valve to open until differential pressure is low~~

Insert Attach

~~The shutdown cooling suction valves, head spray valve, and discharge valve have redundant and diverse interlocks to prevent the valves from being opened when the primary system pressure is above the subsystem design pressure. These valves also receive a signal to close when reactor pressure is above system pressure.~~

~~The LPCI and LPCS discharge valves MO F042 and MO F005 are prevented from opening until differential pressure across the valves is low enough to prevent system overpressurization.~~

7.6.1.3 Leak Detection System - Instrumentation and Controls

The safety-related portions of the Leak Detection System are as follows:

1. Main Steam Line Leak Detection

2. RCIC System Leak Detection

INSERT:

The shutdown cooling suction valves, injection valve (i.e., discharge valve), and the head spray valve are provided with interlocks which prevent the valves from being opened when the primary system pressure is above the subsystem design pressure. The interlocks for the valves inside the primary containment are redundant and diverse from the interlocks to the valves outside the primary containment to assure that at least one of two series valves will always isolate when required. Diversity is provided by selecting pressure sensors from two different manufacturers.

Q. 031.133
(7.7.2)

Your analysis in Section 7.7.2 of the FSAR is incomplete. Provide a discussion of how you conform to the applicable General Design Criteria, Regulatory Guides, and IEEE Standards for the systems described in Section 7.7.

Response:

The systems described in FSAR Section 7.7 are non-safety related and therefore, are not required to, nor do they necessarily confrom to the requirements of the General Design Criteria, Regulatory Guides, and IEEE Standards. If portions of the systems described in 7.7 are required to meet safety related criteria, those portions and their compliance with General Design Criteria, Regulatory Guides, and IEEE Standards are described and analyzed elsewhere in the FSAR as appropriate.

Q. 031.134
(7.7)

It is our position that the operation of the safety-relief valves in the relief mode is important to the normal operation of the plant. Accordingly, provide a discussion and analysis of the instrumentation and control system for these valves in Section 7.7 of the FSAR.

Response:

An SRV writeup has been included in 7.7.*

*Draft page changes attached.

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7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

7.7.1 DESCRIPTION

Section 7.7 describes instrumentation and controls of major plant control systems whose functions are not essential for the safety of the plant. This section also describes instrumentation and controls, not essential for the safety of the plant, which are not discussed in any other FSAR section. The systems include:

1. Reactor Vessel Instrumentation
2. Reactor Manual Control System (RMCS)
3. Recirculation Flow Control System
4. Feedwater Control System
5. Pressure Regulator and Turbine - Generator System
6. Neutron Monitoring System (TIP, SRM, RBM)
7. Process Computer System and Rod Worth Minimizer Function (RWM)
8. Rod Sequence Control System (RSCS)
9. Loose Parts Detection System (LPDS)
10. Refueling Interlocks
11. SAFETY RELIEF VALVE- RELIEF FUNCTION

Refer to Tables 7.7-1 and 7.7-2 for system design and supply responsibility and similarity to licensed reactors, respectively.

7.7.1.1 Reactor Vessel - Instrumentation

Figure 7.3-9 (Nuclear Boiler System P&ID) shows the arrangements of the sensors, and sensing equipment used to monitor reactor vessel conditions.

a. System Function

The purpose of the reactor vessel instrumentation is to monitor key reactor vessel variables to provide the operator with information during normal plant operation, startup and shutdown.

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plug allows control rod movement in this situation. The bypass plug is physically arranged to prevent the connection of the service platform power plug unless the bypass plug is removed.

The rod block interlocks and refueling platform interlocks provide two independent levels of interlock action. The interlocks which restrict operation of the platform hoist and grapple provide a third level of interlock action since they would be required only after a failure of a rod block and refueling platform interlock.

In the refueling mode, the control room operator has an indicator light for "Refueling Mode Select Permissive" whenever all control rods are fully inserted. He can compare this indication with control rod position data from the computer as well as control rod in-out status on the full core status display. Whenever a control rod withdrawal block situtation occurs, the operator receives annunciation and computer logs of the rod block. The operator can compare these outputs with the status of the variable providing the rod block condition. Both channels of the control rod withdrawal interlocks must agree that permissive conditions exist in order to move control rods; otherwise, a control rod withdrawal block occurs. Failure of one channel may initiate a rod withdrawal block, and will not prevent application of a valid control rod withdrawal block from the remaining operable channel (see Table 7.7-3).

In terms of refueling platform interlocks, the platform operator has analog type readout indicators for the platform x-y position relative to the reactor core.

The position of the grapple is shown in a digital indicator immediately below the platform position indicators. Analog load cell indications of hoist loads are given for each hoist by locally mounted indicators. Individual pushbutton and rotary control switches are provided for local control of the platform and its hoists. The platform operator can immediately determine whether the platform and hoists are responding to his local instructions, and can, in conjunction with the control room operator, verify proper operation of each of the three categories of interlocks listed previously.

7.7.1.~~14~~ 15 Design Differences

Refer to Table 7.7-2 for a list of Instrumentation and Control system designs and their similarity to designs of other nuclear power plants.

INSERT FROM FOLLOWING PAGE.

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7.7.1.14 Safety Relief Valves (SRV) - Relief Function

A. SRV Function

The relief function of the SRV's is to relieve high pressure conditions in the nuclear system that could lead to the failure of the Reactor Coolant Pressure Boundary. The system activates the safety relief valves to vent steam to the suppression pool and reduce reactor pressure. See Section 5.2.2 for further details. Also, see Section 7.3.1.1.1.2 for the ADS function of selected SRV's.

B. SRV Operation

Schematic arrangement of system mechanical equipment is shown in Figure 7.3-9. The SRV component control logic is shown in Figure 7.3-10. Instrument location drawings and elementary diagrams are identified in Section 1.7.

The relief function of the SRV's is initiated by pressure switches, one per relief valve. These pressure switches are set to energize the relief valve solenoids in five groups at five respective trip settings. The relief valves will open in groups, the lowest setpoint group first, followed by groups of SRV's at progressively higher setpoints. This feature automatically adjusts the relief capacity to the magnitude of the over pressure condition. The reclose pressure setpoint (reset) for any group is separately adjusted, and adequate deadband is provided to eliminate rapid open/close operation, and minimize system stresses.

To manually open each SRV, remote manual switches are installed in the control room. Lights on the control room panel indicate when the ~~solenoid-operated pilot valves are energized to open a~~ safety relief valves ARE OPEN. This monitoring is in accordance with NUREG 0737, item II, D.3, (See Appendix B of the FSAR).

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TABLE 7.7-1

DESIGN AND SUPPLY RESPONSIBILITY OF PLANT CONTROL SYSTEMS

	GE DESIGN	GE SUPPLY	B&R DESIGN	OTHERS SUPPLY
Reactor Vessel Instrumentation	X	X		
Reactor Manual Control System	X	X		
Recirculation Flow Control System	X	X		
Feedwater Control System	X	X	X	X
Press. Regulator & Turbine Generator System			X	X
Neutron Monitoring System				
SRM	X	X		
RBM	X	X		
TIP	X	X		
Process Comp. & RWM	X	X		
Rod Sequence Control System	X	X		
Loose Parts Detection System			X	X
Refueling Interlocks	X	X		
Safety Relief Valve	X	X		

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TABLE 7.7-2

SIMILARITY TO LICENSED REACTORS

Instrumentation and Controls (System)	Plants Applying for or Having Construction Permit or Operating License	Similarity of Design
(1) Neutron Monitoring System (TIP, SRM, RBM)	LaSalle	Identical
(2) Refueling Interlocks	LaSalle	Identical
(3) Reactor Manual Control System	Zimmer-1	Identical
(4) Reactor Vessel - Instrumentation	Zimmer-1	Identical
(5) Recirculation Flow-Control System	Zimmer-1	Identical
(6) Feedwater Control System	Zimmer-1	Identical
(7) Pressure Regulator and Turbine-Generator System	Zimmer-1	Identical
(8) Rod Sequence Control System	Zimmer-1	Identical
(9) Refueling Interlocks	Zimmer-1	Identical
(10) Process Computer (RWM)	Vermont Yankee & subsequent plants	See Note 1
(11) Loose Parts Detection System	None	—
Safety Relief Valve-Relief Function	ZIMMER-1	IDENTICAL

Note 1:

A General Electric Model 4010 process computer is used on this plant instead of a model 4020, as used on Vermont Yankee. This difference in computer equipment is insignificant.