Round One, Set Five

Materials Engineering Branch

121.10

120.0 MATERIALS ENGINEERING BRANCH

121.0 Materials Integrity Section

121.10 RSP We require that your inspection program for Class 1, 2 and 3 components be in accordance with the revised rules in 10 CFR Part 50, Section 50.55a, paragraph (g) published in the February 12, 1976, issue of the FEDERAL REGISTER. Accordingly, submit the following information:

a. A preservice inspection plan which is consistent with the required edition of the ASME Code. This inspection plan should include any exceptions you propose to the Code requirements.

RESPONSE

The WNP-2 Preservice Inspection Program was submitted to the NRC on March 28, 1979 (D. L. Renberger to S. A. Varga WPPSS Letter No. G02-79-54). This inspection plan includes exceptions and exemptions to the code that the Supply System is planning to use. Exceptions due to access identified during preservice examinations will be submitted in the final report.

b. An inservice inspection plan submitted within six months of the anticipated date for commercial operation.

RESPONSE

The Supply System letter referenced in Part a. states that the WNP-2 Inservice Inspection Program Plan, which will govern the first 10 year inspection interval, will be submitted to you no later than 6 months following the start of commercial operation.

This preservice inspection plan will be reviewed to support the safety evaluation report finding regarding your compliance with preservice and inservice inspection requirements. Our determination of your compliance will be based on:

c. That edition of Section XI of the ASME Code referenced in your PSAR or later editions of Section XI referenced in the FEDERAL REGISTER that you may elect to apply.

RESPONSE

The edition of Section XI of the ASME Code being used for the WNP-2 preservice inspection is identified in Sections 1.0 and 4.0 of the WNP-2 Preservice Inspection Program Plan.

d. All augmented examinations established by the Commission when added assurance of structural reliability was deemed necessary. Examples of augmented examination requirements can be found in the NRC positions on: (1) high energy fluid systems in Section 3.2 of the Standard Review Plan (SRP), NUREG-75/087; (2) turbine disk integrity in Section 10.2.3 of the SRP; and (3) the feedwater inlet nozzle inner radii.

RESPONSE

NRC augmented examination requirements have been addressed in Section 5.0 of the WNP-2 Preservice Inspection Program Plan. Subsequent to the submittal of this Plan, the Supply System has changed some of its inspection requirements for the feedwater nozzles based on NRC question 121.8. The current inspection requirements for feedwater nozzle inner radii can be found in the Supply System response to this question - submitted to the NRC with Round 1 Set 1 responses.

Your response to this item should define the applicable edition(s) and subsections of Section XI of the ASME Code. If any of the examination requirements of the particular edition of Section XI you referenced in the PSAR cannot be met, a request for relief must be submitted, including complete technical justification to support your request.

Detailed guidelines for the preparation and content of the inspection programs to be submitted for staff review and for relief requests are attached as Appendix B to Section 121.0 of our review questions.

RESPONSE

All of the detailed guidelines for the preparation and content of this inspection program as described in Appendix B to Section 121.0 have been met with one exception, that being B.3 which is contained herein.

B.3 Provide the proposed codes and addenda to be used for repairs, modifications, additions or alterations to the facility which might be implemented during this inspection period.

RESPONSE

Repairs, modifications, additions, or alterations to WNP-2 during the Preservice Inspection will use the same code and addenda as that used for the manufacture and installation of the affected components.

Round One, Set Six

Analysis Branch

221.01 - 221.13

Q. 221.01

Section 4.4 of the FSAR contains no discussion of "crud" buildup (i.e., deposits on the surfaces of the fuel elements) and its effect on both the critical power ratio (CPR) and the core pressure drop. Provide your assumptions regarding the sensitivity of the CPR and the core pressure drop due to variations in the amount of crud present. Provide data supporting your assumptions on crud buildup and discuss how crud buildup in the core would be detected.

Response:

In general, the CPR is not affected as crud accumulates on fuel rods (References 1 and 2). Therefore, no modifications to GEXL are made to account for crud deposition. For pressure drop considerations, it is assumed that a conservative amount of crud is deposited on the fuel rods and the fuel rod spacers. This is reflected in a decreased flow area, increased friction factors and increased spacer loss coefficients. The effect of this crud deposition is to increase the core pressure drop by approximately 1.7 psi which can be detected. Further discussion of crud (service-induced variation) and its uncertainty is discussed in Section III of Reference 3 (A revision of Reference 1 in FSAR Section 4.4.7).

References: 1) McBeth, R.V., R. Trenberth, and R.W. Wood, "An Investigation Into the Effects of Crud Deposits on Surface Temperature, Dry-Out, and Pressure Drop, with Forced Convection Boiling of Water at 69 Bar in an Annular Test Section", AEEW-R-705, 1971.

2) Green, S.J., B.W. LeTourneau, A.C. Peterson, "Thermal and Hydraulic Effects of Crud Deposited on Electrically Heated Rod Bundles", WAPD-TM-918, Sept. 1970.

3) General Electric Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application", General Electric Company, January 1977, (NEDO-10958A).

221.02
(4.4.2)

The data base for the approved correlation of the CPR contained in GEXL is for 7x7 fuel bundles and 8x8 fuel bundles having one water rod. However, you have not provided a substantial data base to support the use of the GEXL code to calculate the CPR of the 8x8, two water rod fuel bundles proposed for the WNP-2 facility. Accordingly, demonstrate that the GEXL correlation cited above is applicable to the fuel bundles proposed for the WNP-2 facility by comparison to applicable data. Our requirement in this matter must be satisfied prior to issuance of an operating license for the WNP-2 facility. Alternatively, you may increase the limiting value of the minimum critical power ratio (MCPR) by 0.05 to account for the uncertainty in using the GEXL code for the analysis of the 8x8, two water rod fuel elements.

RESPONSE:

Extensive full-scale boiling transition tests for the 8x8 two water-rod design such as the fuel assembly for Hanford-2 have been conducted in the GE ATLAS test facility. It has been proven that the approved 8x8 GEXL correlation with the appropriate R-factors for the two water-rod design can equally well predict the data as the one water-rod bundle, with a mean ECPR (ratio of the predicted critical power to the measured data) of 0.9879 ($\sim$1.2% conservative) with a standard deviation of 0.0234. Therefore, any MCPR limit penalty due to GEXL correlation is not justifiable. A licensing topical reported was transmitted to NRC in October, 1978. Reference: Letter, from R. L. Gridley (GE) to D. Eisenhut and D. F. Ross (NRC), "General Electric Information Report NEDE-24131,"Basis for 8x8 Retrofit Fuel Thermal Analysis Application", October 5, 1978.

Q. 221.03
(4.4.2)

In Section 4.4.2.5 of the FSAR, you state that: "There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor." Indicate whether this statement refers specifically to the WNP-2 calculations and identify the operating reactor which was used for the data comparison.

Response:

This is a generic statement and is applicable to Hanford 2. This statement is based on actual data reported in 1) "Core Flow Distribution in a Boiling Water Reactor as Measured in Monticello", NEDO-10299A, October, 1976 and 2) H.T. Kim and H.S. Smith, "Core Flow Distribution in a General Electric Boiling Water Reactor as Measured in Quad Cities Unit 1", NEDO-10722A, August, 1976.

Q. 221.04
(4.4.2)

Your discussion on the flow distribution in Section 4.4.2.5 of the FSAR does not address: (1) the effect of uncertainties on the flow distribution; or (2) the effect of channel flow uncertainty, coupled with the other uncertainties listed on Table 4.4-6, on the MCPR uncertainty. Additionally, Table 4.4-6 does not address the flow distribution uncertainties. Accordingly, provide this information.

Response:

The channel flow uncertainty has been inherently considered in its contribution to the MCPR uncertainty when evaluating the probability of a fuel rod subject to a boiling transition in establishing the safety limit MCPR, although it is not explicitly specified in Table 4.4-6. The uncertainties described in Table 4.4-6 are essentially the independent parameters upon which the channel flow (or the core flow distribution) uncertainty depends, except the R-factor and the critical power uncertainties.

In establishing the GETAB safety limit, a BWR core model incorporating these parameter uncertainties (as given in Table 4.4-6) is utilized to evaluate the probability of a fuel rod being subject to a boiling transition by performing a core power and flow distribution calculation through an iterative process by equalizing the pressure drop from the lower plenum to the upper plenum for each fuel assembly. Therefore, the channel flow not only depends on the flow-related parameters such as the total core flow, channel flow area, friction factor multiplier, and the channel friction factor multiplier, but also depends on the core power distribution. The calculated core power distribution is directly dependent on TIP readings and the total core power, through the parameters like feedwater flow, feedwater temperature, and reactor pressure. In additon to those power-related parameters, the channel flow (or flow distribution) is also affected by the core inlet temperature during the pressure drop iteration process.

In conclusion, the channel flow uncertainty is not an independent parameter contributing to the MCPR uncertainty. Although it is not specified in Table 4.4-6, it's effect has been included in evaluating the probability of a boiling transition during a core wide power and flow calculation.

Q. 221.05

In section 4.4.4.5.2 of the FSAR, you that:

> "...analytical models of the individual flow path were developed as an independent check of the tests...When using these models for hydraulic design calculations..."

Provide the equations comprising the model, including your assumptions. Provide a comparison of the results of this model with physical data.

Response:

As stated in 4.4.4.5.2, flow through the bypass flow paths is expressed by the form:

$$W = C_1 \Delta P^{\frac{1}{2}} + C_2 \Delta P^{C_4} + C_3 \Delta P^2$$

The assumptions comprising the model are also discussed in Subsection 4.4.4.5.2. A comparison of model predictions with test data is contained in the following reference:

> "Supplemental Information for Plant Modification to Eliminate Significant Incore Vibrations," (NEDE-21156), Class III, January, 1976.

This reference is being added to Chapter 4 of the FSAR as reference No. 13 of Section 4.4.7.*

*See attached draft pages. The referenced document was sent by R. E. Engel of Operating Plant Licensing, General Electric Co. to D. G. Eisenhut, Operating Plant Licensing of the NRC. The document was approved in May 1976.

The balance of the flow enters the fuel bundle from the lower tie plate and passes through the fuel rod channel spaces. A small portion of the in-channel flow enters the non-fueled rods through three orifice holes in each rod just above the lower tie-plate. This flow, normally referred to as the water-rod flow, remixes with the active coolant channel flow below the upper tie-plate. For a comparison of model predictions with test data, refer to Reference 4.4-13.

4.4.4.5.3 System Heat Balances

Within the fuel assembly, heat balances on the active coolant are performed nodally. Fluid properties are expressed as the bundle average at the particular node of interest and are based on Reference 4.4-6. In evaluating fluid properties, a constant pressure model is used.

The core power is divided into two parts: an active coolant power and a bypass flow power. The bypass flow is heated by neutron-slowing down and gamma heating transferred to the bypass flow from structures and control elements which are themselves heated by gamma absorption and by the (n, α) reaction in the control material. The fraction of total reactor power deposited in the bypass region is very nearly 2%. A similar phenomena occurs within the fuel bundle relative to the active coolant and the water rod flows. The net effect is that 96% of the core power is conducted through the fuel cladding and appears as heat flux.

In design analyses the power is allocated to the individual fuel bundles using a relative power factor. The power distribution along the length of the fuel bundle is specified with axial power factors which distribute the bundle's power among the 24 axial nodes. A nodal local peaking factor is used to establish the peak heat flux at each nodal location.

The relative (radial) and axial power distributions when used with the bundle flow, determine the axial coolant property distribution resulting in sufficient information to calculate the pressure drop components within each fuel assembly type. Once the equal pressure drop criterion has been satisfied, the critical bundle power (the power which would result in critical quality existing at some point in the bundle using the correlation expressed in Reference 4.4-7) is determined by an iterative process for each fuel type.

In applying the above methods to core design, the number of bundles (for a specified core thermal power) and bundle geometry (8x8, rod diameter, etc.) are selected based on power density and linear heat generation rate limits.

### 4.4.7 REFERENCES

4.4-1 "General Electric Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," General Electric Company, November 1973, (NEDO-10958)

4.4-2 "Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello," January 1971. (NEDO-10299)

4.4-3 R. C. Martinelli and D. E. Nelson, "Prediction of Pressure Drops During Forced Convection Boiling of Water," ASME Trans., 70, pp 695-702, 1948.

4.4-4 C. J. Baroozy, "A Systematic Correlation for Two-Phase Pressure Drop," Heat Transfer Conference (Los Angeles), AICLE, Preprint No. 37, 1966.

4.4-5 Jens, W. H., and Lottes, P. A., Analysis of Heat Transfer, Burnout, Pressure Drop, and Density Data for High Pressure Water, USAEC Report-4627, 1972.

4.4-6 Neal, L. G., and Zivi, S. M., "The Stability of Boiling-Water Reactors and Loops," Nuclear Science and Engineering, 30 p. 25, 1967.

4.4-7 S. Levy, et. al., "Experience with BWR Fuel Rods Operating Above Critical Flux," Nucleonics, April 1965.

4.4-8 "Process Instruments and Controls Handbook," Considine, McGraw-Hill Book Co., 1957.

4.4-9 Zadeh and Desoer, "Linear System Theory," McGraw-Hill Book Co., 1963.

4.4-10 "Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactor," General Electric Company, BWR Systems Department, February 1973, (NEDO-10802).

4.4-11 Zuber, N, and Findlay, J. A., "Average Volumetric Concentration in Two-Phase Flow Systems," Trans. ASME, Journal of Heat Transfer, November 1965.

4.4-12 BWR/4 and BWR/5 Fuel Design, General Electric Co., October 1976, (NEDO-20944).

4.4-13 "Supplemental Information for Plant Modification to Eliminate Significant Incore Vibrations," (NEDE-21156); Class III, January, 1976. 2

Q. 221.06
(4.4.4)

Indicate what fraction of the fuel bundle coolant flow is water-rod flow.

Response:

At rated conditions, the water rod flow makes up 1.19% of the total flow.

Q. 221.07
(4.4.4)

In Section 4.4.4.5.1 of the FSAR, you state that "...the nominal expected bypass flow fraction is approximately 10%." Indicate the fraction of calculated bypass flow for the WNP-2 core, including the uncertainty of this value.

Response:

The calculated bypass flow fraction for WNP-2 at rated conditions is 10.0% of the total flow. The one-sigma uncertainty of the bypass flow is estimated to be 2.5%.

QUESTION 221.08 Identify the name of the computer program cited in Section 4.4.4.5. Provide references which document the code.

RESPONSE: The digital computer program used for thermal hydraulic analysis is a proprietary code which has not been documented in the form of a Licensing Topical Report to the NRC.

The code is a parallel flow path computer program used to perform the steady-state BWR reactor core thermal-hydraulic analysis, as is described in Section 4.1.4.6. Program input includes the core geometry, operating power, pressure, coolant flow rate and inlet enthalpy, and power distribution within the core. Output from the program includes core pressure drop, coolant flow distribution, critical power ratio and axial variations of quality, density, and enthalpy for each channel type. This computer program has been used in one form or another in the design and licensing of all BWR 2 through 6 class plants.

The code is available for review at General Electric in San Jose.

Q. 221.09
(4.4.4)

In Section 4.4.4.6.7 of the FSAR, you state that the most limiting condition for thermal-hydraulic stability occurs at the end of core life with power peaking toward the bottom of the core. Indicate whether the typical values of core stability provided in this section are bases on the core characteristics at the end of core life. If not, provide values of the decay ratio for this conditon. Provide the power profile and the void reactivity coeffiients used for this analysis.

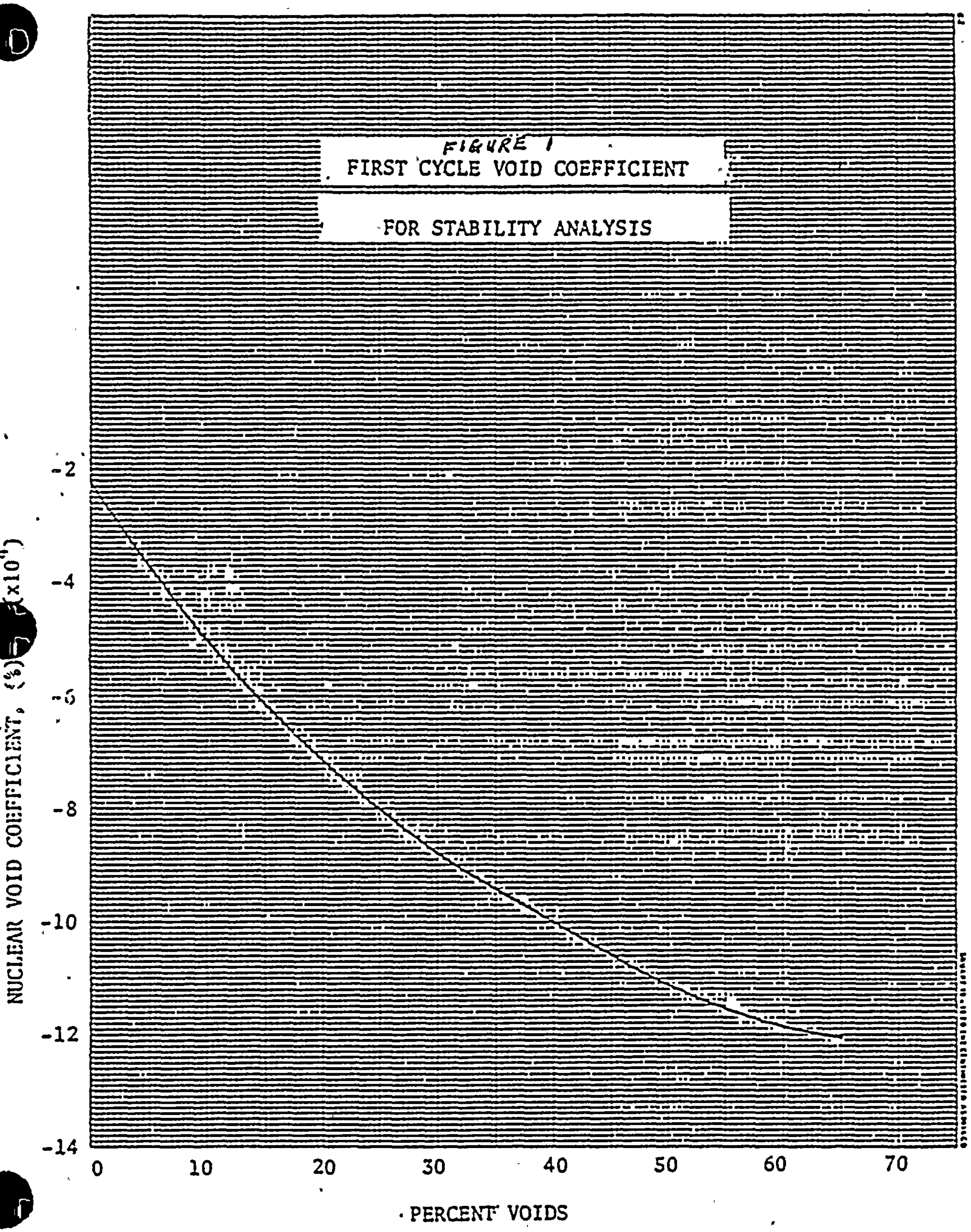
Response:

The core stability results provided are based on core characteristics near end of cycle 1* condition, which is the most limiting for cycle 1* plant operation.

The void reactivity coefficient and axial power profile used in the analysis are provided in the attached Figures 1 and 2 respectively.

*Note: Core stability data for those cycles beyond cycle 1 will be provided with the reload license application.

HANFORD 2 Q221.9



[illegible]

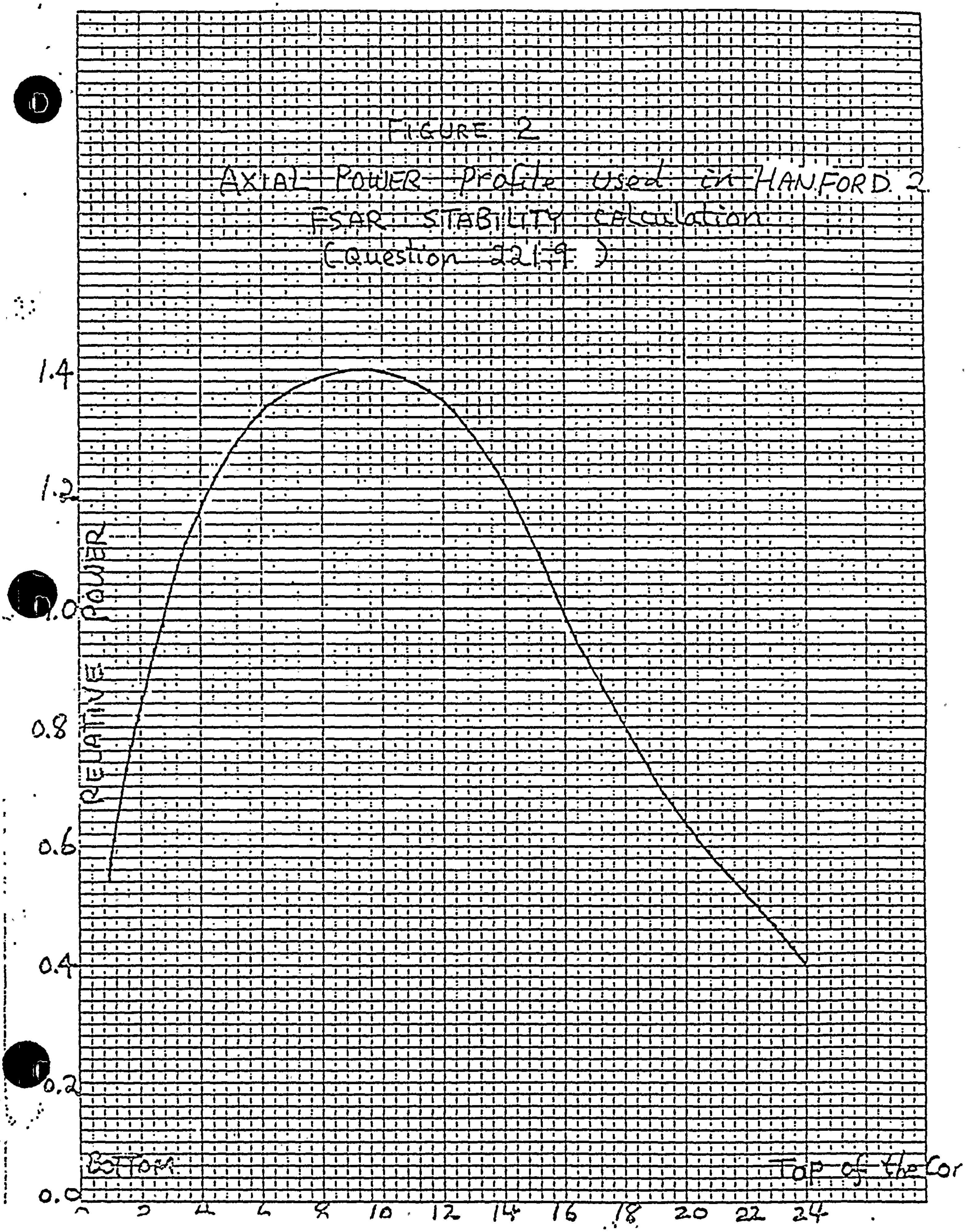


FIGURE 2
AXIAL POWER Profile used in HANFORD 2
FSAR STABILITY calculation
(Question 221.9 )
RELATIVE POWER
1.4
1.2
1.0
0.8
0.6
0.4
0.2
0.0
BOTTOM
Top of the Cor
2
4
6
8
10
12
14
16
18
20
22
24

Q. 221.10
(4.4.4)

In Section 4.4.4.6 of the FSAR, you state that as new experimental or reactor operating data are obtained, the analytical model is refined to improve its capability and accuracy." This implies that a comparison of older versions of the model with test data, as shown in Figure 4.4-4, are meaningless for the WNP-2 facility if it has been analyzed with an updated version. Indicate whether the comparisons of the analytical model with the test data, as given in Figure 4.4-4, are based on the same version of the model as was used for the WNP-2 facility. If not, provide comparisons using the present WNP-2 model. In addition, provide a description of the code. While you may reference a submittal on another docket, references to KAPL reports on the code, STABLE, are unacceptable.

Response:

The comparisons of the model with data, as given in Figure 4.4-4, are based on the same version of the model as was used for WNP-2 facility. The stability licensing topical report, NEDO-21506, provides a description of the analytical methods used in the code as well as model qualification through comparison with test date.

Reference: Licensing Topical Report, "Stability and Dynamic Performance of the General Electric Boiling Water Reactor", January 1977 (NEDO-21506)

Q. 221.11
(4.4.4)

In Section 4.4.4.6 of the FSAR, you reference the GE topical report NEDO-10802 for the model used to perform system stability calculations. You also state that this model is periodically refined as new experimental or reactor operating data are obtained. Indicate whether the version of the model used for the analysis of the WNP-2 facility is described in NEDO-10802. If not, describe the changes.

Response:

The same version of the code described in NEDO-10802 has been used for WNP-2 facility System Stability calculations.

Q. 221.12
RSP
(4.4.4)

We require that a loose parts monitoring (LPM) system be installed in the WNP-2 facility and that it be operational prior to startup testing. Accordingly, provide a description of your proposed LPM system so that we may evaluate it prior to issuance of an operating license. Our positions on the design criteria for a LPM system can be found in Section C of draft Regulatory Guide 1.133, "Loose-Part detection Program for the Primary System of Light-Water-Cooled-Reactors," September 1977. Indicate when you will submit a description of your proposed LPM system.

Response:

A Loose Parts Monitoring System (LPM) will be installed at WNP-2. The system has been specified and meets the intent of draft Reg. Guide 1.133. The below description is taken from substantive parts of the LPM system specification:

The Loose part detection sensors shall be mounted on the exterior of primary coolant system and shall be located at natural collection points where loose parts will be most likely to impact. The general locations shall be:

a. Main Steam Line A & B (26" line): 2 sensors.

b. Feedwater Line A & B (12" line): 2 sensors.

c. Recirculation Water Outlet A & B (24" line): 2 sensors.

d. Reactor Vessel Bottom Head (3/4" to 1" CRD lines): 4 sensors.

All sensors shall be piezo electric accelerometers. Sensors shall be provided with customized mounting blocks suitable for strapping around lines at the above general locations. Magnetic mounts are not permitted.

The Loose Parts Detection System shall provide on-line monitoring of 10 channels with the following performance requirements:

a. Sensitivity at the sensor = 0.05 ft. lb.

(Function of mass of loose part, impact velocity, geometry, and distance from sensor and surface being impacted).

b. System range = 0.05 to 5 ft. lb.

c. Alarm adjustment = 0.2 to 2 ft. lb.

d. Alarm setpoint accuracy, repeatability, stability, drift = 1% of adjustable range.

e. Alarm should be actuated if loose parts weighing 0.25 to 30 lb. impact with a kinetic energy of 0.5 ft. lb. on the inside surface of the reactor vessel within 3 ft. of a sensor.

f. Frequency range: 1.5 HZ to 15 KHZ within 3 db.

g. The signal associated with the loose part impact shall be of sufficiently high magnitude such that background noises (running of pumps, flow induced vibrations, induced voltages due to RFI and magnetic couplings, etc.) do not prevent the detection of loose parts in the primary coolant.

The LPDS shall be provided with loudspeaker, volume control, audio amplifier, and channel selector switch for monitoring unfiltered signal of any one of ten channels. This audio monitoring shall be independent of any automatic/manual recording being performed at the same time. The loudspeaker, volume control, and channel selector switch shall be installed on the front control panel and loudspeaker (center point) shall be mounted at 5'0" above the floor.

One on-line four-channel-cassette, FM/direct tape recorder shall be provided to automatically record unfiltered signal upon detection of metal impact (by preselected alarm setting) by any one of ten channels. The tape recording system shall record preselected four channels simultaneously. For simultaneous manual recording of any four channels, a switching matrix shall be provided. This manual recording will be automatically bypassed upon detection of loose parts by any one channel. Provisions shall be provided for playing back any one out of four recorded channels through the loudspeaker.

For prevention of nuisance alarms, the LPDS shall have provisions for temporarily bypassing the monitoring of loose parts. The bypass signal will be provided by a normally open contact.

For locating and determining the impact energy of a loose part, a loose parts locator, operating on the principle of differences in transit time between sensors, with a printer shall be provided with the following provisions:

Location accuracy: Within a cubic meter

Printout: Date

Time in hours, minutes, and seconds.

Lag time of each channel from the reference channel.

Distance of impact.

Peak impact energy.

Provisions for testing, resetting and spurious indication shall also be included as a part of this locator.

Sections 1.5.1.2.1, 4.4.6, and 7.7 of the FSAR have been revised to account for the LPM system.*

*See attached draft revised pages.

Response:

The consequences of an anticipated transient without scram (ATWS) are mitigated by tripping the recirculation pumps and by manual insertion of the control rods. (For more information, see 15.8).

1.5.1.2 Current Development Program

1.5.1.2.1 Loose Parts Detection ~~Instrumentation for Vibration and Loose Parts Detection~~

~~Vibration testing for reactor internals has been performed on virtually all GE-BWR plants. At the time of issue of NRC Regulatory Guide 1.20, test programs for compliance were instituted.~~

~~General Electric Company has entered a long-term program for the purposes of development of a vibration monitoring system for light water reactors. The objective of the program is the development of a system requiring sensors on only the outside surface of the reactor pressure vessel to provide continual monitoring for the impact and vibration of loose parts during reactor operation.~~

~~The General Electric Company has been working for several years to develop a method of monitoring for excessive vibration of reactor internals or loose parts inside the pressure vessel. The original goal of the development program was to be able to identify the vibration of specific components and to characterize both the amplitude and the frequency of vibration of that component. However, it was recognized that the above stated objective is still beyond the current state-of-the-art capability of the present vibration monitoring systems. Therefore, the development program has been redirected towards a presently feasible goal -- to detect the presence of loose parts in the pressure vessel through an impact detection system. The system under development will also attempt to utilize sensors mounted on the exterior of the reactor pressure vessel to locate the source of excessive vibration such that corrective action can be taken in a timely and expeditious manner. This total development program is a joint effort between the General Electric Co. and the Empire State Atomic Development Associates, Inc. (ESADA).~~

change to

A Loose Parts Detection System will be provided. See Section 7.7.2.15 for a system description.



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4.4.6 INSTRUMENTATION REQUIREMENTS

The reactor vessel instrumentation monitors the key reactor vessel operating parameters during planned operations. This ensures sufficient control of the parameters. The following reactor vessel sensors are discussed in 7.7.1.1.

a. Reactor Vessel Temperature

b. Reactor Vessel Water Level

c. Reactor Vessel Coolant Flow Rates and Differential Pressures

d. Reactor Vessel Internal Pressure

e. Nuclear Incore Monitoring System

4.4.6.1 Loose Parts Monitoring

~~WNP-2 does not have a loose parts monitoring system; however, studies are being performed to evaluate the various systems available. These studies are expected to be complete prior to plant startup.~~

change to

THE Loose part detection sensors ~~shall be~~ is mounted on the exterior of primary coolant system and ~~shall be~~ is located at natural collection points where loose parts will be most likely to impact. The general locations ~~shall be:~~ are:

a. Main Steam Line A & B (26" line): 2 sensors.

b. Feedwater Line A & B (12" line): 2 sensors.

c. Recirculation Water Outlet A & B (24" line): 2 sensors.

d. Reactor Vessel Bottom Head (3/4" to 1" CRD lines): 4 sensors.

See section 7.7.2.15 for further information.

## 7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

### 7.7.1 DESCRIPTION

This subsection discusses instrumentation controls of systems whose functions are not essential for the safety of the plant and permits an understanding of the way the reactor and important subsystems are controlled. The systems include:

a. Reactor vessel - instrumentation

b. Reactor manual control system - instrumentation and controls, or rod control and information system - instrumentation and controls

c. Recirculation flow control system - instrumentation and controls

d. Reactor feedwater system - instrumentation controls

e. Pressure regulator and turbine - generator system - instrumentation and controls

f. Neutron monitoring system (TIP) - instrumentation and controls

g. Process computer system - instrumentation

h. Reactor water cleanup system - instrumentation and controls

i. Area radiation monitoring system - instrumentation and controls

j. Gaseous radwaste system - instrumentation and controls

k. Liquid radwaste system - instrumentation and controls

l. Solid radwaste system - instrumentation and controls

INSERT → m. Loose Parts Detection System - instrumentation and controls

#### 7.7.1.1 Reactor Vessel - Instrumentation

Figure 7.3-15 shows the instrument numbers, arrangements of the sensors, and sensing equipment used to monitor the reactor vessel conditions. Because the reactor vessel sensors



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connection with other test panel signals will not disturb the plant function from which the signal originates. Any isolation amplifier can be removed without disturbing the plant parameter at the amplifier signal input.

The design of the transient test instrumentation is such as to ensure that no fire or other emergency is created by the test circuitry. No electrical faults in the panel will cause misoperation, disturbance or failures in the systems being tested or monitored.

The entire setup is a nonessential system for initial transient tests. It does not control the plant nor does it perform any safety functions.

Cables between isolation amplifiers and the startup computer are nondivisional and these cables can be run in nondivisional trays.

Operation of the recording equipment is accomplished by the startup test engineer--by demand.

The recording equipment is semi-portable and needs only be disconnected from the test panel for removal.

7.7.2.15. [Insert from next page]



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o Add the following beginning on page 7.7-82.

7.7.2.15 Loose Parts Detection System

7.7.2.15.1 System Identification

7.7.2.15.1.1 General

The Loose Parts Detection System (LPDS) shall monitor the reactor vessel for the presence of internal loose parts. Internal movement of components or core vibration of internals is not required to be monitored by this equipment.

7.7.2.15.1.2 Classification

The system is designed to operate during normal plant conditons and is classified as not related to safety.

7.7.2.15.1.3 Reference Design

Similarity to the design of other plants is not known.

7.7.2.15.1.4 Power Sources

The LPDS will be supplied from a non 1E 120 Vac power source.

7.7.2.15.1.5 Equipment Design

The Loose Parts Detection Sensors will be mounted on the exterior of the Primary Coolant System and will be located at natural collection points where loose parts will most likely impact. During startup and normal plant operation the LPDS will be on line to provide visual and audio information to the operator. Also, this data will be 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.

7.7.2.15.1.6 Testability

All system components, with the exception of the sensors, can be tested during plant operation. The sensors can be tested during plant shutdown.

7.7.2.15.1.7 Environmental Considerations

The sensors are exposed to and will be qualified to the environmental conditions of the primary containment as shown in Table 3.11-1. The LPDS panel will be installed in and qualified to the environmental conditions of the reactor building as shown in Table 3.11-1.

7.7.2.15.1.8 Operational Considerations

7.7.2.15.1.8.1 General Information

Audio output will be provided in the main control room.

7.7.2.15.1.8.2 Reactor Operater Information

System status information will be presented to the operator through annunciation.

Q. 221.13
(4.4.2)

In Table 4.4-6 of the FSAR, you list uncertainties used in the statistical analysis performed to establish the limit which ensures the integrity of the fuel cladding. Provide a discussion of the experimental data base used to derive the uncertainty values listed in Table 4.4-6 and provide appropriate references to this data base, where possible. In particular, describe the applicability of these values to the 8x8, two-water rod fuel bundle which you propose for the WNP-2 facility.

Response:

Except for the critical power uncertainty in Table 4.4-6, all the uncertainties are unaffected by the two water-rod assembly design. The GEXL critical power predictability for the 8x8 two water-rod design has been shown to be similar to the standard one water-rod design (see the response to Question 221.02), and therefore, the uncertainties in Table 4.4-6 are conservatively applicable to Hanford-2 two water-rod fuels. The discussion of the uncertainties and the bases used to derive these uncertainties are described in the approved licensing topical report, NEDO-10958-A "General Electric Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application", January 1977.

Round One; Set Five

Quality Assurance Branch

422.1 - 422.9

423.11 - 423.29

Q. 422.1

You state that the Plant Superintendent is responsible for fire protection activities at the plant level. Describe any further delegation of these responsibilities for the fire protection program such as maintenance of fire protection systems, testing of fire protection equipment, fire safety inspections, fire fighting procedures, and fire drills.

Response

Delegation of fire protection program responsibilities include the following:

Shift Supervisors

The Shift Supervisors are responsible for responding to fire related incidents in accordance with the fire brigade handbook and emergency plans and procedures. Testing of selected fire protection systems are performed in accordance with the surveillance testing program procedures.

Plant Industrial Safety Specialist

The Plant Industrial Safety Specialists's responsibilities include:

a. Periodic fire safety audits to ascertain fire defenses are in place, emergency equipment is readily available and in operating order, combustibles are held to minimal quantities and housekeeping is maintained at a high level.

b. Assisting in training of the plant fire brigade, plant staff and non-resident (both WPPSS and Contractors) workers in fire protection aspects.

c. Assist in the preparation of the fire brigade handbook and fire fighting procedures.

d. Reviewing selected work requests for the purpose of identifying possible fire hazards and recommending positive control as requested.

e. Preplanning and critiquing the performance of fire brigade drills.

Plant Training Coordinator

The Plant Training Coordinator is responsible for scheduling and coordinating fire protection training for the plant staff, maintaining records for each person receiving training and maintaining records of fire brigade drills.

Maintenance Supervisors

The maintenance supervisors are responsible for testing of selected fire protection devices in accordance with the surveillance testing program procedures. Preventative and corrective maintenance of fire protection equipment is performed in accordance with plant procedures.

Housekeeping Area Responsible Persons

The applicable Housekeeping Area Responsible Persons defined in plant procedures are responsible for inspecting their assigned plant areas and resolving housekeeping deficiencies in a timely manner.

Fire Brigade Training

The Fire Brigade personnel are trained and retrained in responding to and fighting fires in accordance with the Fire Brigade Training Program. This training program is prepared and instructed by qualified personnel from the Health, Safety and Security Division. Fire Brigade leaders may also conduct this training.

QUESTION 422.2

Describe the proposed composition of your plant fire brigade.

RESPONSE

A minimum fire brigade team complement of five persons will be maintained on all shifts. Fire brigade composition for each shift is as follows:

Fire Brigade Team Leader

Senior employee licensed by the N.R.C.

Assistant Team Leader

Health Physics Technician

Team Members

Equipment operators (2)

Security Force officer

The assigned Security Force officer will participate in brigade emergency actions for a maximum period of 30 minutes and may be relieved earlier upon arrival of the fire department.

The Plant Industrial Safety Specialist is assigned the position of Fire Brigade Chief.

Q. 422.3

Indicate the number of professional persons reporting to each of the following: (1) Manager, Test and Startup Programs; (2) Chief, Design Engineering; (3) Project Engineer, WNP-2; and (4) Manager, Quality Assurance.

Response:

The following number of professional persons were reporting as of January 1979*:

| | |
|---|---|
| o Manager, Test and Startup, WNP-2 | 28 |
| o Chief, Design Engineer | |
| Mechanical/Nuclear | 28 |
| Civil/Environment | 2 |
| Electrical/I&C | 8 |
| Total | 38 |
| o Project Engineer, WNP-2 | 22 |
| o Manager, Quality Assurance | |
| Vendor/Audits | 7 |
| WNP-2, Projects | 15 |
| Operations QA, WNP-2 | 4 |
| Total | 26 |

*The individuals identified are available from the corporate office or working at the WNP-2 site.

Q. 422.4
(13.1.1)

Provide the resume of the persons filling the position of Plant Operations Division Manager and Project Engineer - WNP-2.

Response:

Please see the attached for the resume of the Project Engineering Manager for WNP-2. The "Plant Operations Division Manager" position has been recently deleted from the WPPSS organization. The Plant Manager now reports directly to the Assistant Director, Generation. Chapter 13 is being revised to include updated organization charts and resumes.

| | | | |
|---|---|---|---|
| Title: | Project Engineering Manager - WNP-2 | | |
| Name: | K. D. Cowan | | |
| Education: | 1961 | B.S. Mechanical Engineering | No. Dakota State Univ. |
| | 1965 | Registered Professional Engineer | California |
| | 1971 | M.B.A. | University of Santa Clara, California |

Experience

1978 - Present — Washington Public Power Supply System
Project Engineering Manager

Responsible for all engineering on project, including A/E and NSSS supplier. (30 personnel)

1978 - — General Electric Co.
Senior Program Manager
Advance Engineering

Systems design for next generation BWR.

1976 - 1978 — General Electric Co.
Senior Program Manager
Engineering Department
Systems Engineering

Multidisciplined, Technical and Administrative Program Management. $Three Million Budget.

1976 - — General Electric Co.
Section Manager
Engineering Support
Nuclear Strategy Project Dept.

- Program cost and schedule system - Budget of $8 Million for department
- Program measurement, reporting and control.
- Project cancelled before completion.

| | |
|---|---|
| 1973 - 1976 | General Electric Co.<br>Subsection Manager<br>Project Applications Engineering<br>Project Dept. |

- Standard Plant Project
- Change control on all projects (3 engineers)
- Design/Project issues program on all projects (6 engineers)
- C&I and Computer Application Engineers - 25 projects (12 engineers)
- $1M Budget

| | |
|---|---|
| 1971 - 1973 | General Electric Co.<br>Unit Manager<br>Plant Applications Engineering<br>Project Dept. |

- Requisition project engineers on select projects (10 engineers)
- Operating Plant Applications engineers (3 engineers)
- C&I and Computer Application engineers 20 projects (8 engineers)
- $1M Budget

| | |
|---|---|
| 1969 - 1971 | General Electric Co.<br>Senior Project Engineer<br>Browns Ferry Nuclear Project<br>Project Dept. |

One of 4 project engineers providing total technical integration of Engineering and Licensing.

| | |
|---|---|
| 1968 - 1969 | General Electric Co.<br>Senior Systems Engineer<br>Engineering Department |

- Established product improvement program for department
- Systems evaluation of technical, and cost factors for new product line

| | |
|---|---|
| 1966 - 1968 | General Electric Co.<br>Technical Design Leader<br>Engineering Department |

- Plant layout, piping design and procurement, servicing and refueling equipment design and procurement (8 engineers, 10 draftsmen)
- Heavy customer and A/E interface

| | |
|---|---|
| 1964 - 1966 | General Electric Co.<br>Design Engineer<br>Engineering Department |

Plant layout of mechanical, piping, structural and electrical features, and design of mechanical equipment (10 draftsmen)

| | |
|---|---|
| 1962 - 1964 | General Electric Co.<br>Design Engineer<br>Research Reactors<br>Engineering Department |

One of 3 engineers responsible for design and project engineering of total plant; mechanical, C&I, structural

| | |
|---|---|
| 1961 - 1962 | General Electric Co.<br>Engineering Training Program |

HVAC redesign, fuel testing, site services engineering.

Q. 422.5
(13.1.1)

Provide the requirements for the specific level of experience for your Mechanics, Electricians, and Technicians shown in Figure 13.1-5 of the FSAR. (Refer to subsections 4.5.2 and 4.5.3 of ANSI N15.1-1971.)

Response

As stated in the FSAR, paragraph 13.1.3.1.4, responsible craftsmen (Mechanics, Electricians and I&C Technicians) will have journeymen experience (4 years) in their specialty.

Craftsmen who have less than 4 years experience will assist and work under the direction of qualified journeymen in areas where not previously qualified.

Q. 422.6
(13.1.3)

Provide the resumes of the persons filling the positions of Nuclear Engineer and I&C Engineer shown in Figure 13.1-5 of the FSAR.

Response

The position of I&C Engineer is temporarily unfilled. The resume of the Nuclear Engineer is attached.

Title: Nuclear Engineer

Name: Ian Jenkins

Education: 1965 Associate, Mathematics; Phoenix College, Phoenix, Arizona

1968 B.S., Mathematics; Arizona State University, Tempe, Arizona

Training: 1968 ( 5 weeks) GE PAC 4000 Process Computer Programming by General Electric

1969 (16 weeks) Nuclear Engineering Fundamentals Part II by General Electric

1972 (16 weeks) Fundamentals of Nuclear Instrumentation by General Electric

1978 ( 5 weeks) Station Nuclear Engineering by General Electric

1979 ( 1 week) 4000 Freetime IV Programming by Honeywell

Experience: 1978-Present Washington Public Power Supply System
Nuclear Engineer for WNP-2

Perform nuclear performance evaluations and inplant fuel management activities to assure compliance with the WNP-2 operating license and technical specifications. Assist the WPPSS nuclear fuel management group in optimizing the fuel management program. Responsibilities include the preparation and maintenance of the Plant's nuclear performance evaluation procedures, maintenance of the nuclear engineer's jobs, review and monitoring core performance to assure safety, updating and testing of the on-site process computer, and as the Special Nuclear Material Custodian, responsible for inplant fuel accountability.

1975-1978 General Electric Company
Engineer, Plant Performance
Analysis & System Specifications

Responsible for generation of initial cycle and reload fuel and limit database for all GE BWRs in Japan for inclusion in the sites process computers. Responsibilities also included the instruction of utilities technical staff in the implementation of PCIOMR, teaching at the sites, refuel outage support, programming portions of BUCLE, Back Up Core Limits Evaluations, implementing and instructing its use to plant personnel. Was responsible for specifying portions of NSS software and implemented and evaluated the first GEXL correlation module in the field. Participated in startup tests and post completion reviews for Tokai II, Japan, in addition to qualifying retrofit NSS software for other Far East plants.

1974-1975 General Electric Company
Engineer, Startup & Training,
Operating Plant Services

Responsible for servicing technical requests from nuclear plants post turnover. Performed tests and qualified retrofit NSS software at KKM, Switzerland and Fukushima I & II, Japan. Also responsible for the writing of Service Information Letters on TIP for distribution to all operating and requisitions of GE BWRs.

1968-1974 General Electric Company
Senior Programmer, Control and
Instrument Department

Responsibilities included the development testing and site implementation of scan log and alarm, BOP and nuclear steam supply software for various nuclear plant's process computers. Approximately one year assignment to Nine Mile Point I during fuel load and startup tests as startup support. Participated in startup tests at Pilgrim, Quad Cities and Millstone. Member of task force to resolve axial misalignment of nuclear instruments at seven domestic nuclear plants.

Q. 422.7
(13.4.1)

Describe the functions and responsibilities, the quorum requirements, the meeting frequency, the authority, and the recordkeeping provisions of your Plant Operations Committee.

Response:

From Plant Procedure Manual, Procedure No. 1.2.1, "Plant Operations Committee".

Functions and Responsibilities

The POC serves as a review and advisory organization to the Plant Manager on all matters related to nuclear and radiological safety.

Responsibilities of the POC shall include:

1) Review all safety related procedures covering activities referenced in Section 6.8 of the Technical Specifications and changes thereto, and any other proposed procedures or changes thereto as determined by the Plant Manager to affect nuclear safety.

2) Review all proposed tests and experiments which may affect nuclear safety. Submit proposed tests which may constitute an unreviewed safety question to Corporate Nuclear Safety for review.

3) Review all proposed changes to the Technical Specifications or operation license.

4) Review all proposed changes or modifications to plant systems or equipment that affect nuclear safety. Submit proposed changes or modifications which may constitute an unreviewed safety question to Corporate Nuclear Safety for review.

5) Review of all reports prepared for the Assistant Director, Generation and the Manager, Corporate Nuclear Safety concerning violations of the Technical Specifications and associated evaluation and recommendations to prevent re-currence.

6) Review of those Reportable Occurrences requiring 24 hour notification to the NRC.

7) Perform special reviews and investigations and render reports thereon as requested by the Manager, Corporate Nuclear Safety.

8) Initial review of the Plant Security Plan and the Emergency Plan and subse-quent changes thereto.

## Authority

The POC shall be advisory to the Plant Manager. In the event of disagreement between the recommendation of the POC and the Plant Manager, the course determined by the Plant Manager to be the more conservative shall be followed. Immediate written notification of the disagreement between the POC and the Plant Manager shall be provided to Corporate Nuclear Safety and to the Assistant Director, Generation.

The POC shall have authority to:

1) Recommend to the Plant Manager written approval or disapproval of items (1) through (4) above.

2) Render determinations in writing with regard to whether or not each item (1) through (5) above constitutes an unreviewed safety question.

## Meetings/Records/Quorum Requirements

The POC shall meet at least once a month following issuance of the operating license on call of the Chairman or Vice-Chairman. The quorum for the POC shall consist of four members or alternates, one of which will be the Chairman or Vice-Chairman, and no meeting shall be conducted unless a quorum is present. For all scheduled POC meetings, an effort will be made to have all regular members or approved alternates present.

A record of POC action shall be maintained in the form of minutes of all POC meetings including identification of all documentary material reviewed, conclusions, and recommendations. The minutes shall be reviewed by the POC and approved by the Plant Manager, filed at the plant site, and distributed promptly to:

Each member and alternate of the POC
Manager, Corporate Nuclear Safety
Assistant Director, Generation
Lead Operational Quality Assurance Engineer
Others as needed

Q. 422.8
(13.4.2)

Describe the membership, responsibilities, authority, and method of operation of your Safety Review Board.

Response:

The "Safety Review Board" no longer applies. This organization is now titled "Corporate Nuclear Safety" (CNS).

Membership

One manager and, at a minimum, four reviewing engineers, none of which shall have line responsibility for operation of the unit.

Responsibilities

The CNS group shall function to provide an independent off-site review of designated activities described in section 6.5.2.7 of the NRC Standard Tech. Specs. (STS). In addition, audits of activities described in section 6.5.2.8 of the STS shall be performed under the cognizance of the CNS group.

Authority

The Manager, CNS, shall report to and advise the Assistant Director, Generation, on those areas of responsibility specified above. The Manager, CNS, shall have direct access to the Managing Director as necessary to ensure timely and appropriate resolution of problems identified.

The CNS group shall have authority to contact and communicate as required with others within WPPSS on matters concerning nuclear safety.

Method of Operation

The Manager, CNS, would coordinate the review by the assigned reviewing engineers of safety related activities. Those reviewing engineers would be responsible for the conduct of specifically assigned reviews and monitoring the disposition of CNS recommendations.

The Manager, CNS, shall schedule periodic formal meetings of the CNS staff for the purpose of fostering interaction in reviews of specific activities. Such meetings will be held at least once per calendar quarter during the initial year of unit operation and at least once per six months thereafter.

Records of CNS activities shall be prepared, approved, and distributed as follows:

1. Minutes of each meeting shall be prepared, approved, and forwarded to the Assistant Director, Generation.

2. Reports of reviews encompassed by section 6.5.2.7 of the STS shall be prepared, approved, and forwarded to the Assistant Director, Generation.

3. Audit reports encompassed by section 6.5.2.8 of the STS shall be forwarded to the Assistant Director, Generation, and to the management positions responsible for the areas audited.

Q. 422.9
(13.43)

Describe your audit program for operational phase activities which will satisfy the requirements contained in Section 4.5 of ANSI N18.5-1976. (Refer to Subsection 6.5.2 of the NRC Standard Technical Specifications.)

Response:

For details of the audit program for Operational phase activities at WNP-2, please refer to WPPSS Topical Report WPPSS-QA-004, Rev. 3, "Operational Quality Assurance Program Description", which has been submitted to and approved by the Nuclear Regulatory Commission. This audit program will be applied to activities that are detailed in subsection 6.5.2.8 of the NRC Standard Technical Specifications (NUREG 0123, Rev. 1).

423.0 Initial Test Program

423.11 RSP Your replies to Items 423.2, 423.6, and 423.7 do not clearly identify the amount of participation by General Electric, Burns & Roe, and WPPSS personnel, other than identifying them as members of the Test Working Group and the Plant Operating Committee who prepare, conduct, and review preoperational and startup tests. Additionally, your responses do not indicate that all personnel involved in the preparation, conduct or review of tests will be qualified. In your responses, you should have clearly established minimum requirements for the qualification of supervisory and review positions. Our position in this matter is that, in general, the minimum qualification requirements for individuals who direct preoperational tests or startup tests or who review test procedures are:

a. The minimum qualifications of individuals, at the time the individuals are assigned, who will direct or supervise the conduct of individual preoperational tests are:

   1. A bachelor's degree in engineering or the physical sciences (or the equivalent) and one year of applicable power plant experience. At least three months of indoctrination/training in nuclear power plant systems and component operation of a nuclear power plant should be included in the applicable experience requirement. The experience at a nuclear power plant should be at one which is substantially similar in design to the WNP-2 facility.

   2. Alternatively, a high school diploma, or the equivalent, and four years of power plant experience. Credit for up to two years of this four year experience may be given for related technical training on a one-for-one time basis. At least three months of indoctrination/training in nuclear power plant systems and component operation of a nuclear power plant should be included in the applicable experience requirement. The experience at a nuclear power plant should be at one which is substantially similar in design to the WNP-2 facility.

b. Minimum qualifications of individuals, at the time the individuals are assigned, who direct or supervise the conduct of individual startup tests are:

   1. A bachelor's degree in engineering or the physical sciences (or the equivalent) and two years of applicable power plant experience of which at least one year shall be applicable nuclear power plant experience.

2. Alternatively, a high school diploma or the equivalent and five years of applicable power plant experience of which at least two years shall be applicable nuclear power plant experience. Credit for up to two years of non-nuclear experience may be given for related technical training on a one-for-one time basis.

c. Minimum qualifications of individuals, at the time the specific activity is to be performed, who are assigned to groups responsible for review and approval of preoperational and startup test procedures and/or review and approval of test results are:

1. Eight years of applicable power plant experience with a minimum of two years of applicable nuclear power plant experience. A maximum of four years of non-nuclear experience may be fulfilled by satisfactory completion of academic training at the college level.

Question 423-11

Response:

The following describes the participation by General Electric, Burns and Roe and WPPSS personnel in preparing, conducting and reviewing Preoperational and Startup tests:

All Preoperational tests are prepared by WPPSS personnel with the direct supervision of the WNP-2 Test and Startup Program Department Test Group Supervisors. The Preoperational tests are reviewed by Test Working Group (TWG) members and approved by the TWG Chairman following resolution of all comments. This action is documented in TWG meeting minutes. All Startup tests are prepared by WPPSS personnel. The Startup tests are reviewed by Plant Operation Committee (POC) members and approved by the Plant Superintendent following resolution of all comments. This action is documented in POC meeting minutes.

The Preoperational tests will be performed under the direction of the assigned Test Director who is a Test Engineer in the WPPSS WNP-2 Test and Startup Program Department. The actual manipulation of controls, switches and valves will be made by WPPSS Operations personnel,in general, these will be the same personnel assigned to operate the plant during normal operations.

The Startup tests which are part of the initial fuel load and power ascension program conducted after issuance of the operating license by the NRC are performed with the direction of a WPPSS test engineer by the licensed Plant Operations staff as described in FSAR Chapter 13.

The General Electric Company (GE), the Nuclear Steam Supply System (NSSS) supplier, will provide test engineers to provide technical direction to the WPPSS test engineers and WPPSS Plant Operations personnel for a period from six months prior to fuel load through the power ascension program. The General Electric Test Engineers will advise WPPSS personnel in the testing and operation of systems provided by the General Electric Company. In addition, the GE Operations Manager, who supervises,the GE Test Engineers, will be a conditional member of TWG during the review of the Preoperational

Question 423.11

Response:

tests procedures and results on systems provided by GE. Prior to the arrival of the GE Operations Manager on Site, a GE Senior Project Engineer will act in this capacity. The revision to Section 14.2.2.0 of the FSAR reflects the above comments.*

Burns and Roe will provide a member of their Engineering organization to act as a conditional member of TWG during the review of Preoperational test procedures and results on systems for which Burns and Roe had design responsibility. The revised Section 14.2.2.8 of the FSAR reflects the above comments.*

The WPPSS Project Engineering group provides a TWG member. The Project Engineering TWG member is responsible for obtaining a technical review of Preoperational procedures from the assigned project engineer. The revised Section 14.2.2.6 and 14.2.2.7 of the FSAR reflects the above comments.*

The following describes our position on qualification requirements for all individuals who direct Preoperational tests or Startup tests or who review and approve test procedures..

The revised Section 4.2.2.10 details these minimum qualifications.* It should be noted that GE, Burns and Roe and WPPSS Project Engineering personnel do not direct testing or approve test procedures. The personnel from these organizations provide technical reviews of test procedures as members or conditional members of the TWG. The technical direction given by the GE test engineers is given for testing and operation under the direct control of qualified WPPSS personnel. Therefore, there are no minimum qualifications required for GE, Burns and Roe or WPPSS Project Engineering personnel who are providing support to WPPSS personnel. Although this is apparently contrary to the position indicated in the text of the NRC question 423.11, it is our position that only personnel that direct testing or approve testing procedures are required to fulfill minimum qualifications.

*See attached draft pages.

New Title
Plant
~~and~~ Operations

The ~~Startup Program~~ Manager is Chairman of the TWG and is responsible to convene and conduct TWG meetings and achieve agreement from its membership on the administrative and technical content of program activities.

New Title Operations

The ~~Assistant Plant~~ Superintendant is responsible for providing an operational review of test documents and for submitting safety-related documents to the Plant Operations Committee for review and for communicating the Committee's decisions to the TWG. He provides detailed plant operating procedures and surveillance procedures to be used for plant operation and testing during the Test and Startup Program.

The Operations Q.A. representative to the project is responsible for obtaining a review of proposed activities and technical documents prepared by organizations represented on the TWG, and their results as appropriate, for conformance with applicable standards, codes, and design criteria.

The Project Engineering representative is responsible for obtaining a technical review of proposed activities and test documents by assigned project ~~discipline~~ engineers and for representing Project Management's concern in the Test and Startup Program.

Conditional Members are representative of any organization having responsibility and/or expertise in the area of the TWG meeting agenda. In this situation the representative will be requested to attend the meeting by the TWG chairman.

14.2.2.6 Plant Organization Functions and Responsibilities

The plant organization has overall responsibility for the safe and efficient operation of plant systems and equipment, from provisional acceptance through commercial operation including responsibility for maintenance and operational control. ~~The plant technical section and maintenance section will provide test engineers and test technicians respectively who will be functionally assigned to the Startup Programs Section in support of the Test and Startup Program.~~ Plant organization responsibilities in supporting the Test and Startup Program are discussed in 14.2.2.7.1 below.

The responsibility of the plant organization representative to the Test Working Group is as defined in 14.2.2.5.2.

## 14.2.2.7 WPPSS Support of the Test and Startup Program

### 14.2.2.7.1 Plant Organization

In addition to the responsibilities described in section 14.2.2.6, the plant operating, technical, and maintenance sections provide manpower for development, implementation, and review of testing.

#### 14.2.2.7.1.1 Support During Test and Startup Program Development

Assistance during the development of the Test and Startup Program is provided formally through the plant organization's Test Working Group representative. Input to test procedures and other testing documentation by the plant staff assures:

a. That the operational requirements of test procedures are based on the knowledge and experience of the operating staff.

b. That the technical considerations receive the concurrence of the Plant Technical Staff.

c. That important nuclear and operational safety considerations receive attention by the Plant organization.

#### 14.2.2.7.1.2 Support During Testing

deleted

~~Initial review~~ of test results ~~will be performed by the~~ assigned ~~test~~ engineer, assigned test technician, if applicable, and the plant operations sec~~tion personnel assigned to execute the test~~ to provide an ~~early assessment of the~~ adequacy of a particular test.

Detailed review and analysis of test results will be performed by the plant technical section and/or plant operations section where their particular expertise is deemed necessary by the plant representative to the TWG to support approvals of completed tests.

14.2.2.7.2 Project Division

The WNP-2 Project Division is responsible for the performance of the organizations involved in the design, procurement, and construction of generating projects. The Project Division Manager supports the Test and Startup Program by providing and implementing project control systems, project engineering services, and engineering support services.

The WPPSS Project ~~Site~~ Manager supports the Test and Startup Program by maintaining a high level of current status information available to the Startup Program Organizations to assure that all startup program scheduling and preparation is based on an accurate assessment of the condition of systems and equipment being readied for testing. The ~~Site~~ Project Manager provides liaison with Construction Management for the provision of construction craft support for the implementation of various system lineup and preoperational tests.

14.2.2.7.3 Quality Assurance ~~Technical~~ Division

~~The Technical Division representatives to the project will, in support of the Startup Program, maintain surveillance of program activities to assure technical adequacy in the area of instrumentation and control, electrical, mechanical, fuel and civil engineering applications, quality assurance and testing. Surveillance will include all phases of the program through planning, implementing and results evaluation.~~ delete

The functions of the Quality Assurance organization during the Test and Startup Program will be to survey ongoing efforts to determine that the controls required by various regulations, guides, and standards are effectively implemented. The activities of the Test Working Group will be monitored to assure that the proper degrees of control for safety related activities are being maintained and that required activities are completed where they are prerequisite to another testing activity.

14.2.2.8 Architect-Engineer Support of the Test and Startup Program

Burns and Roe, Inc. is responsible to provide information required to assure timely completion of construction testing and equipment turnover for provisional acceptance. Burns and Roe also provides system-oriented engineers to assist the

Test and Department

WPPSS Startup Program ~~Section~~, as requested by WPPSS, in the provision of system boundary definitions, a preoperational test index, ~~a review of test specifications~~ prepared by WPPSS and technical direction and/or advice and consultation during system and component testing through preoperational testing.

14.2.2.9 General Electric Company Support of the Test and Startup Program

The General Electric Company (GE) is the supplier of the BWR nuclear steam supply system (NSSS) for WNP-2. GE is responsible for generic and specific WNP-2 designs and for the supply of the NSSS. During the construction phase of the plant cycle the GE Resident Site Manager is responsible for all NSSS equipment disposition. When the testing phase of the project begins, the responsibility of GE-NSSS activities are assigned to Preoperational and Startup Group. The GE Preoperational and Startup staff responsibilities are outlined below. ~~Their qualifications are discussed in 14.2.7.3d.~~ Delete

14.2.2.9.1 Staff Responsibilities

14.2.2.9.1.1 GE Operations Manager

The GE Operations Manager is the senior NSSS vendor representative onsite at or near official fuel loading, and is the official site spokesman for GE for preoperational and startup testing. He coordinates with ~~the Station~~ Plant Superintendent for the performance of his duties which are as follows: ~~the Startup and Operations Manager and~~

a. reviewing ~~and approving~~ all NSSS test procedures, including ~~or other test procedures that may be assigned;~~ changes to test procedures, and test results as a conditional member of TWG.

b. providing technical direction to the station staff;

c. managing the activities of the GE site personnel in providing technical direction to WNP-2 personnel in the testing and operation of GE supplied systems;

d. providing liaison between the site and the GE San Jose home office to provide rapid and effective solution to problems which cannot be solved onsite; and

e. participating as a conditional member of the Test Work Group when required.

14.2.2.9.1.2 GE Operations Superintendent

The GE Operations Superintendent is responsible to the GE Operations Manager for supervising the activities of GE Shift Superintendents. He works directly with the WNP-2 Operations Supervisor in providing GE technical direction to the operating organization.

14.2.2.9.1.3 GE Shift Superintendents

The GE Shift Superintendents provide technical direction to WNP-2 shift personnel in the testing and operation of GE supplied systems. They provide 24-hour per day shift coverage as required beginning with fuel loading. They report to the GE Operations Superintendent.

14.2.2.9.1.4 GE Lead Engineer - Startup Test, Design, and Analysis

The GE Lead Engineer - Startup Test, Design, and Analysis is responsible to the GE Operations Manager for supervising the GE shift engineers and for verifying core physics parameters and characteristics and documenting that performance of the NSSS and components conform to test acceptance criteria.

He works with the WNP-2 Technical Supervisor to coordinate and effect implementation of the Startup Test Program instrumentation including special test equipment required to confirm these acceptance criteria.

14.2.2.10 Qualifications of Personnel Supporting the Test and Startup Program

The qualifications described in this section are for those persons having authority to direct testing, review and approve test documentation and results or otherwise have direct influence on the conduct of testing and quality of acquired data. Although other personnel, specifically GE, Burns and Roe
and WPPSS technical specialists, are also involved in these processes, they are under the direction of individuals whose qualifications are described herein and who review and approve all test and startup program activities.

Test and Department

14.2.2.10.1 Startup Program ~~Section~~ Personnel Qualifications

(Plant Manager)

a. At the time of appointment to the active position, the Manager, Startup and Operations shall have ten years of responsible thermal power plant experience such as, but not limited to, managerial, technical or administrative positions, of which a minimum of three years shall be nuclear power plant experience. A maximum of four years of the remaining seven years of experience may be fulfilled by academic training on a one-to-one time basis. This academic training shall be in engineering or the indivdual shall have acquired the experience and training normally required for examination by NRC for a senior operator license whether or not the examination is taken.

~~qualifications on a two years for one basis up to a total minimum requirement of fourteen years, at least ten of which were involved in the nuclear field as described above. Requirements include thorough understanding of Quality Assurance, licensing and regulatory requirements of NRC, state and other agencies.~~

b. Minimum qualifications for the Supervising Test Engineer ~~Lead Engineer and Startup Coordinator~~ are a Bachelor of Science degree in Engineering or related field and six years of applicable experience, at least four of which are in construction, testing or operation of nuclear power generation, propulsion or similar scale test or production facilities. Related experience and training may be substituted for academic qualifications on a two years for one basis up to a total minimum requirement of twelve years, eight of which are involved in the nuclear field as described above. ~~Require~~ Requirements include a demonstrated supervisory ability at the Test Group Supervisor or equivalent level with previous preoperational testing experience, a good understanding of Quality Assurance and regulatory requirements and an ability to effectively communicate with others.

~~or related field and four years of applicable experience, at least three of which are in construction, testing or operation of nuclear power generation, propulsion or similar scale test or production facilities. Related experience and training may be substituted for academic qualification on a two years for one basis up to a~~

(for the Startup Administrative Supervisor)

c. Minimum qualifications are a Bachelor of Science degree in Engineering or related field and five (5) years of applicable experience at least three of which are in the construction testing or operation of nuclear power generating, propulsion of similar scale test or production facilities. Related experience may be substituted for academic requirements when the candidate's professional background and level of achievements clearly demonstrate capabilities to fill the position. Previous testing experience is desirable. A good understanding of quality assurance and regulatory requirements and an ability to effectively communicate with others are necessities. A demonstrated leadership in his discipline and necessary work experience at an equivalent level is evidence of required proficiency.

(for Test Group Supervisor)

d. Minimum qualifications are a Bachelor of Science degree in Engineering or related field and five (5) years of applicable experience, at least three of which are in testing or operation of nuclear power generation, propulsion or similar scale test or production facilities. Related experience may be substituted for academic requirements when the candidate's professional background and level of achievement clearly demonstrate capabilities to fill the position. Previous preoperational testing experience, is required. A good understanding of Quality Assurance and regulatory requirements and an ability to effectively communicate with others are necessities. A demonstrated technical leadership in his discipline and necessary work experience at the Senior Test Engineer or equivalent level is evidence of required proficiency.

e. Minimum qualifications for a Test Engineer are a Bachelor of Science degree in Engineering or related field or, an Associate of Science/ Arts degree in Engineering or related field and two years of related experience or, a graduate of a technical or vocational school in an Engineering or related field and two years related experience. Related experience above the required minimum may be substituted for academic requirements when the candidate's record for performance clearly indicates the ability to fill the position without question. A good understanding of engineering principles and the ability to understand new concepts and to effectively communicate with others is a necessity.

14.2.2.10.2 Plant Organization Personnel Qualifications

Qualifications of the Plant Superintendent, Assistant Plant Superintendent, technical section engineers, and operations supervision are found in 13.1.3.1.

14.2.2.10.3 Architect-Engineer Startup Personnel Qualifications delete

Qualifications of Burns and Roe, Inc., engineers engaged in Test and Startup Program activities will be the same as for WPPSS test engineers engaged in the same capacity.

delete

14.2.2.10.4 NSSS Supplier Startup Personnel Qualifications

The qualifications of General Electric Personnel are discussed in 14.2.7.3.d.

14.2.3 TEST PROCEDURES

14.2.3.1 Development of Test Procedures

~~WPPSS~~ Test Procedures ~~are~~ developed by the Test and Startup Program Department ~~under WPPSS direction~~ to provide a detailed method to demonstrate the capability of the system to perform its design function under ~~[illegible]~~ anticipated operating and accident conditions.

### 14.2.3.3 Approval of Test Procedures

Test procedures will be approved by the Test Working Group (Chairman) by means of consensus of the TWG membership after review of the test procedure as described in 14.2.3.2. Startup test procedures will be approved by the Plant Operating Committee in a similiar manner.

Indication of approval of individual test procedures will be evidenced by the signatures of the chairmen of the Test Working Group and Plant Operations Committee, as required. Evidence of the consensus of these two committees supporting their respective chairmens' signatures of approval will be contained in the minutes of the groups' meetings.

The administrative procedures governing the exercise of approval of test procedures are contained in the WNP-2 Test and Startup Instructions.

## 14.2.4 CONDUCT OF TEST PROGRAM

### 14.2.4.1 Administrative Procedures for Preoperational Testing

#### 14.2.4.1.1 Test Performance Authorization

A significant period of time may have elapsed between the time a preoperational test procedure was approved and the time a test is to be performed. The test procedure is therefore reviewed just prior to initiating the test. Any changes in the system since original approval of the test procedure will be thoroughly researched and the test procedure revised as necessary. The ~~Startup Program~~ Manager will then ~~submit~~ approve the test procedure ~~to the Test Working Group for approval to~~ ~~and~~ Plant Operations for ~~perform~~ (Performance of: the test.

#### 14.2.4.1.2 Preoperational Test Prerequisites

Approval by the Test Working Group to perform a preoperational test also requires consideration of the prerequisite testing required to qualify components and systems for operation. In general, completion of the system lineup testing (see 14.2.1.3) will qualify the system for preoperational testing. System Lineup testing, as a prerequisite to Preoperational testing, will include:

a. Instrumentation and protective relay checks, including calibration, setpoint adjustments, logic verification and line checks;

d. Regulatory Guide 1.58 "Qualifications of Nuclear Power Plant Inspection Examination and Testing Personnel".

WPPSS ~~and Burns and Roe~~ Test and Startup personnel involved in testing meet the requirements of Regulatory Guide 1.58.

delete

General Electric Startup Operations Personnel qualifications meet the requirements of this guide as described below:

1. General Electric personnel are selected and trained according to the criteria of ANSI N18.1-1971 (NRC Regulatory Guide 1.8), with the exception of NRC licensing.

2. The Operations Manager meets the equivalent of ANSI N18.1 paragraph 4.2.2, Operations Manager. The Operations Manager is normally present for pre-operational testing, and therefore will be qualified at the time that preoperational testing is begun.

3. The Operations Superintendent meets the equivalent of ANSI N18.1, paragraph 4.3.1. Supervisors requiring AEC Licenses. The Operations Superintendent will normally be present for preoperational testing and therefore will be qualified at the time that preoperational testing is begun.

4. The Shift Superintendents meet the equivalent of ANSI N18.1, paragraph 4.3.1, Supervisors requiring AEC Licenses. They will be qualified at the time of initial core loading or appointment to the position. Their on-site responsibilities begin prior to fuel loading.

5. The Lead Startup Test Design and Analysis Engineer meets the qualifications of ANSI N18.1, paragraph 4.4.1, Reactor Engineering and Physics. He will be qualified at the time of initial core loading or appointment to the position. His on-site responsibilities begin just prior to fuel loading.

delete

6. The Startup Test Design and Analysis Engineers meet the qualifications of ANSI N18.1, paragraph 3.3, Technical Support Personnel. Their on-site responsibilities begin just prior to final loading.

1. 7. The Startup Control and Instrumentation Engineer meets the qualifications of ANSI N18.1, paragraph 4.4.2, Instrumentation and Control. He will be qualified at the time preoperational testing is begun.

2. 8. The Startup Chemist meets the qualifications of ANSI N18.1, paragraph 4.4.3, Radio-chemistry; utilizing cumulative experience from several reactor starting programs. He will be qualified at the time of initial core loading.

### 14.2.8 UTILIZATION OF REACTOR OPERATING AND TESTING EXPERIENCES IN THE DEVELOPMENT OF THE TEST PROGRAM

As a matter of Supply System policy, a continuous program of review of reactor operating experience is coordinated by the Operations Division of WPPSS. The sources of information reviewed in compliance with this policy are NRC information bulletins, operating experience reports, preoperational test summaries and startup reports from other plants, administrative and test procedures from other plants' startup programs, personal contacts with other nuclear plant licensees or applicants, and additional information supplied by WPPSS Technical and Operations Division's members. All available sources are utilized; relevance to particular WPPSS Nuclear projects is determined in the review process.

The information is reviewed by WNP-2 Startup Program personnel for applicability to the WNP-2 test and startup program, for incorporation into test procedures or for consideration in the administrative control of testing.

### 14.2.9 TRIAL USE OF PLANT OPERATING AND EMERGENCY PROCEDURES

Question 423.12: Several sections of the FSAR, including Sections 14.2.4.4, 14.2.5.2, and 14.2.6.1, reference the WPPSS Test and Startup Program Manual. Incorporate the applicable portions of this manual into the FSAR.

Response:

The WPPSS Test and Startup Program Manual (TSPM) establishes WPPSS policy and procedures for implementation of administrative controls and procedures by which plant equipment and systems are initially tested, documented and accepted by WPPSS. This manual is occasionally referenced in the FSAR and where appropriate, the procedures in the manual are quoted directly or paraphrased. It is not our intent to incorporate directly major portions of the TSPM in the FSAR since the TSPM will be revised as necessary to provide for flexible management of the Startup Program. Of course, if revisions to the TSPM are contrary to the words or intent of the FSAR then appropriate revisions will be made to the FSAR. It is our intention that the intent of the TSPM is accurately described.

In keeping with the above comments, the following action has been taken on the three FSAR Sections referenced in the text of the question:

a. <u>Section 14.2.4.4</u> - This section has been revised to more clearly reflect the intent of the TSPM.*

b. <u>Section 14.2.5.2</u> - No change in the FSAR is required, it is the intent of WPPSS and reflected in the TSPM that all problems and deficiencies discovered during testing are documented and resolved in a manner which allows for auditing to ensure that problems and deficiencies are resolved.

c. <u>Section 14.2.6.1</u> - No change in the FSAR is required, it is the intent of WPPSS and reflected in the TSPM that all testing documents and records be maintained in an appropriate manner and in compliance with applicable Standards and Regulatory Guides.

*See attached draft pages

b. The official working copy of the test procedure is identical to that contained in the master file, including the latest TWG approved revisions or test procedure field changes (see 14.2.4.4).

c. Prerequisite tests have been completed. If TWG and Plant Superintendent approval of the completed test is also a prerequisite that approval will have been obtained.

d. The test procedure has been made available for shift operator review and familiarization. Operator support has been scheduled, as necessary.

e. Test equipment is available or in place as required. Calibration or other readiness requirements have been completed. System instrumentation to be used in the test has been calibrated within the required time period established for surveillance testing and/or preventative maintenance.

f. Test and operating personnel involved in the performance of the test have been briefed immediately prior to starting the test.

14.2.4.4 Modification of Test Procedures During Testing

The Test and Startup Program Manual provides a means of controlling modifications to TWG-approved test procedures during testing. This administrative procedure, contained more specifically in the WNP-2 Test and Startup Instructions, applies to changes made to an approved test procedure during preoperational and startup testing. The procedure does not apply to revisions made during the preparation of test procedures.

The procedure provides controls ~~for~~ of revisions which change~~s~~ the intent or acceptance criteria of the test procedure ~~as defined in the WNP-2 Test and Startup Instructions.~~

The required changes, when identified by the responsible test engineer, are described on a special form (Test Change Notice) which identifies the affected test procedure or plant procedure, justifies the change, and contains spaces for the appropriate approvals.

A Test Change Notice for a Preoperational test is reviewed by the TWG and approved by the Startup and Operations Manager (TWG Chairman).

(for a Startup Test)

The Test Change Notice is approved by the POC and Plant Superintendent prior to implementation.

The Test Change Notice forms a permanent part of the test record.

## 14.2.5 REVIEW, EVALUATION AND APPROVAL OF TEST RESULTS

### 14.2.5.1 Control of Test Results Review

The individuals responsible for reviewing the results of particular tests will be designated by the Startup ~~Program~~ AND OPERATIONS Manager. These reviews will be obtained through TWG members in accordance with their represented areas of responsibility. TWG members will provide names of individuals in their represented organizations who meet the requirements of Regulatory Guide 1.58 - November 1973, for evaluation of inspection and test results.

Based on the recommendations of the qualified reviewers, the completed test will be approved by the TWG, as attested by the Startup Program Manager's signature. POC review and Plant Superintendent approval of startup test results is required.

### 14.2.5.2 Design Organization Participation in Problem Resolution

Failures of tests to meet acceptance criteria and other problems discovered in the course of testing will be documented as deficiencies in accordance with the requirements of the Test and Startup Program Manual. Reports of such deficiencies will indicate the parties or organizations deemed responsible for providing an acceptable resolution of the deficiency. The responsible organization will be requested to provide a resolution of the defined problem.

Documentation of the final resolution will include the recommendation of the responsible organization and a description of the measures implemented in accordance with that recommendation. Design problems will require resolution by the appropriate WPPSS Technical Division Department, Project Engineering, Plant Technical Staff or original design organization, depending upon the technical nature of the problem.

Question 423.13: Indicate the approximate numbers (by job position) and the approximate schedule relative to fuel loading, for providing test personnel.

Response:

The WPPSS Test and Startup personnel are presently at the WNP-2 Site and reporting to the Startup and Operations Manager. Below is a table of manpower by position.

| | |
|---|---|
| Startup and Operations Manager | 1 |
| Principal Engineer | 1 |
| Supervising Test Engineer | 1 |
| Test Group Supervisors | 3 |
| Startup Administrative Supervisor | 1 |
| Test Engineers | 29 |

In addition, the NSSS manufacturer (GE), by contract, will supply test personnel relative to fuel loading as follows:

| | | Months from Fuel Load |
|---|---|---|
| Operations Manager | (1) | 9 |
| Operations Superintendent | (1) | 6 |
| Startup Engineer | (4) | 6 |
| Lead STD&A Engineer | (1) | 3 |
| STD&A Engineer | (3) | 0 |
| C&I Engineer | (1) | 6 |

Also, the Architect Engineer and Turbine Generator manufacturer will have a representative integrated into the Startup Organization.

Q. 423.14

In response to Item 423.4, you modified Section 14.2.4.1.5 of the FSAR to address the matter of significant modifications or repairs to safety-related systems. Define the term "significant modifications and repairs" and designate the group or individuals authorized to determine the significance of a modification or repair and to determine the requirements for retesting the affected system. Indicate how modifications and repairs which are not considered significant, are to be controlled.

Response:

The answer to this question has been delayed and will be submitted in September 1979.

Question 423.15: Revise Section 14.2.11 of the FSAR to ensure that test procedures will be available not less than 60 days prior to fuel loading.

Response:

Section 14.2.11 will be revised to indicate that test procedures will be available not less than 60 days prior to fuel loading. *

*Draft FSAR changes attached.

quickly. The remaining test is performed to properly adjust the control loop of the recirculation system. For all of these tests the plant performance is monitored by recording the transient behavior of numerous process variables, the one of principal interest being neutron flux. Other imposed transients are produced by step changes in demand core flow, simulating loss of a feedwater heater and simulating failure of the operating pressure regulator to permit takeover by the backup regulator. Table 14.2-3 indicates the power and flow levels at which all these stability tests are performed.

(13) The category of major plant transients includes full closure of all the main steam isolation valves, fast closure of turbine-generator control valves, fast closure of turbine-generator stop valves, loss of the main generator and offsite power, tripping a feedwater pump and several trips of the recirculation pumps. The plant transient behavior is recorded for each test and the results may be compared with the acceptance criteria and the predicted design performance. Table 14.2-2 shows the operating test conditions for all the proposed major transients.

(14) A test is made of the relief valves in which leaktightness and general operability are demonstrated.

(15) At all major power levels the jet pump flow instrumentation is calibrated.

(16) The as-built characteristics of the recirculation system are investigated as soon as operating conditions permit full core flow.

(17) The local control loop performance, based on the drive pump, jet pumps and control equipment is checked.

## 14.2.11 TEST PROGRAM SCHEDULE

The test program schedule for preoperational and startup tests are indicated on Table 14.2-4 and Figure 14.2-4. The test procedures will be made available for review at least 60 ~~30~~ days prior to the test date of fuel load.

Question 423.16: In Section 14.2.7 of the FSAR, you discuss conformance of test programs with applicable Regulatory Guides. Expand this section to discuss conformance with Regulatory Guides 1.52, 1.56, 1.68.1, 1.68.2 and 1.108. Modify the appropriate test descriptions to reflect the staff's positions in these Regulatory Guides.

Response:

FSAR section 14.2.7.3.c. states "All other regulatory guides pertaining to individual tests will be complied with unless noted otherwise in 14.2.12." In addition Appendix C of the FSAR contains information on our conformance to USNRC Regulatory Guides, Division 1. Therefore, no revisions to the FSAR are planned in response to this question. For clarity we are responding with our position for the Regulatory Guides mentioned in the question.

a. Regulatory Guide 1.52, Rev.2, March 1978.
   Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants:

   The preoperational and startup testing program complies with the intent of this regulatory guide. See FSAR Appendix C.3 and Section 6.5.1 for statements of compliance concerning this Regulatory Guide.

b. Regulatory Guide 1.56, Rev 0, June 1973.
   Maintenance of Water Purity in Boiling Water Reactor:

   The preoperational and startup testing program complies with the intent of this Regulatory Guide. See FSAR Appendix C.3 for statements of compliance concerning this Regulatory Guide.

c. Regulatory Guide 1.68.1, Rev 1, January 1977.
   Preoperational and Initial Startup of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants:

   The attached revised FSAR, page C.3-60 (Appendix C.3) describes our compliance to this Regulatory Guide.

Question 423.16:

Response: (cont'd)

d. Regulatory Guide 1.68.2, Rev 0, January 1977.
Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants:

The attached revised FSAR, page C.3-61 (Appendix C.3) describes our compliance to this Regulatory Guide.

e. Regulatory Guide 1.108, Rev 1, August 1977.
Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems for Nuclear Power Plants:

The preoperational and startup testing program complies with the intent of this Regulatory Guide. See FSAR Appencix C.3 for statements of compliance concerning this Regulatory Guide.

Regulatory Guide 1.68, Rev. 1, January 1977

Initial Test Programs for Water-Cooled Reactor Power Plants

Compliance or Alternate Approach Statement:

~~(To be provided at a later date)~~ See attached

General Compliance or Alternate Approach Assessment:

~~(To be provided at a later date)~~ See attached

Specific Evaluation Reference:

~~(To be provided at a later date)~~ See attached

DRAFT Revision to page C.3-59

Regulatory Guide 1.68, Rev. 1, January 1977

Initial Test Programs for Water-Cooled Reactor Power Plants

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to the WNP-2 initial test program since revision 0 of this regulatory guide is committed to in FSAR Section 14.2.7. However, 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:

Refer to FSAR Chapter 14 for description of initial testing program and to section 14.2.7 and Appendix C.2.0 for statements concerning compliance with Regulatory Guide 1.68, Rev. 0. Revision No. 1 of this guide in general clarifies Revision No. 0 and therefore there are no exceptions to the intent of this procedure.

Specific Evaluation Reference:

FSAR Section 14.2.7 and Appendix C.2.0 discussion Reg. Guide 1.68, Rev. No. 0.

Regulatory Guide 1.68.1, Rev. 1, January 1977

Preoperational and Initial Startup of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants

Compliance or Alternate Approach Statement:

(~~To be provided at a later date~~) See attached

General Compliance or Alternate Approach Assessment:

(~~To be provided~~ at a later date)

Specific Evaluation Reference:

~~(To be provided~~ at a later date)

REVISION TO PAGE C.3-60

Regulatory Guide 1.68.1, Rev 1, January 1977 - Preoperational and Initial Startup of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants.

Compliance or Alternate Approach Statements:

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:

The preoperational testing and the initial Startup testing as described in FSAR, Chapter 14, complies with the intent of this Regulatory Guide. However, due to the limitations of the auxiliary steam supply system, the confirmation that the feedwater pumps satisfy required head, flow rate and suction head will not occur until the startup phase of the initial test program when the normal steam supply is available to the feedwater pump turbines.

Specific Evaluation Reference: FSAR Section 14.2.12.1.1

Regulatory Guide 1.68.2, Rev. 0, January 1977

Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants

Compliance or Alternate Approach Statement:

(~~To be provided at a later date~~)

General Compliance or Alternate Approach Assessment:

(To be ~~provided at a later~~ date)

Specific Evaluation Reference:

(To ~~be provided at a later~~ date)

See attached

REVISION TO PAGE C.31-61

Regulatory Guide 1.68.2, Rev 0, January 1977 - 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:

The Startup test described in FSAR section 14.2.12.3.28 complies with the Regulatory Guide with the following exceptions:

a. The test will be initiated by scramming plant from the control room versus a location outside the control room as described in Section C.3 of the Regulatory Guide. This exception is made to better simulate the actual procedure which would be followed if a control evacuation were to occur.

b. The Cold Shutdown Demonstration Procedure as described in section C.4 of the Regulatory Guide may not be performed immediately following the demonstration of achieving and maintaining safe hot standby from outside the control room. Rather this cooldown portion may be performed when cooldown is required during the course of the normal power ascension test program. Although this is an exception to Regulatory Guide 1.68.2, Rev 0, Revision 1 of this Guide contains provisions for a delay in the demonstration of cooldown.

Specific Evaluation Reference: FSAR Section 14.2.12.3.28

Q. 423.17

In Section 14.2.10.1.4 of the FSAR, you refer to the preoperational testing listed in Table 14.2-4. However, Table 14.2-4 is a list of startup tests. Correct this discrepancy.

Response:

Section 14.2.10.1.4 is being revised to refer to Table 14.2-1, Preoperational Tests.*

*Draft FSAR change attached.

operated as systems prior to fuel loading. The intent is to observe any unexpected operational problems from either an equipment or a procedural source and to provide an opportunity for operator familiarizations with the system-operating procedures under operating conditions.

Some of the cold functional testing will be accomplished during the preoperational test program. For example, integrated and simultaneous operation of the following systems may take place during the flush of the total system: Condensate System, Condensate Demineralizer System, LPCI System, Core Spray System, RWCU System, Service Water System, CCW System, and others. As required, additional integrated systems performance will be demonstrated prior to fuel loading.

14.2.10.1.3 Routine Surveillance Testing

Because of the interval between completion of a preoperational test on a system and the requirement for that system to be operated may be of considerable length, a number of routine surveillance tests must be performed prior to fuel loading and must be repeated on a routine basis. The Technical Specifications (Chapter 16) detail the test frequency. In general, this Surveillance Test Program (specified in the Technical Specifications) is instituted prior to fuel loading by the plant operating staff.

14.2.10.1.4 Master Startup Checklist

1

A detailed list of items that must be complete, including the preoperational tests, work requests, design changes and proper disposition of all exceptions noted during preoperational testing listed in Table 14.2-X is rechecked to verify completion just prior to the final approvals for fuel loading and at each significant new step such as heat up, opening MSIV's and power operation.

14.2.10.1.5 Initial Fuel Loading

Fuel loading requires the movement of the full core complement of assemblies from the fuel pool to the core, with each assembly identified by number before being placed in the correct coordinate position. The procedure controlling this movement is arranged so that shutdown margin and subcritical

Q. 423.18

Several of your prerequisites for preoperational tests include the requirement that support systems must have the capability to verify their readiness to function. Provide a description of this readiness verification and indicate which individuals or groups are authorized to make this determination.

Response

Section 2.0 of each preoperational test contains a list of those activities, items or temporary installations that must be completed or installed prior to commencing the test. Subsection 2.3 lists other systems that are required to be in operation or out of operation for the procedure to be performed, including power sources, controls and communications. Each item is signed off by the assigned test engineer prior to performance of the test. The test engineer determines that the related system is capable of supporting the preoperational test by direct inspection and review of that system or component status.

Question 423.19: In our review of your preoperational test phase, we found that several systems and design features may not be scheduled for preoperational testing. Our evaluation of your preoperational test program was based on a comparison of your proposed test program with the structures, systems and components of the WNP-2 facility that:

a. will be relied upon for a safe shutdown and cooldown of the reactor under normal plant conditions;

b. will be relied upon for safe shutdown and cooldown of the reactor under faulted, upset, or emergency conditions;

c. will be relied upon for establishing conformance with safety limits or limiting conditions for operation that will be included in the WNP-2 Technical Specifications;

d. are classified as engineered safety features or will be relied upon to support or assure the operation of engineered safety features within design limits;

e. are assumed to function, or for which credit is taken, in the accident analysis of the WNP-2 facility; and

f. will be utilized to process, store, control, or limit the release of radioactivity.

Accordingly, expand or modify the description of your preoperational test phase to address your plans relative to preoperational testing of the following:

(1) The logic, controls, valves, and components used in the condensate and feedwater heating systems.

Response:

The logic, controls, valves and components used in the Condensate System will be tested during the Condensate System Preoperational System test. Revised section 14.2.12.1.2 describes this test.*

The tests performed on the logic, controls, valves and components of the feedwater heating system are contained in the Acceptance Test for Extraction Steam and Heater Vents and Drain System. Note that this is not described in the FSAR because they are not required to be preoperationally tested by Regulatory Guide 1.68, Rev. 0 which is the basis for our preoperational program, see FSAR section 14.2.7.1. The feedwater heating systems do not perform any of the functions listed in the text of this question.

For your information the Acceptance Tests are the same as preoperational tests with the exception of the review procedure. Acceptance Test procedures and results are not reviewed by the Test Working Group (TWG) but they are submitted to the Startup and Operations Manager for approval.

*See attached draft pages.

Question 423.19 (2): The valves in the automatic depressurization system (ADS), including demonstrations of operability using all alternate pneumatic supplies and a demonstration of the operability of the pneumatic air supply systems. (Refer to Reg. Guide 1.80)

Response:

The Nuclear Boiler System Preoperational Test described in section 14.2.12.1.6 will verify the proper operation of ADS logic and the associated safety/relief valves air piston operation. The demonstration of the valve operability can not be made in place until steam from the reactor is available during power operation. See section 14.2.12.3.26 for description of relief valve testing during the power ascension program. Operation of safety/relief valves air pistons using the air supply from the pneumatic accumulators will be included in the preoperational test in compliance with the intent of Regulatory Guide 1.80.

Question 423.19 (3): The logic of the WNP-2 system which is used to mitigate the consequences of anticipated transients without scram (ATWS) events, including the instrumentation and controls and the hardware to mitigate the ATWS events.

Response:

At present there is no specific ATWS system. However any such system or design changes to existing system will be considered as plant modifications and will be tested or systems will be retested as appropriate and per the requirements of administrative procedures. See FSAR section 14.2.4.1.5.

Question 423.19 (4): The leak-tightness of the control room.

Response:

During the Control, Cable and Critical Switchgear Rooms HVAC System Preoperational Test (FSAR section 14.2.12.1.51) the design requirements that the Control Room can be maintained at a positive pressure using either of the alternate HVAC systems will be demonstrated. FSAR section 9.4.1

Question 423.19 (4): (Cont'd)

Response: (Cont'd)

describes the HVAC design requirements. The demonstration of "leak-tightness" is not required since the HVAC is designed to minimize the ingress of smoke or combustible vapors due to an external fire, or of airborne radioactive contaminants released due to the design basis accident by pressurizing the Control Room.

Question 423.19 (5): The diesel-generator air starting system.

Response:

The diesel-generator air starting system is an integral part of the diesel generator system which is tested during the Standby AC Power System Preoperational Test which is described in FSAR section 14.2.12.1.43.

Question 423.19 (6): The "keep-full" systems for the high pressure core spray (HPCS) system, the low pressure core spray (LPCS) system and the residual heat removal (RHR) system pumps.

Response:

The "keep-full" or water leg pump systems described in FSAR section 6.3.2.2.5 are an integral part of the HPCS, LPCS and RHR system and will be tested as part of the preoperational tests on those systems as described in sections 14.2.12.1.14, 13.2.12.1.13 and 14.2.12.1.7 respectively.

Question 423.19 (7): The automatic transfer of suction from the condensate storage tank (CST) to the suppression pool for the HPCS system.

Response:

The automatic transfer of suction from the condensate storage to the suppression pool is an integral part of HPCS valve controls and interlocks and therefore will be tested during the High Pressure Core Spray System Preoperational test described in section 14.2.12.14.

Question 423.19 (8): The temperature monitoring instrumentation and the heat tracing associated with the CST tank.

Response:

The temperature monitoring instrumentation and the heat tracing associated with the Condensate Storage Tank will be energized and tested as a System Lineup Test (SLT) as a prerequisite to the preoperational testing of the condensate storage and transfer system. (See Section 14.2.1.3)

Question 423.19 (9): The manual isolation capability between the main condenser and the offgas system.

Response:

The manual isolation valves between the main condenser and the offgas system are an integral part of the offgas system pipes and valves and therefore will be tested as part of the Offgas.System Preoperational test described in FSAR section 14.2.12.1.27.

Question 423.19 (10): The manual operations (local-manual) of all valves or dampers that are provided with manual operators for those systems classified as engineered safety features. Your response should indicate whether this will be done as a part of each individual preoperational test, as a test prerequisite, or as a construction acceptance test.

Response:

The manual operations (local-manual) of all valves or dampers that are provided with manual operators will be demonstrated and verified as a System Lineup Test (SLT), which are a prerequisite to the peroperational testing of a system (See section 14.2.1.3). The performance of SLT's is documented in the Prerequisite section of the Preoperational Test.

Question 423.19 (11): The timing test for the flow control valves of the recirculation system.

Response:

The flow control valves, their control systems and hydraulics system will be tested and demonstrated, including setting and verify timing the valve stroke per the system design requirements during the Reactor Recirculation System and Control Preoperational test described in section 14.2.12.1.9. Section 14.2.12.19.c(3) specifically lists recirculation valves and related controls.

Question 423.19 (12): The leak-tightness tests for the emergency core cooling systems (ECCS)

Response:

The leak-tightness tests for ECCS are performed as System Lineup Tests (SLT's) described in section 14.2.1.3 which are a prerequisite to the preoperational testing of a system. In the case of ECCS the SLT will document the appropriate type of required leak rate testing.

Question 423.19 (13): The test firing of squib explosive devices in the traversing in-core probe (TIP) system and the standby liquid control (SLC) system.

Response:

The squib explosive devices in SLC system will be fired during the preoperational test described in section 14.2.12.1.5. As stated in this section the preoperational test will include an injection of demineralized water from SLC tank into the reactor vessel through both of the flow paths. To perform this demonstration, the squib valves will be fired.

The explosive shear valves in the TIP system are not operated during the preoperational test program. The electrical logic and the shear valve control monitor lamp is verified during the performance of TIP system preoprational test described in section 14.2.12.1.20. The explosive valves are not fired in place because their operation is destructive to the TIP cable and tubing.

Question 423.19 (14): The response time testing of engineered safety features including the initiating logic.

Response:

The response time including the initiating logic, of most components and systems of engineered safety features is a design requirement and therefore will be measured and compared to the requirements during the preoperational test program. For the RHR, LPCS and HPCS, the description of the preoperational testing in sections 14.2.12.1.7, 14.2.12.1.13, and 14.2.12.1.14 specifically lists verification of the time from the initiation signal to full flow.

Question 423.19 (15): The heating, air conditioning and ventilating systems in the following areas: the main control room cable spreading room/critical switchgear area, the emergency diesel-generator building, the diesel-generator cable area corridor, the radwaste building, the reactor building emergency cooling system and critical electrical equipment area cooling system.

Response:

For clarity the areas are considered separately:

a. The main control room cable spreading room/critical switchgear area, Heating, Ventilating and Air Conditioning (HVAC) system is part of control cable and critical switchgear rooms HVAC system preoperational test described in section 14.2.12.1.51.

b. The emergency diesel-generator building and the diesel-generator cable area corridor HVAC system are part of the Diesel·Generator Building. H&V System preoperational test described in section 14.2.12.1.56.

c. The Radwaste building HVAC system is tested in the Radwaste Building HVAC system test. The description of this test is contained in the revised section 14.2.12.1.30.*

d. The Reactor building emergency cooling system area HVAC system is part of the Reactor Building Emergency Cooling System Preoperational described in section 14.2.12.1.50.

e. The critical electrical equipment area cooling system, i.e., the HVAC system for Battery Charger Rooms, Battery Rooms, Cold Shutdown Room, and RPS Rooms is part of the Control, Cable and Critical Switchgear Rooms HVAC System Preoperational test described in section 14.2.12.1.56.

Question 423.19 (16): The standby gas treatment system.

Response:

The standby gas treatment system preoperational test is described in FSAR section 14.2.12.1.36.

*See attached draft pages

Question 423.19 (17): The main steam line isolation valve leakage control system.

Response:

The Main Steam Line Isolation Valve Leakage Control System (MSIV-LCS) is a subsystem of the Main Steam System and will be tested in conjunction with the Main Steam System. For clarity,section 14.2.12.1.29 is revised to indicate that the MSIV-LCS valves, heaters, blowers and initiating logic is tested as part of the Main Steam System preoperational test described in that section. *

Question 423.19 (18): The oriented spray cooling system of the ultimate heat sink.

Response:

The oriented spray cooling system of the ultimate heat sink is an integral part of the Standby Service Water System and therefore is tested for proper performance during the Standby Service Water System Preoperational Test described in section 14.2.12.1.48. The details of the testing to verify the heat rejection rate of the oriented spray cooling system are now under development. The method of testing, the engineering basis for the testing and the results of that testing will be reported to the NRC in a report devoted to that subject.

*See attached draft pages.

Revisions required by response to 423.19



TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

2. Reactor feedwater pumps, turbines and auxiliaries

3. Control logic

4. Annunciators and protective devices

14.2.12.1.2 ~~Deleted~~ Insert attached

14.2.12.1.3 Deleted

14.2.12.1.4 Reactor Water Cleanup System Preoperational Test

a. Purpose

To verify the operation of the Reactor Water Cleanup (RWCU) System, including pumps, valves, and filter/demineralizer equipment.

b. Prerequisites

The system lineup tests have been completed and the TWC has reviewed and approved the test procedure and the initiation of testing. Filter aid, and anion and cation resin should be available. Closed Cooling Water (CCW) System and instrument air system must have readiness verification.

c. General Test Methods and Acceptance Criteria

Verification of the RWCU System capability is demonstrated by the proper integrated operation of the following:

(1) Drain flow regulator flow interlocks

(2) System isolation and logic

(3) Valve-operating sequence

(4) Pump operation and related control and logic

14.2.12.1.2 Insert

14.2.12.1.2: Condensate System Preoperational Test

a. Purpose

To verify the operation of Condensate System, including pumps, valves and control systems.

b. Prerequisites

The System Lineup Tests (SLT's) have been completed and the TWG has reviewed the procedure and the Startup and Operations Manager has approved the procedure and the initiation of testing. The condenser, condensate filter demineralizers, feedwater, and control air systems are capable of supporting this test as necessary.

c. General Test Method and Acceptance Criteria

The performance of the Condensate System is verified by the demonstration of the proper operation of the following:

1. Valves and related controls, interlocks and position indicators.
2. Condenstate pumps, condensate booster pumps and auxiliaries.
3. Control logic.
4. Annunciators and protective devices.

14.2.12.1.28 Environs Radiation Monitoring Preoperational Test

a. Purpose

To verify the operation of the environs radiation monitoring system, including sensors and channels, sampling pump and filter equipment.

b. Prerequisites

The system lineup tests have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing. Additionally, indicator and trip units, power supplies and sensor/converters are calibrated according to the vendor's instruction manual.

c. General Test Method and Acceptance Criteria

Verification of the environs radiation monitoring system capability is demonstrated by the proper integrated operation of the following:

(1) Trip Point Check

(2) Annunciation

(3) Recorder

(4) Channel Calibration

(5) Sample Equipment

14.2.12.1.29 Main Steam System Preoperational Test

a. Purpose

To verify the proper operation of the (MSIV's) Main Steam Isolation Valves and related controls.

b. Prerequisites

The System Lineup Tests have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing.

c. General Test Methods and Acceptance Criteria

Verification of the Main Steam System is demonstrated by the following:

(1) Automatic isolation of the MSIV's.

(2) Minimum closing times are met.

(3) MSIV accumulator capacity tests are satisfactory.

(4)

14.2.12.1.30 ~~Deleted.~~ Insert attached

Insert - Revision to FSAR 14.2.12.1.29, page 14.2-64

(4) Valves, heaters, blowers and initiating logic of the Main Steam Isolation Valve Leakage Control System.

14.2.12.1.~~52~~ 30 ~~Standby Service Water Pumphouse H & V~~ Radwaste Building HVAC System Preoperational Test

a. Purpose

To verify that the ~~Standby Service Water Pump House H & V~~ Radwaste Building HVAC System will function in accordance with the design requirements as set forth in the design specifications.

b. Prerequisites

The system lineup tests have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing. the Startup and Operations Manager has

The 480 VAC Power System, Control Air Supply - Service Air system and the Turbine Service Water system is capable of supporting this test as necessary.

c. General Test Methods and Acceptance Criteria

Verification of the ~~Standby Service Water Pump house H & V~~ Radwaste Building HVAC System is demonstrated by the proper operation of the following:

1. Ventilation fans and their related controls
2. Filters and instrumentation
3. Dampers and controls
4. Annunciators and protective devices

TABLE 14.2-1

PREOPERATIONAL TESTS

| Subsection Reference | Test Title | Page Reference |
|---|---|---|
| 14.2.12.1.1 | Reactor Feedwater System Preoperational Test | |
| 14.2.12.1.2 | ~~Deleted~~ Condensate System Preoperational Test | |
| 14.2.12.1.3 | Deleted | |
| 14.2.12.1.4 | Reactor Water Cleanup System Preoperational Test | |
| 14.2.12.1.5 | Standby Liquid Control System Preoperational Test | |
| 14.2.12.1.6 | Nuclear Boiler System Preoperational Test | |
| 14.2.12.1.7 | Residual Heat Removal System Preoperational Test | |
| 14.2.12.1.8 | Reactor Core Isolation Cooling System Preoperational Test | |
| 14.2.12.1.9 | Reactor Recirculation System and Control Preoperational Test | |
| 14.2.12.1.10 | Reactor Manual Control System Preoperational Test | |
| 14.2.12.1.11 | Control Rod Drive Hydraulic System Preoperational Test | |
| 14.2.12.1.12 | Fuel Handling and Vessel Servicing Equipment Preoperational Test | |
| 14.2.12.1.13 | Low Pressure Core Spray System Preoperational Test | |

TABLE 14.2-1 (Continued)

PREOPERATIONAL TESTS

| Subsection Reference | Test Title | Page Reference |
|---|---|---|
| 14.2.12.1.14 | High Pressure Core Spray Preoperational Test | |
| 14.2.12.1.15 | Fuel Pool Cooling and Cleanup System Preoperational Test | |
| 14.2.12.1.16 | Leak Detection System Preoperational Test | |
| 14.2.12.1.17 | Liquid and Solid Radwaste System Preoperational Test | |
| 14.2.12.1.18 | Reactor Protection System Preoperational Test | |
| 14.2.12.1.19 | Neutron Monitoring System Preoperational Test | |
| 14.2.12.1.20 | Traversing In-Core Probe System Preoperational Test | |
| 14.2.12.1.21 | Rod Worth Minimizer System Preoperational Test | |
| 14.2.12.1.22 | Process Radiation Monitoring System Preoperational Test | |
| 14.2.12.1.23 | Area Radiation Monitoring System Preoperational Test | |
| 14.2.12.1.24 | Process Computer Interface System Preoperational Test | |
| 14.2.12.1.25 | Rod Sequence Control System (RSCS) Preoperational Test | |
| 14.2.12.1.26 | Remote Shutdown Preoperational Test | |

TABLE 14.2-1 (Continued)

PREOPERATIONAL TESTS

| Subsection Reference | Test Title | Page Reference |
|---|---|---|
| 14.2.12.1.27 | Offgas System Preoperational Test | |
| 14.2.12.1.28 | Environs Radiation Monitoring System Preoperational Test | |
| 14.2.12.1.29 | Main Steam System | |
| 14.2.12.1.30 | ~~Not Applicable~~ Radwaste Building HVAC System Preoperational Test | |
| 14.2.12.1.31 | Closed Cooling Water System Preoperational Test | |
| 14.1.12.1.32 | Primary Containment Atmospheric Control System | |
| 14.2.12.1.33 | Primary Containment Cooling System | |
| 14.2.12.1.34 | Primary Containment Instrument Air System | |
| 14.2.12.1.35 | Primary Containment Atmospheric Monitoring System | |
| 14.2.12.1.36 | Standby Gas Treatment System | |
| 14.2.12.1.37 | 230/115KV Distribution System | |
| 14.2.12.1.38 | 6.9 KV Distribution System | |
| 14.2.12.1.39 | 4.16KV Distribution System | |
| 14.2.12.1.40 | 480V Distribution System | |
| 14.2.12.1.41 | Instrument Power System | |
| 14.2.12.1.42 | Emergency Lighting | |
| 14.2.12.1.43 | Standby AC Power System | |

TABLE 14.2-1 (Continued)

PREOPERATIONAL TESTS

| Subsection Reference | Test Title | Page Reference |
|---|---|---|
| 14.2.12.1.44 | 250 Vdc Distribution System | |
| 14.2.12.1.45 | 125 Vdc Distribution System | |
| 14.2.12.1.46 | 24 Vdc Distribution System | |
| 14.2.12.1.47 | Plant Service Water System | |
| 14.2.12.1.48 | Standby Service Water System | |
| 14.2.12.1.49 | Plant Communication System | |
| 14.2.12.1.50 | Reactor Building Emergency Cooling System | |
| 14.2.12.1.51 | Control Cable and Critical Switchgear Rooms HVAC System | |
| 14.2.12.1.52 | Standby Service Water pumphouse H&V System | |
| 14.2.12.1.53 | Reactor Building Crane | |
| 14.2.12.1.54 | Primary Containment integrated leakrate test | |
| 14.2.12.1.55 | Secondary Containment integrated leakrate test | |
| 14.2.12.1.56 | Diesel Generator Building H&V System Preoperational Test | |

Question 423.20: We could not conclude from our review of the preoperational test phase and the test abstracts provided in Table 14.2 of the FSAR whether comprehensive testing is scheduled for several of the tests described in this table. Accordingly, clarify or expand the description of the preoperational test phase to address the following:

a. Modify the individual test descriptions of the a-c and the d-c distribution systems or provide an integrated test description to verify proper load group assignments. (Refer to Regulatory Guide 1.41).

Response:

The individual preoperational test on the 230/115KV , 6.9KV, 4.16KV and 480V Distribution Systems have been combined into the Loss of Power and Safety Testing Preoperational Test which is described in the revised FSAR section 14.2.12.1.37.*

Question 423.20 b. Provide your plans to verify that: (1) the dc loads are consistent with the assumptions regarding the sizing of the batteries; and (2) the supplied loads remain operable at the minimum terminal voltage of the batteries which is equivalent to that measured in the initial and periodic load discharge tests. Modify the test descriptions of the 250 Volt d-c, 125 Volt d-c, and 24 Volt d-c systems to include these testing requirements. Provide acceptance criteria for these tests.

Response:

(1) Preoperational tests of the 250V d-c and 125V d-c power systems described in section 14.2.12.1.44 and 14.2.12.1.45 will verify battery capacity and sizing per the loads listed in FSAR Table 8.3-4.

(2) The operability of the 250V d-c and 125V d-c circuits at or below minimum battery voltage will be verified and documented as System Lineup Tests (SLT's) which are prerequisite to the preoperational tests. However, the operability of Class I d-c motors and instrument inverters at or below the minimum battery will be part of 250V d-c or 125V d-c power system preoperational test. This type of current versus time profile testing is not required for the 24V d-c power system because the accident loads are a constant loading condition equivalent to a 3 hour discharge test.

*Draft revised FSAR sections attached.

Question 423.20 b. Cont'd

Response:

(2) (cont'd) The test description in section 14.2.12.1.44, 14.2.12.1.45, and 14.2.12.1.46 have been revised to reflect these comments.*

Question 423.20 c. State how operability of emergency loads using offsite power, will be demonstrated during the tests of the a-c and d-c systems.

Repsonse:

The operability of emergency loads using offsite power will be demonstrated during the Loss of Power and Safety Testing Preoperational test described in the revised FSAR section 14.2.12.1.37*

Question 423.20 d. Modify the description of the primary containment leak rate test to address the progression of test pressures and the method of closure of the containment isolation valves. Clarify whether the type B and C local leak tests will be conducted as a part of the construction testing or of the preoperational testing. (Refer to Appendix J of 10 CFR Part 50.)

Response:

In the revised section 14.2.12.1.54, the following areas have been modified or clarified.*

a. The test pressure progression listed in FSAR section 6.2.6.1 is:
   (1) Performing a reduced pressure test at not less than 17.4 psig.
   (2) Performing a peak pressure test at not less than 34.7 psig.

b. The closure of the containment isolation valves will be by the normal method initiated by an isolation signal.

c. The type Band C local leak tests are described in FSAR section 6.2.6.2 and 6.2.6.3. These tests will be performed and documented as System Lineup Tests (SLT's) which are a prerequisite to the containment leak rate test.

*Draft revised FSAR sections attached.

Question 423.20 e. Identify the testing you will perform to verify the amount of drywell floor bypass leakage. Provide quantitative acceptance criteria for this test.

Response:

The revised section 14.2.12.1.54 which is attached indicates that the drywell-wetwell leakage test will be performed as part of the Primary Containment Integrated Leak Rate Preoperational Test. The acceptance criteria is described in FSAR section 3.8.3.7 is an equivalent leakage no greater than $(A/\sqrt{K})$ of 0.0045 $Ft^2$. The structural integrity of the drywell floor is also tested at $25 \pm^{1}_{0}$ psid.

Question 423.20 f. Provide your plans for testing the primary containment isolation system, including the response times of the containment isolation valves.

Response:

The primary containment isolation system, specifically those valves listed in FSAR Table 6.2-16, are tested for proper operation in their respective system preoperational test. During the preoperational test program, each isolation valve which constitutes the primary containment isolation system will be verified to be operational per the requirements of FSAR Table 6.2-16. This will include verification of the primary and secondary modes of activation provided for isolation valves, the initiation logic and associated sensors and the response times of the containment isolation valves.

Question 423.20 g. Revise the test description of the reactor protection system to provide your plans for assuring that the effects of interfacing hardware (e.g., snubbers and pulse dampers) located between measured variables and the input to the sensors for the reactor protection system, do not compromise the requirements for the channel response time. Provide acceptance criteria that reflect the effect of these interactions.

Response:

There are no snubbers, pulse dampers or other interfacing hardware between measured variables and the input to the sensors for the reactor protection system therefore no provisions have been made in the preoperational test program to measure the effects of such hardware.

Question 423.20 h. Modify the description of the reactor recirculation system and control test to demonstrate the proper operation of the rate limiters of the recirculation flow control valve on the flow controllers. Demonstrate that individual control valve stroke rates do not exceed the assumptions in your analysis of accidents in Section 15 of the FSAR. Provide quantitative acceptance criteria for these tests.

Response:

As stated in the response to Question 423.19 (11), the flow control valves, their control systems and hydraulics system will be tested and demonstrated, including setting and verify timing the valve stroke per the system design requirements during the Reactor Recirculation System and Control Preoperational test described in section 14.2.12.1.9. Section 14.2.12.19.c(3) specifically lists recirculation valves and related controls. The electronic limit of valve stroking is $\leq$10%/second, the value used in the FSAR Section 15.3.2 and 15.4.5 is 11%/second for control failure which closes or opens both valves. The maximum valve stroking time limited by the hydraulic actuator is greater than 10%/second and less than 18%/second, the value used in section 15.3.2 and 15.4.5 for single valve failure is 30%/second. The preoperational test will verify that the valve stroke timing is within the these limits.

replace with insert attached

14.2.12.1.37 ~~230/115KV Distribution System Preoperational Test~~

a. Purpose

~~To verify the operation of the 230/115KV Distribution System.~~

b. Prerequisites

~~The system lineup tests have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing.~~

c. General Test Methods and Acceptance Criteria

~~Verification of the 230/115KV Distribution System shall be accomplished by demonstrating circuit integrity and integrated operation of:~~

1. ~~Circuit breakers~~

2. ~~Controls and interlocks~~

3. ~~Instrumentation~~

4. ~~Protective devices and alarms~~

This Testing is included as a Part of the Loss of Power Safety System Test procedure

14.2.12.1.38 ~~6.9KV Distribution System Preoperational Test~~ Delete

a. Purpose

~~To verify the operation of the 6.9KV Distribution System~~

b. Prerequisites

~~The system lineup tests have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing.~~

c. General Test Methods and Acceptance Criteria

~~Verification of the 6.9KV Distribution System shall be accomplished by demonstrating circuit integrity and integrated operation of:~~

insert "Deleted"

14.2.12.1.37 Loss of Power and Safety Testing Preoperational Test

a. Purpose

To verify the operation of the 230/115KV, 6.9KV, 4.16KV and 480V distribution systems.

To verify the integrated ability of the plant electrical distribution and safety systems to operate on normal and standby power sources during accident conditions.

To verify that loss of a single AC or DC distribution system division (exclusive of the HPCS diesel-generator and batteries) will not prevent the remaining systems from actuating during an accident condition.

b. Prerequisites

The System Lineup Tests have been completed and the procedure has been reviewed by TWG and approved by the Startup and Operations Manager. Initiation of testing has been approved by Startup and Operations Manager. The 125V DC system and the ECCS are available to support testing.

c. General Test Methods and Acceptance Criteria

Verification of the 230/115 KV, 6.9KV, 4.16KV and 480V distribution system operability shall be demonstrated by the following:

(1) Demonstration of circuit integrity and integrated operation of circuit breakers, controls and interlocks, instrumentation, automatic transfer features and protective devices and alarms.

(2) Demonstration of proper system response to a loss of the 230 KV and 115KV distribution systems independently and simultaneously both with and without LOCA/Containment Isolation signals.

(3) Demonstration of proper system response to a loss of the 230/115KV distribution systems and one individual standby Diesel-Generators during an ECCS/Containment Isolation actuation.

Signals for these tests shall be simulated from the actual initiating devices when this is practical.

1. Circuit breaker
2. Controls and interlocks
3. Instrumentation
4. Protective devices and alarms

14.2.12.1.39 ~~4160 Vac Power System Preoperational Test~~ Same Delete

a. Purpose

To verify the operation of the 4160 Vac Power System.

b. Prerequisites

The system lineup tests have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing.

The 125 Vdc System is energized and capable of applying control power to the 4160 Vac switchgear control circuitry.

insert "Deleted"

Same

c. General Test Methods and Acceptance Criteria

Verification of the 4160 Vac Power System shall be accomplished by demonstrating circuit integrity and integrated operation of:

(1) Circuit breakers
(2) Controls and interlocks
(3) Automatic transfer features
(4) Instrumentation
(5) Protective devices and alarms

14.2.12.1.40 ~~480 Vac Power System Preoperational Test~~ Same Delete

~~a Purpose~~

~~To verify the operation of the 480 Vac Power System.~~

insert "Deleted"

b. Prerequisites

The system lineup tests have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing.

The 4160 Vac System is energized and capable of supplying power to the 480 Vac Power Systems.

c. General Test Methods and Acceptance Criteria

Verification of the 480 Vac Power System shall be accomplished by demonstrating circuit integrity and integrated operation of:

(1) Circuit breakers

(2) Controls and interlocks

(3) Instrumentation

(4) Protective devices and alarms

14.2.12.1.41 Instrument Power Preoperational Test

a. Purpose

To verify the operation of the Instrument Power (IP) Systems.

b. Prerequisites

The system lineup tests have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing.

The 125Vdc and the 480Vac power systems are energized and capable of supplying power to the Instrument Power Systems.

c. General Test Methods and Acceptance Criteria

Verification of the Instrumentation Power Systems shall be accomplished by demonstrating circuit integrity and integrated operation of:

(2) Sequential loading of each diesel-generator unit.

(3) Maintenance of specified frequency and voltage during the loading sequence.

(4) Specified full- and over-load performance capabilities.

(5) The diesel-generator's capability to supply power to vital equipment during loss of station normal power conditions and simulated plant operating conditions.

(6) The diesel-generator's capability to reject and re-start their largest single load anytime after the design loading sequence is complete.

14.2.12.1.44 250 Vdc Power System Preoperational Test

a. Purpose

To verify the operation of the 250 Vdc Power System including batteries, chargers, controls, interlocks, instruments and protective devices.

b. Prerequisites

The system lineup tests have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing. Battery room ventilation and 480 Vac power supply to the chargers have received readiness verification.

c. General Test Methods and Acceptance Criteria

Verification of the 250 Vdc Power System is demonstrated by the proper integrated operation of the following:

1. Battery chargers including capability to recharge their battery in accordance with section 8.3.2.1.4.13

2. Batteries (including charge and discharge rate/capacity tests and load profiles described in ~~section~~ table 8.3-5)

3. Protective relays and devices

4. System control logic

5. Instrumentation (including ground detection)

6. Breakers

7. Annunciators

14.2.12.1.45 125 Vdc Power System Preoperational Test

a. Purpose

To verify the operation of the 125 Vdc Power System including batteries, chargers, controls, interlocks, instruments and protective devices.

b. Prerequisites

The system lineup tests have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing. Battery room ventilation and 480 Vac power supply to the chargers have received readiness verification.

c. General Test Methods and Acceptance Criteria

Verification of the 125 Vdc Power System is demonstrated by the proper integrated operation of the following:

1. Battery chargers ~~[illegible]~~ including capability to recharge the battery in accordance with section 8.3.2.1.1.3

2. Batteries (including charge and discharge rate/capacity tests and load profiles described in Table 8.3-4a and b)

3. Protective relays and devices

4. System control logic

5. Instrumentation (including ground detection)

6. Breakers

7. Annunciators

14.2.12.1.46 24 Vdc Power System Preoperational Test

a. Purpose

To verify the operation of the 24 Vdc Power System.

b. Prerequisites

The system lineup test have been completed and the TWG has reviewed and approved the test procedures and the initiation of testing.

Battery room ventilation is operating and 120/240 Vac power is available and capable of supplying the battery chargers.

c. General Test Methods and Acceptance Criteria

~~Verification of the 24 Vdc Power System shall be accomplished by demonstrating circuit continuity and integrated operation of:~~

~~(1) Batteries and battery chargers (including discharge capacity and profile tests)~~

Verification of the 24 VDC power systems shall include demonstations of battery capacity and ~~the~~ battery charger capabilities described in section 8.3.2.1.3.3

14.2.12.1.47 Plant Service Water System Preoperational Test

a. Purpose

To demonstrate the proper operation of the Plant Service Water (TSW) System, including pumps, valves and related controls.

b. Prerequisites

The system lineup tests have been completed and the TWG has reviewed and approved the test

TABLE OF CONTENTS (Continued)

TABLE 14.2-1 (Continued)

PREOPERATIONAL TESTS

| Subsection Reference | Test Title | Page Reference |
|---|---|---|
| 14.2.12.1.27 | Offgas System Preoperational Test | |
| 14.2.12.1.28 | Environs Radiation Monitoring System Preoperational Test | |
| 14.2.12.1.29 | Main Steam System | |
| 14.2.12.1.30 | Not Applicable | |
| 14.2.12.1.31 | Closed Cooling Water System Preoperational Test | |
| 14.1.12.1.32 | Primary Containment Atmospheric Control System | |
| 14.2.12.1.33 | Primary Containment Cooling System | |
| 14.2.12.1.34 | Primary Containment Instrument Air System | |
| 14.2.12.1.35 | Primary Containment Atmospheric Monitoring System | |
| 14.2.12.1.36 | Standby Gas Treatment System | |
| 14.2.12.1.37 | Loss of Power and Safety Testing ~~230/115KV Distribution System~~ | |
| 14.2.12.1.38 | ~~6.9 KV Distribution System~~ Deleted | |
| 14.2.12.1.39 | ~~4.16KV Distribution System~~ Deleted | |
| 14.2.12.1.40 | ~~480V Distribution System~~ Deleted | |
| 14.2.12.1.41 | Instrument Power System | |
| 14.2.12.1.42 | Emergency Lighting | |
| 14.2.12.1.43 | Standby AC Power System | |

cedure and initiation of testing. Contractor use of the Reactor Building Crane for construction purposes is complete.

c. General Test Methods and Acceptance Criteria

Verification of the Reactor Building Crane is demonstrated by the proper operation of the following:

(1) Crane Traverse components

(2) Hook traverse and hoist components

(3) Controls and indicators

(4) Safety devices

(5) Instrumentation

14.2.12.1.54 Primary Containment Integrated Leak Rate Preoperational Test

a. Purpose

To verify overall primary containment integrity by pressurizing to specified test pressures and conducting integrated leak rate measurements.

b. Prerequisites

The system lineup test have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing. The following supporting activities, systems or components must have been completed or received readiness verification:

(1) All (type B and C) local leak testing completed, documented and verified as a System Lineup Test, refer to FSAR section 6.2.6.2 and 6.2.6.3

(2) All containment isolation valves fully operable and closed in the normal manner using the isolation signal.

(3) All containment-associated piping hangers, supports, restraints and anchors have been installed and properly set

(4) Residual Heat Removal and Core Spray Systems preoperational tests complete

(5) A containment area survey completed to locate, isolate or remove any instrumentation, light bulbs, etc., which may be damaged by high external pressure

c. General Test Methods and Acceptance Criteria

Verification of primary containment integrity is demonstrated by pressurizing to the required test pressure. See FSAR section 6.2.6.1 for a detailed test description.

The drywell-wetwell leakage test will be performed as part of this test to verify the acceptance criteria described in FSAR section 3.8.3.7.

14.2.12.1.55 Secondary Containment Integrated Leak Rate Preoperational Test

a. Purpose

To verify overall secondary containment integrity by subjecting the Reactor Building to a specified negative pressure and measuring the inleakage.

b. Prerequisites

The system lineup tests have been completed and the TWG has reviewed and approved the test procedures and the initiation of testing. The following supporting activities or systems/components must have been completed or received readiness verification:

(1) Reactor Building structure complete with personnel and railroad air lock doors installed and operable

(2) Reactor Building conduit, pipe and other structural penetrations sealed

(3) Standby Gas Treatment System

Question 423.21: In our review of your proposed startup testing phase, we found that some tests may not fully conform with the staff positions described in Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," November 1973. Describe how you will conform with the following staff positions in Appendix A to Regulatory Guide 1.68: (1) C.2.f and C.2.i; (2) D.2.a for the high pressure core spray system; (3) D.2.k with respect to the operation of a bypass valve; (4) D.2.o; (5) D.2.r with respect to the trip of two recirculation pumps when the plant is operating at 100 percent of rated power; and (6) D.2.v.

Response:

(1) C.2.f. - "Effluent radiation monitors - verification of response to known source."

The intent of this requirement is met by calibration of the radiation monitors using a known liquid source for the liquid effluent monitors and a known gaseous source for gaseous effluent monitors. The calibrations are performed as System Lineup Tests (SLT's) which are prerequisites to the Process Radiation Monitoring System Preoperational test described in section 14.2.12.1.22 which is preformed prior to fuel loading. In addition, initially just prior to fuel load and continuing through the power ascension program, the chemical and Radiochemical startup test as described in section 14.2.12.3.1 will be performed. During this test additional calibration and response verification will be made where effluent radiation levels are in the response range of the radiation monitors.

C.2.i. - "Reactor vessel head cooling system functional test at operating temperature and pressure."

As stated in FSAR Appendix C.2., pg C.2-70, the functional requirement of the reactor head cooling system design is required at operating pressures less than or equal ot 135 psig. Therefore, for this paragraph to be applicable "(135 psig)" should be part of last sentence. The demonstration of the vessel head spray cooling function of the Residual Heat Removal System will be part of the Residual Heat Removal System Startup Test described in 14.2.12.3.37.

Response:

(2) D.2.a. - "High Pressure Coolant Injection System including injection (25%)."

This system has been replaced with a High Pressure Core Spray (HPCS) which due to the configuration of sprays directly on the core, cannot be tested at power. The capability of the HPCS system to inject to the vessel is demonstrated during the High Pressure Core Spray System Preoperational Test described in section 14.2.12.1.14. The revision to Appendix C.2, page C.2-70 reflects this position.*

(3) D.2.k. - The response of the systems to operation of a Turbine Bypass valve is determined during the performance of the Turbine Valve Surveillance startup test described in section 14.2.12.3.24.

(4) D.2.o. - "Rod pattern exchange demonstration (at the maximum power that rod exchange will be permitted during operation)."

The rod pattern exchange is not a part of the Startup Power Ascension Program since it does not involve the approach of any safety margin or operating limit.

The rod pattern exchange procedure at power is part of the Nuclear Performance Evaluation Procedures and will be performed during the fuel cycle as necessary. The revision to Appendix C.2, page C.2-72 reflects this position.*

(5) D.2.r. - The simultaneous trip of both recirculation pumps is not performed at 100 percent of rated power. The analysis of this event (see FSAR section 15.3.1) indicates there is no decrease in the Minimum Critical Power Ratio (MCPR) and therefore it does not involve the approach of any safety margin or operating limit. Following the trip of both pumps, the vessel water level swell is expected to reach the high level trip thereby shutting down the main turbine and feed pump turbines, and indirectly initiating scrams as a result of the main turbine trip. Since the event would result in a turbine

*See attached draft change.

trip and scram at the 100 percent power test plateau, it is deleted from the test program. The revision to Appendix C.2., page C.2-72 reflects this position.*

(6) D.2.v.- "Effluent monitoring systems - verification of calibration by laboratory analysis of samples (as early in power ascension as possible and repeated at major plateaus)."

Initially just prior to fuel load and continuing through the power ascension program, the Cemical and Radiochemical startup test as described in section 14.2.12.3.1 will be performed. During this test at each major plateau, gaseous and liquid effluents samples will be taken and analyzed. If effluent radiation levels are in the response range of the effluent monitors, calibration verifications will be made.

*See attached draft change.

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

could lead to reduction of system safety performance below expected levels and not the minor procedural and test details which would not cause such a reduction.

Section C.2.c: The generic simulation test appearing in 14. should appear by reference in preoperational and initial startup test programs where on-site full simulation tests are not possible. The guide wording would change to ". . . less than full simulation should be provided or referenced for test where full. . ."

Appendix A, Section C.2.h: The comparison of critical control rod pattern with predicted patterns (Appendix A, Section C.2.d) provides required knowledge of effective overall rod worth. Individual control rod calibrations cannot be performed in a meaningful manner in a large multi-rodded BWR. Therefore, this part of the guide is not applicable to boiling water reactors.

Appendix A, Section C.2.i: The functional requirement of the reactor head cooling system design is required at operating pressures less than or equal to 135 psig. Therefore, for this paragraph to be applicable "(135 psig)" should be part of last sentence.

Appendix A, Section D.2.a: (Insert attached) ~~A 50% power test is safer than the 25% test since the water level and core power are less severe.~~

Appendix A, Section D.2.b: Friction tests are performed on four drives at rated pressure.

Appendix A, Section D.2.f: It is necessary to make more than two calibrations and, therefore, it is not appropriate to limit the test to 50% and 100% power levels.

Appendix A, Section D.2.g: At least six chemical analyses of fluid system are necessary; therefore, the limitations of 25%, 50%, 75%, and 100% are not appropriate.

Appendix A, Section D.2.1: Since this plant design does not include an emergency condenser, this section is not appropriate.

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Insert - Appendix A, Section D.2.a. (Page C.2-70)

The High Pressure Coolant Injection has been replaced by a High Pressure Core Spray System. Due to the configuration of the sprays directly on the core, this system cannot be operated at power. The HPCS injection/core spray is demonstrated during the preoperational test program.

Question 423.22: Modify Section 14.2 of the FSAR to specifically identify each startup test listed in Table 14.2 that you do not consider "essential" to demonstrate the operability of structures, systems, and components which meet any of the criteria listed below.

a. Those that will be used for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period.

b. Those that will be used for safe shutdown and cooldown of the reactor: (1) under transient conditions which are infrequent or moderately frequent events; (2) under postulated accident conditions; and (3) for maintaining the reactor in a safe condition for an extended shutdown period following such conditions.

c. Those that will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the WNP-2 Technical Specifications.

d. Those that are classified as engineered safety features or will be used to support operations, or ensure that operations of engineered safety features are within design limits.

e. Those that are assumed to function, or for which credit is taken, in the WNP-2 accident analysis in Section 15 of the FSAR.

f. Those that will be used to process, store, control, or limit the release of radioactive materials.

Response:

The Electrical Output and Heat Rate startup test is not considered essential. The purpose of this test is to establish baseline performance data and verify warranty conditions.

No modification will be made to Section 14.2 or to Table 14.2-4 since this chapter describes the program which is designed to meet objectives as stated in section 14.2.1.1, which includes "demonstration of the capability of structures, components, and systems to meet performance requirements.

Question 423.23: In our review of the test abstracts provided in your FSAR, we found they are not sufficiently descriptive to permit us to conclude that a comprehensive testing program is planned or that satisfactory test acceptance criteria have been established.

The individual tests abstracts should be modified as indicated below:

a. Provide technical justification for the Level 2 acceptance criterion of ± 7 percent of rated power for the calibration test of the average power range monitor (APRM).

Response:

The Design Basis for the APRM System from which all safety related documents are based is APED-5706, "In-Core Neutron Monitoring System for General Electric Boiling Water Reactors." In Section V of this reference two separate characteristics of the system are stated.

The first characteristic describes the ability of the system to track average reactor power over a wide range of flow control or power change due to control rod manipulation. The maximum error for these transients remained less than 5% in power.

The second characteristic is that the APRM meters can be set to the desired values with ± 2% precision for the flows above 50% of design. This is primarily a specification of the electronics system and refers only to the setting of the meters at calibration and does not refer to tracking accuracy.

Combining the two errors could result in a combined error of 7% and safety analyses have taken this into account.

Question 423.23: b. Modify the test abstract for the reactor core isolation cooling (RCIC) system to provide for five cold, quick starts of the system. Indicate the system conditions for a cold, quick start. The Level 1 acceptance criteria in your FSAR refer to operating restrictions presented in Figure 14.2-3 of the FSAR if these acceptance criteria are not met. However, this figure does not contain these restrictions. Provide the appropriate operating restrictions if Level 1 criteria are not met.

Response:

The response to (b) will be submitted in September 1979.

Question 423.23: c. Expand the description of the provisions which ensure reproducibility of the core power distribution measured in the test of the TIP system, considering both random noise and geometric components. Provide assurance that the process computer properly calculates the power distribution in the reactor core for both symmetrical and non-symmetrical rod patterns. Provide technical justification for the Level 2 acceptance criteria.

*Draft FSAR pages attached.

Response:

Following review of the Startup Test Number 18 - Core Power Distribution with the NSSS supplier and based on their recommendation a revised FSAR Section 14.2.12.3.18 was developed.* Note that this test concerns only the determination of TIP system readings reproducibility which are used by the process computer to determine the core power distribution.

Although there is an option available in the process computer program for a core power distribution calculation with non-symmetrical rod patterns this option will not be used. All core power distribution determinations and operating rod patterns will be for symmetrical rod patterns. Assurance that such calculations are proper are contained in Startup Test Number 13 - Process Computer described in FSAR Section 14.2.12.3.13.

The Level 2 criteria for the revised procedure was provided by the NSSS supplier and is reduced from 7.8% uncertainty in the old FSAR Section 14.2.12.3.18.

Question 423.23: d. Expand the description in Section 14.2.12.3.16 of the FSAR, of the test methods in the selected process temperatures test. Revise the acceptance criteria to be consistent with the stated purposes of the tests.

Response:

See the revised test description in Section 14.2.12.3.16 of the FSAR.* For clarity please note that purpose item (1) is specifically verified by the Level 1 criteria item (1), the purpose item (2) is specifically verified by the Level 2 criteria and the purpose item (3) is specifically verified by the Level 1 criteria items (1) and (2).

*Draft FSAR changes attached.

Question 423.23: e. Modify or clarify the acceptance criteria for the system thermal expansion test during heatup of the WNP-2 facility to provide assurance that the design stress levels or fatique limits will not be exceeded.

Response:

Revised FSAR Section 14.2.12.3.17.4 adds to the acceptance criteria for the system thermal expansion test a Level 1 criteria that allowable displacement values will be available and compared to the actual measured values.

Question 423.23: f. In the description of the core power-void mode test, indicate the mode of control (i.e., either auto or manual) of each of the principal control systems for each test condition. Provide technical justification, or the bases, for the Level 2 acceptance criterion to assure, if the acceptance criterion is just satisfied, that stable performance can be expected throughout core life.

Response:

The attached description of the core power-void mode test, Section 14.2.12.3.21.3, states that the principal control systems will be in the normal modes for the operating condition at which the test is performed.* Therefore, the recirculation system control should be in Master Manual flow control, feedwater control in automatic three element mode and the pressure control in automatic on the turbine control valves.

The revised acceptance criteria of the core power-void mode test, Section 14.2.12.3.21.4 deletes the Level 2 acceptance criterion.* This deletion was made on the recommendation NSSS supplier since the purpose of the test does not involve the response of a particular control system. As stated in the test purpose the test verifies that core power-void dynamic response is within the specified limit. The limit is stated in the Level 1 criterion.

*Draft FSAR page changes attached.

Question 423.23: g. In the description of the pressure regulator startup test, indicate the mode of control (i.e., either auto or manual) of each of the other principal control systems for each test condition. Provide technical justification, or the bases, for the Level 2 acceptance criterion (paragraph 1) to assure, if the acceptance criterion is just satisfied, that stable performance can be expected throughout core life.

Response:

The revised description of the pressure regulator startup test (Section 14. 2.12.3.22.3) states that the principal control systems will be in normal modes for the operating conditions at which the test is performed; in addition at test conditions 3, 5 and 6 the test will be performed with recirculation control system in the local manual control mode.* Therefore, the feedwater control system should be in the automatic three element mode and the recirculation control system should be in the local manual control mode at Test Conditions 1 and 2 and in master manual flow control at test conditions 3, 5 and 6.

The Level 2 criterion is based on the NSSS supplier design objectives and the industry accepted standard for control system response; there is no experience which indicate that control system performance will deteriorate or change with core life.

*Draft page changes attached.

Question 423.23: h. Modify the description of the feedwater system startup test to indicate the mode of control (i.e., either auto or manual) of each of the other principal control systems for each test condition for the feedwater control setpoint changes. Provide technical justification, or your bases, for the Level 2 acceptance criterion (paragraph 1) to assure, if the acceptance criterion is just satisfied, that stable performance can be expected throughout core life. Additionally, modify the test description to include a feedwater heater trip and to specifically identify: (1) the type of trip to be initiated; (2) the feedwater heater(s) involved; and (3) a discussion of how the planned trip compares with the worst case limiting event that could be caused by either a single equipment failure or by an operator error. Modify the acceptance criteria for this latter transient to: (1) identify the parameters or variables to be monitored; (2) provide assurance that the test results of this transient will be compared with the predicted response for the actual test case; and (3) provide quantitative acceptance criteria, and their bases, for the comparison of the actual test results with the predicted repsonse of those variables and parameters which will be monitored.

Response:

The revised description of the feedwater system startup test, Section 14.2.12.3.23.3, states that the other principal control systems will be in their normal operating mode for the given test conditions.* In addition at test conditions 3, 5 and 6 the level setpoint changes will be performed with the recirculation control system in the local manual control mode. Therefore, the pressure control should be in automatic on the turbine control valves and the recirculation control system should be in the local manual control mode at Test Conditions 1 and 2 and in master manual flow control at test conditions 3, 5 and 6.

The Level 2 criterion is based on the NSSS supplier design objective and the industry accepted standard for control system response, there is no experience which indicates that control system performance will deteriorate or change with core life.

*Draft page changes attached.

The demonstration of stable reactor response to subcooling changes during the feedwater system startup test will consist of a loss of a portion of feedwater heating capability. The description and acceptance criteria for this part of the test will not be developed until the studies and analysis are completed for our response to NRC question 211.87 contained in NRC Set 9, round one question, received April 23.

Question 423.23: i. Modify the description of the turbine valve surveillance test to ensure that the rate of valve stroking and timing of the close-open sequence is consistent with the conditions which will be experience during surveillance tests.

Response:

The revised description of the turbine valve surveillance test, Section 14.2.12.3.24.3, contains a statement to ensure that the rate of valve stroking and timing of the close-open sequence is the same as that which will occur during normal surveillance testing. FSAR Section 14.2.12.3.24.1 has also been changed to indicate "periodic" rather than "daily" testing since at present only weekly testing is required; however, this requirement is being reviewed.*

Question 423.23: j. Modify the acceptance criteria for the tests of the main steam isolation valves (MSIV's) which demonstrate that full closure of the MSIV's at 100 percent power is obtained, so as to: (1) identify the parameters and variables that will be monitored; (2) provide assurance that the results of the transients tests will be compared with predicted response for the actual test case; and (3) provide quantitative acceptance criteria, and their bases, for the comparison of the actual test results with the predicted response of those variables and parameters which will be monitored. Additionally provide acceptance criteria for the performance of the safety/relief valves and the RCIC during this transient. Provide acceptance criteria for minimum valves of individual valve closure times.

*Draft page changes attached.

Response:

The acceptance criteria for the main steam isolation valves (MSIV's), Section 14.2.12.3.25.4 has been revised.* The revised section divides the acceptance criteria for the individual MSIV test and the simultaneous full closure at 100% power and clarifies the criterion. The complete list of reactor and plant process parameters which will be recorded during these tests will be determined when the detailed Startup Test Procedures are developed. The parameters and variables which are required to verify the conditions listed in the acceptance criteria will be recorded and compared to the limiting values. In addition, other plant parameters will be recorded during the transients. These additional parameters are recorded to provide analysis information to be used if the acceptance criteria is exceeded and to provide general information concerning reactor and plant transient behavior. The determination of the predicted values referred to in the acceptance criteria are not made until the development of the detailed Startup Test Procedure, which will be available 60 days prior to fuel loading. The determination of the predicted values are in many cases different than those used in accident analysis type prediction which necessarily use more conservative plant response values. Therefore, the prediction for a particular transient Startup Test are not made until the best estimate of actual plant response and conditions are available.

Question 423.23: k. Modify the description of the turbine trip and generator load rejection tests to: (1) indicate that both a turbine trip and a generator load rejection test will be conducted from approximately full power; (2) correct the Level 1 acceptance criteria to be consistent with your design; (3) identify the variables or parameters to be monitored for each trip; (4) provide assurance that the test results will be compared with the predicted response for the actual tests for each type of trip; (5) provide quantitative acceptance criteria, and their bases, for the comparison of the test results with the predicted response for the variables and parameters which will be monitored for each type of trip; and (6) provide acceptance criteria for grid stability with respect to both voltage and frequency, following generator load rejection trips.

*Draft page changes attached.

Response:

The acceptance criteria for the turbine trip and generator load rejection test, Section 14.2.12.3.27.4, has been revised.* The revised section expands and clarifies the acceptance criteria. The complete list of reactor and plant process parameters which will be recorded during these test will be determined when the detailed Startup Test Procedure is developed. The parameter and variables which are required to verify the conditions listed in the acceptance criteria will be recorded and compared to the limiting values. In addition, other plant parameters will be recorded during the transient. The additional parameters are recorded to provide analysis information to be used if the acceptance criteria is exceeded and to provide general information concerning reactor and plant transient behavior. The determination of the predicted values referred to in the acceptance criteria are not made until the development of the detailed Startup Test Procedures, which will be available 60 days prior to fuel loading. The determination of the predicted values are in many cases different than those used in accident analysis type predictions which necessarily use more conservative plant response values. Therefore, the prediction for a particular transient Startup Test are not made until the best estimate of actual plant responses and conditions are available.

As indicated in FSAR Table 14.2-4 only a turbine trip or generator load rejection will be conducted at approximately full power. This is supported in FSAR Appendix C.2, page C.2-71 which addresses General Compliance or alternate approach assessment for Regulatory Guide 1.68; Revision 0, November 1973.

There is no acceptance criteria for grid stability, the determination of grid response to this test is not a purpose of this test. Due to the nature of the grid no significant response is anticipated.

*Draft page changes attached.

Question 423.23: l. Modify the abstract of the recirculation flow control startup test to specify the mode of control (i.e., either auto or manual) for each of the other principal control systems for each test condition where system stability checks will be conducted. Provide technical justification, or your bases, to assure, if Level 2 acceptance criteria are just satisfied, that stable performance can be expected throughout core life.

Response:

The revised description of the recirculation flow control startup test, Section 14.2.12.3.29.3, states the principal control systems will be in the normal modes for the operating conditions at which the test is performed.* Therefore, the feedwater control system should be in the automatic three element mode. The pressure regulator will be in the automatic mode on the turbine control valves.

The Level 2 criterion is based on the NSSS supplier design objective and the industry accepted standard for control system response; there is no experience which indicates that control system performance will deteriorate or change with core life.

Question 423.23: m. Modify the description of the recirculation system startup test to define the various types of trips, including the single and double pump trips, to be conducted for each test condition and the manner in which the pumps will be tripped. Additionally, modify the test description for the pump trips of the recirculation pump, and provide the appropriate acceptance criteria for: (1) flow coastdown; and (2) the transient setpoints for the APRM flow biased rod block and scram. Provide stability criteria for the performance of the WNP-2 facility following these trips.

Response:

The revised description of the recirculation system startup test, Section 14.2.12.3.30.3,defines type of trip, test condition and manner in which the pumps will be performed. In addition, the revised acceptance criteria, Section 14.2.12.3.30.4 clarifies and expands the criteria. The stability type startup tests for test condition 4 on FSAR Table 14.2-4 are planned for performance following the double pump trip

*Draft page changes attached.

and the acceptance criteria for those tests are contained in the FSAR Section 14.2.12.3. No stability testing is planned following the one pump trips as the operating conditions which occur following one pump trip and during restart are within the normal operating power-flow envelope. Note that all pump trips will be performed in conjunction with Startup Tests Number 33 and 34 to record drywell piping and reactor internal vibration during and following pump trips.

Question 423.23: n. Modify the abstracts of the loss of turbine-generator test and the loss of offsite power tests to: (1) describe the initial plant conditions for each test, including the lineup of the plant's electrical system; (2) describe the type of trip to be conducted; (3) identify the variables, parameters and plant equipment to be monitored; (4) provide assurance that the test results will be compared with the predicted response for the actual test case; (5) provide quantitative acceptance criteria, and their bases, for the comparison of the test results with the predicted response for the variables and parameters which will be monitored; and (6) provide acceptance criteria for plant equipment required to function during or following these tests.

Response:

See the revised FSAR Section 14.2.12.3.27, Test Number 27 - Turbine Trip and Generator Load Rejection and 14.2.12.3.31, Test Number 31-Loss of Turbine-Generator and Offsite Power.* These revisions modify the test Prerequisites, Description and Criteria for the tests in response to your question. However, please note that as in the response to item j of this question the complete list of reactor and plant process parameters which will be recorded during these tests will be determined when the detailed Startup Test Procedures are developed. Also note the acceptance criteria for the Turbine Trip and Generator Load rejection test does not refer to electrical system or other plant equipment. The purpose of this test is to demonstrate the response of the reactor and its control system and therefore a specific criteria concerning other plant equipment is not required.

*draft pages attached.

Question 423.23: o. In your description of the test of the reactor water cleanup system, you state that this test will be run in three modes as described in the system process diagram. However, the system process diagrams (i.e., Figures 5.4-17a, 5.4-17b, 5.4-17c and 5.4-18) do not define the three modes. Modify the Level 2 acceptance criteria in this test description to correspond with the information presented in Section 5 of the FSAR. Additionally, expand this test description to discuss automatic isolation on the initiation of the standby liquid control system.

Response:

The three modes of operation referred to in the test description are defined by FSAR Figure 5.4-17b. On this figure "Normal" operation is defined by Mode A, "Hot Standby" operation is by defined as Mode B, and "Blowdown" is defined in note E as a blowdown flowrate to condenser of 126.0 GPM, which occurs during normal operation of the RWCU system during reactor heatup or reactor hot standby operation. See also the revised Criteria, Section 14.2.12.3.36.4.* The automatic isolation features of the RWCU are not demonstrated during the Startup Test. These functions are demonstrated in the RWCU System Preoperational test described in Section 14.2.12.1.4.

Question 423.23: p. In your description of the acceptance criteria for the residual heat removal (RHR) system test, you state that these acceptance criteria are based on flow rates and temperatures in the process diagrams. However, the RHR system process diagrams do not contain this information. Modify the acceptance criteria to include the necessary information.

Response:

See the revised sections 14.2.12.3.37.3 and 14.2.12.3.37.4 which are the Description and Criteria of the Startup Test Number 71-Residual Heat Removal System.* The revisions are based on the stated purpose of this test (Section 14.2.12.3.37.1) which is to demonstrate the ability to operate in the shutdown cooling mode and the steam condensing mode. The criteria has deleted quantitative values, however, such values will be determined during the development of the detailed Startup Test Procedure. These values will be based on the operating conditions anticipated during the performance of the Startup Test

*Draft page changes attached.

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## LIST OF FIGURES

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Revised FSAR Pages 14.2-112 and 14.2-113

14.2.12.3.16 Test Number 16 - Selected Process Temperatures

14.2.12.3.16.1 Purpose

The purpose of this test procedure is to (1) verify the setting of low flow control limiter for the recirculation pumps to avoid coolant temperature stratification in the reactor pressure vessel bottom head region, (2) assure that the measured bottom head drain temperature corresponds to bottom head coolant temperature during normal operations, and (3) identify any reactor operating modes during recirculation pump restarts or one pump operation that cause temperature stratification.

14.2.12.3.16.2 Prerequisites

The preoperational tests have been completed, the POC has reviewed and approved the test procedures, and initiation of testing. System and test instrumentation have been calibrated.

Revised FSAR pages 14.2-112 and 14.2-113

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Revised pages 14.2-112 and 14.2-113

## 14.2.12.3.16.3 Description

The adequacy of bottom drain line temperature sensors will be determined by comparing it with recirculation loop coolant temperature when core flow is 100% of rated.

During initial heatup while at hot standby conditions, the bottom drain line temperature, recirculation loop suction temperature and applicable reactor parameters are monitored as the recirculation flow is slowly lowered to either minimum stable flow or the low recirculation pump speed minimum valve position whichever is the greater. The effects of cleanup flow, CRD flow and power level will be investigated as operational limits allow. Utilizing this data it can be determined whether coolant temperature stratification occurs ~~when the recirculation pumps are on~~ and if so, what minimum recirculation flow will prevent it.

Monitoring the preceeding information during planned pump trips (and restarts) will determine if temperature stratification occurs in the idle recirculation loops or in the lower plenum when one or more loops are inactive.

All data will be analyzed to determine if changes in operating procedures are required.

## 14.2.12.3.16.4 Criteria

Level 1

(1) The reactor recirculation pumps shall not be started nor flow increased unless the coolant temperatures between the steam dome and bottom head drain are within 100°F (38°C).

(2) The recirculation pump in an idle loop must not be started unless the loop suction temperature is within 50°F (28°C) of the steam dome temperature.

Level 2

During two pump operation at rated core flow, the bottom head temperature as measured by the bottom drain line thermocouple should be within 30°F (17°C) of the recirculation loop temperatures.

(Revised p. 14.2-116 & 117)  SCN 79-174 423.23 C



14.2.12.3.18 Test Number 18 - Core Power Distribution

14.2.12.3.18.1 Purpose

The purpose of this test is to determine the reproducibility of the TIP system readings.

14.2.12.3.18.2 Prerequisites

System installation and preoperational tests have been completed, the POC has reviewed and approved the test procedures and initiation of testing. The TIP detector and dummy detector, ball valve time delay, core top and bottom limits, clutch, x-y recorder, and purge system will have been shown to be operational. Instrumentation has been calibrated and installed.

14.2.12.3.18.3 Description

TIP reproducibility consists of a random noise component and a geometric component. The geometric component being due to variation in the water gap geometry and TIP tube orientation from TIP location to location. Measurement of these components is obtained by taking repetitive TIP readings at a single TIP location, and by analyzing pairs of TIP readings taken at TIP locations which are symmetrical about the core diagonal of fuel loading symmetry.

One set of TIP data will be taken at the 50% power level and at least one other set at 75% power or above.

The TIP data will be taken with the reactor operating with an octant symmetric rod pattern and at steady state conditions.

The total TIP reproducibility is obtained by dividing the standard deviation of the symmetric TIP pair nodal ratios by $\sqrt{2}$. The nodal TIP ratio is defined as the nodal BASE value of the TIP in the lower right half of the core divided by its symmetric counterpart in the upper left half. The total TIP reproducibility value that is compared with the test criterion is the average value of the data sets taken.

The random noise uncertainty is obtained from successive TIP runs made at the common hole, with each of the TIP machines making six runs. The standard deviation of the random noise is derived by taking the square root of the average of the variances at nodal levels 5 through 22, where the nodal variance is obtained from the fractional deviations of the successive TIP values about their nodal mean value.

The geometric component of TIP reproducibility is obtained by statistically subtracting the random noise component from the total TIP reproducibility.

14.2.12.3.12.4 Criteria

Level 2

The total TIP uncertainty (including random noise and geometrical uncertainties) obtained by averaging the uncertainties for all data sets shall be less than 6.0%.

The data acquired for random noise uncertainty does not have specific acceptance criteria value and is used only to aid in the analyses of the TIP uncertainty.

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14.2.12.3.17 Test Number 17 - System Expansion

14.2.12.3.17.1 Purpose

The purpose of this test is to verify that the reactor drywell piping system is free and unrestrained in regard to thermal expansion and that suspension components are functioning in the specified manner. The test also provides data for calculation of stress levels in nozzles and weldments.

14.2.12.3.17.2 Prerequisites

The preoperational tests have been completed, the POC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been installed and calibrated.

14.2.12.3.17.3 Description

Record hanger positions of major equipment and piping in the nuclear steam supply system and auxiliary systems after each major thermal cycle until shakedown has taken place (normally about three cycles). During initial heatup a visual inspection is made at an intermediate reactor water temperature to assure components are free to move as designed. Adjustments are made as necessary. Devices for measuring continuous pipe deflections are mounted on main steam and recirculation lines. Motion measured during heatup is compared with calculated values.

14.2.12.3.17.4 Criteria

Level 1

There shall be no evidence of blocking of the displacement of any system component caused by thermal expansion of the system.

Hangers shall not be bottomed out or have the spring fully stretched.

used to measure pipe deflections,

The displacements at the established transducer locations shall not exceed the allowable values, ~~as provided later by Piping Design Subsection.~~ The allowable values of displacement shall be based on not exceeding ASME Section III Code stress allowables.

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14.2.12.3.21 Test Number 21 - Core Power-Void Mode

14.2.12.3.21.1 Purpose

The purpose of this test is to measure the stability of the core power-void dynamic response and to demonstrate that its behavior is within specified limits.

14.2.12.3.21.2 Prerequisites

The preoperational tests have been completed, the POC has reviewed and approved the test procedures and initiation of testing. System instrumentation installed and calibrated and test and instrumentation calibrated.

14.2.12.3.21.3 Description

The core power void loop mode, that results from a combination of the neutron kinetics and core thermal hydraulic dynamics, is least stable near the natural circulation end of the rated 100 percent power rod line. A fast change in the reactivity balance is obtained by moving a very high worth rod only 1 or 2 notches. The recorded response must be carefully examined for lower stability mode responses in the 0.4 Hz area. A special mid pass filter is placed on the flux signals and dome pressure to emphasize this area, as well as to suppress noise in the signal background. The principal control systems will be in the normal modes for operating conditions at the test condition.

14.2.12.3.21.4 Criteria

Level 1

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to control rod movement.

Level 2

Not applicable

~~a. The decay ratio must be less than or equal to 0.25 for each total core process variable that exhibits oscillatory response to control rod movement when the plant is operating above the lower limit setting of the Master Flow Controller.~~

b. The decay ratio must be less than or equal to 0.50 for each localized process variable (LPRM) that exhibits oscillatory response to control rod movement when the plant is operating above the lower limit setting of the Master Flow Controller.

## 14.2.12.3.22 Test Number 22 - Pressure Regulator

### 14.2.12.3.22.1 Purpose

The purposes of this test are a) to determine the optimum settings for the pressure control loop by analysis of the transients induced in the reactor pressure control system by means of the pressure regulators, b) to demonstrate the takeover capability of the backup pressure regulator upon failure of the controlling pressure regulator and to set spacing between the set points at an appropriate value and c) to demonstrate smooth pressure control transition between the control valves and bypass valves when reactor steam generation exceeds steam used by the turbine.

### 14.2.12.3.22.2 Prerequisites

The preoperational tests have been completed, the POC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

### 14:2.12.3.22.3 Description

The pressure set point will be decreased rapidly and then increased rapidly by about 10 psi (0.7 kg/cm$^2$) and the response of the system will be measured in each case. It is desirable to accomplish the set point change in less than 1 second. At specified test conditions the load limit setpoint will be set so that the transient is handled by control valves, bypass valves and both. The backup regulator will be tested by simulating a failure of the operating pressure regulator so that the backup regulator takes over control. The response of the system will be measured and evaluated and regulator settings will be optimized. ~~At certain conditions, the results of the Backup Regulator test will be included with the test report on the Core Power Void Mode test~~. Because the near step transient occurs downstream of the lag filter, this disturbance yields valuable stability data in the midfrequency range (i.e., 0.13.0 Hz).

The principal control system will be in their normal operating mode for the given test condition, In addition ~~th~~ at test conditions 3,5 and 6 the test will be performed with the recirculation control system in the local manual control mode,

## 14.2.12.3.23 Test Number 23 - Feedwater System

### 14.2.12.3.23.1 Purpose

The purposes of this test are a) to adjust the feedwater control system for acceptable reactor water level control, b) to demonstrate stable reactor response to subcooling changes, c) to demonstrate the capability of the automatic core flow runback feature to prevent low water level scram following the trip of one feedwater pump.

### 14.2.12.3.23.2 Prerequisites

The preoperational tests have been completed, the POC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

### 14.2.12.3.23.3 Description

Reactor water level set point changes of approximately 5 to 6 inches (12.5 to 15.3 cm) will be used to evaluate and adjust the feedwater control system settings for all power and feedwater pump modes. The level set point changes will also demonstrate core stability to subcooling changes.

Under normal operation with one feedwater loop in manual, a manual flow step will be initiated and each loop's flow response will be determined.

One of the operating feedwater pumps will be tripped and the automatic flow runback circuit will act to drop power to within the remaining capacity of the system.

### 14.2.12.3.23.4 Criteria

<u>Level 1</u>

The decay ratio must be less than 1.0 for each process variable that exhibits oscillatory response to feedwater system changes.

The maximum feedwater temperature decrease for the feedwater heater loss test must be less than or equal to 100°F.

The other principal control systems will be in their normal operating mode for the given test condition, In addition at test conditions 3, 5 and 6 the level setpoint changes will be performed with the recirculation control system in the local normal control mode.

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14.2.12.3.24 Test Number 24 - Turbine Valve Surveillance

14.2.12.3.24.1 Purpose

The purpose of this test is to demonstrate Periodic the acceptable procedures and maximum power levels for ~~daily~~ surveillance testing of the main turbine throttle, governor, interceptor, reheat stop and bypass valves without producing a reactor scram.

14.2.12.3.24.2 Prerequisites

The preoperational tests have been completed, the POC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

14.2.12.3.24.3 Description

Individual main turbine control, stop and bypass valves are tested routinely during plant operation as required for turbine surveillance testing. At several test points the response of the reactor will be observed, and although it is not required, it is recommended that the maximum possible power level for performance of these tests along the 100% load line be established. First actuation should be between 45 and 65% power, and used to extrapolate to the next test point between 70 and 90% power and ultimately to the maximum power test condition with ample margin to scram. Note proximity to APRM flow bias scram point. Each valve test will be manually initiated and reset. Rate of valve stroking and timing of the close-open sequence will be such that the minimum practical disturbance is introduced. If the stroking and timing rates are changed - the test will be repeated to ensure acceptable performance of the periodic surveillance test.

Level 1

The decay ratio of any oscillatory response must be less then 1.0.

Level 2

a. Peak neutron flux must be at least 7.5% below the scram trip setting. Peak vessel pressure must remain at least 10 psi below the high pressure scram setting.

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14.2.12.3.25.4 Criteria

Level 1

Individual Valve Closure

MSIV closure time, exclusive of electrical delay shall be no faster than 3.0 seconds (average of the fastest valve in each steam line) and no slower than 5.0 seconds, (each valve, not averaged). The electrical time delay at 100% open shall be less than or equal to 0.5 second and the fastest valve closure time shall be $\geq$2.5 seconds.

Full Reactor Isolation

The positive change in vessel dome pressure occurring within 30 seconds after closure of all MSIV valves must not exceed the Level 2 criteria by more than 25 psi. The positive change in simulated heat flux shall not exceed the Level 2 criteria by more than 2% of rated value.

Feedwater control system settings must prevent flooding of the steam lines.

Level 2

Individual Valve Closure

During full closure of individual valves peak vessel pressure must be 10 psi (0.7 kg/cm$^2$) below scram, peak neutron flux must be 7.5% below scram, an steam flow in individual lines must be 10% below the isolation trip setting. The peak heat flux must be 5% less than its trip point.

Full Reactor Isolation

The RCIC system shall adequately main water level. The relief valves must reclose properly (without leakage) following the pressure transient.

The positive change in vessel dome pressure and simulated heat flux occurring within the first 30 seconds after the closure of all MSIV must not exceed the predicted values. These values will be referenced to actual test conditions of initial power level and dome pressure and will use BOL nuclear data. In addition it will be corrected for the measured control rod insertion speed and the time from the start of MSIV motion to the start of control rod motion.

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14.2.12.3.27 Test Number 27 - Turbine Trip and Generator Load Rejection

14.2.12.3.27.1 Purpose

The purpose of this test is to demonstrate the response of the reactor and its control systems to protective trips in the turbine and generator.

14.2.12.3.27.2 Prerequisites

The preoperational tests have been completed, the POC has reviewed and approved the test procedures and initiation of testing. All controls and interlocks are checked and instrumentation calibrated. The plant electrical system will be aligned in the normal mode for the operating condition at which the test is performed.

14.2.12.3.27.3 Description

The turbine throttle valves will be tripped at selected reactor power levels and the main generator breaker will be tripped in such a way that a load imbalance trip occurs. Several reactor and turbine operating parameters will be monitored to evaluate the response of the bypass valves, relief valves, and reactor protection system (RPS). Additionally, the peak values and change rates of reactor steam pressure and heat flux will be determined. The effect of recirculation pump overspeed, if any, will be checked during the generator load rejection. The ability to ride through a load rejection within bypass capacity without a scram will be demonstrated.

14.2.12.3.27.4 Criteria

Level 1

For Turbine and Generator trips there should be a delay of less than 0.1 seconds following the beginning of control or stop valve closure before the beginning of bypass valve opening. The bypass valves should be opened to a point corresponding to greater than or equal to 80 percent of their capacity within 0.3 seconds from the beginning of control or stop valve closure motion.

Feedwater system settings must prevent flooding of the steam line following these transients.

The two pump drive flow coastdown transient during the first three seconds must be equal to or faster than that specified in Test 30.

The positive change in vessel dome pressure occurring within 30 seconds after either generator or turbine trip must not exceed the Level 2 criteria by more than 25 psi.



For the turbine trip, the main generator breakers remain loaded for a time so there is no rise in turbine generator speed, whereas, in the generator trip, the main generator breaker opens and the residual turbine steam will cause a momentary rise in the generator speed.

Level 2

The MSIV shall not be tripped closed at anytime during the test transients.

The positive change in vessel dome pressure and in simulated heat flux which occur within the first 30 seconds after the initiation of either generator or turbine trip must not exceed the predicted values.

Predicted values will be referenced to actual test conditions of initial power level and dome pressure and will use BOL (Beginning of Life) nuclear data. In addition the predictions will be corrected for the measured control rod insertion time and the delay from beginning of turbine control valve or stop valve motion to the generation of the scram signal. The predicted pressure and heat flux will be corrected for the actual measured values of these two parameters.

) For the Generator trip within the bypass valves capacity, the reactor shall not scram for initial thermal power values within that bypass valve capacity.

~~must meet RTS specifications.~~

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423.232



### 14.2.12.3.29 Test Number 29 - Recirculation Flow Control

#### 14.2.12.3.29.1 Purpose

The purposes of this test are a) to demonstrate the core flow system's control capability over the entire flow control range, including valve position, core flow, neutron flux and load following modes of operation, and b) to determine that all electrical compensators and controllers are set for desired system performance and stability.

#### 14.2.12.3.29.2 Prerequisites

The preoperational tests have been completed, the POC has reviewed approved the test procedures and initiation of testing. All controls are checked and instrumentation calibrated.

#### 14.2.12.3.29.3 Description

The testing of the Recirculation Flow Control System follows a "building block" approach while the plant is ascending from flow to high power levels. Components and inner control loops are tested first, followed by drive flow control and plant power maneuvers to adjust and then demonstrate the outer loop controller performance. By far the most extensive testing will be performed along the mid power load line where most of the systems final adjustments are determined. The core power distribution will be adjusted by control rods to permit broader range of maneuverability with respect to Preconditioning Cladding Interim Operating Management Recommendation (PCIOMR). In general, the controller dials and gains will be raised to meet the maneuvering performance objectives. Thus the system will be set to be the slowest that will perform satisfactorily, in order to maximize stability margins and to minimize equipment wear by avoiding controller overactivity. The other principal control system will be in their normal operating mode for the given test condition. Because of PCIOMR power maneuvering rate restrictions, the fast flow maneuvering adjustments are performed along a mid power rod line, and an extrapolation made to the expected results along the 100 percent rod.

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423.23 m

Revised pg 14.2-138



14.2.12.3.30 Test Number 30 - Recirculation System

14.2.12.3.30.1 Purpose

The purposes of this test are a) to determine transient responses and steady-state conditions following recirculation pump trips at selected reactor power levels, b) to obtain recirculation system performance data, c) to verify that no recirculation system cavitation occurs on the operable region of the power/flow map, and d) to verify that the feedwater control system can satisfactorily control water level without a resulting turbine trip scram following recirculation pump trips.

14.2.12.3.30.2 Prerequisites

The preoperational tests have been completed and the POC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate. All control systems will be in their normal operating mode for the given test conditions.

14.2.12.3.30.3 Description

Single recirculation pump trips with LFMG set breaker locked open will be made at ~~designated~~ near rated flow at 75% and 100% power levels, by opening 6190 VAC breaker. Simultaneous trip of both recirculation pumps without transfer to LFMG set power supply will be made at near rated flow and power levels, by using the two pump circuit trip system. Reactor operating parameters will be recorded during the transient and at steady-state conditions to experimentally determine the power and flow coastdown rates.

Both the jet pumps and the recirculation pumps will cavitate at conditions of high flow and low power where NPSH demands are high and little feedwater subcooling occurs. However, the recirculation flow will automatically runback upon sensing a decrease in feedwater flow, to lower the reactor power. The maximum recirculation flow is limited by appropriate stops which will run back the recirculation flow from the possible cavitation region. It will be verified that these limits are sufficient to prevent operation where recirculation pump or jet pump cavitation occurs.

14.2.12.3.30.4 Criteria

Continued on attached page

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Level 1

The two pump drive flow coastdown transient during the first 3 seconds must be equal to or greater than that specified on Figure ~~30.2-1~~. 14.2-6.

Level 2

The reactor water level margin to avoid a higher level trip shall be >3.0 inches during the one pump trip.

The simulated heat flux margin to avoid a scram shall be $\geq$5.0 percent both during the one pump trip and also during the recovery.

The APRM margin to avoid a scram shall be >7.5% during the one pump trip recovery.

Runback logic shall have settings adequate to prevent operation in areas of potential cavitation.

Additional FSAR Figure

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423.23 m

Figure 14.2-6

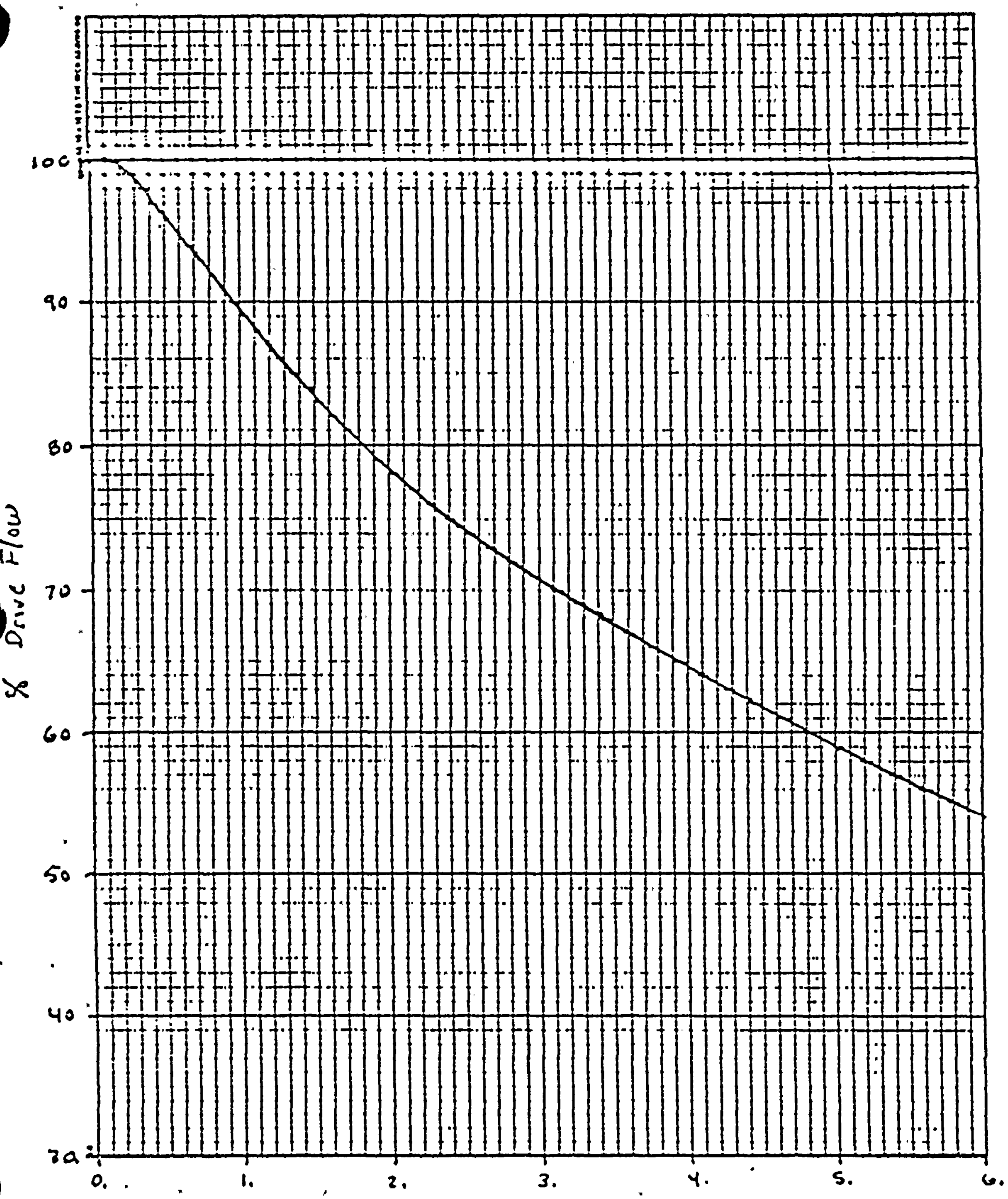


Figure 14.2-6.

Maximum Acceptable Drive Flow Response

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### 14.2.12.3.31 Test Number 31 - Loss of Turbine-Generator and Offsite Power

#### 14.2.12.3.31.1 Purpose

The purpose of this test is to determine the reactor transient performance during the loss of the main generator and all offsite power, and to demonstrate acceptable performance of the station electrical supply system.

#### 14.2.12.3.31.2 Prerequisites

The preoperational tests have been completed, the POC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate. The plant electrical system will be aligned in the normal mode for the operating condition at which the test is performed

#### 14.2.12.3.31.3 Description

The loss of auxiliary power test will be performed at 20 to 30% of rated power. The proper response of reactor plant equipment, automatic switching equipment, and the proper sequencing of the diesel generator load will be checked. Appropriate reactor parameters will be recorded during the resultant transient.
The trip will be initiated by tripping the main turbine and opening the breakers supplying offsite power or preventing the closure of breakers supplying offsite power.

#### 14.2.12.3.31.4 Criteria

Level 1

Reactor pressure shall be maintained below the set point of the first safety valve, during the transient following the loss of the main generator and all offsite power. All safety systems, such as the Reactor Protection System, the diesel-generator, RCIC and HPCS, must function properly without manual assistance.

Level 2

The maximum reactor pressure should be less than 40 psi (2.8 $kg/cm^2$) below the first safety valve set point, during the transient following the loss of the main generator and all offsite power. Normal reactor cooling systems should be able to maintain adequate suppression pool water temperature, maintain adequate drywell cooling, and prevent actuation of the auto-depressurization system.

The response of the plant electrical system and equipment for load-shedding and load alignment following the initial transient will be in accordance with the system design. (Included in the test procedure will be detailed tables of breaker position and operating equipment which indicate before and after status).

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14.2.12.3.36 Test Number 70 - Reactor Water Cleanup System

14.2.12.3.36.1 Purpose

The purpose of this test is to demonstrate specific aspects of the mechanical operability of the Reactor Water Cleanup System. (This test, performed at rated reactor pressure and temperature, is actually the completion of the preoperational testing that could not be done without nuclear heating.)

14.2.12.3.36.2 Prerequisites

The preoperational tests have been completed, the POC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

14.2.12.3.36.3 Description

With the reactor at rated temperature and pressure process variables will be recorded during steady state operation in three modes as defined by the System Process Diagram: Blowdown, Hot Standby, and Normal.

14.2.12.3.36.4 Criteria

<u>Level 1</u>

Not applicable.

<u>Level 2</u>

The temperature at the tube side outlet of the non-regenerative heat exchangers shall not exceed 130°F in ~~any mode~~ the blowdown mode and shall not exceed 120°F in the normal mode.
The pump available NPSH will be 13 feet or greater during the hot standby mode defined in the process diagrams.

The cooling water supplied to the non-regenerative heat exchangers shall be within the flow and outlet temperature limits indicated in the process diagrams. (This is applicable to "normal" and "blowdown" modes.)

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14.2.12.3.37 Test Number 71 - Residual Heat Removal System

14.2.12.3.37.1 Purpose

The purpose of this test is to demonstrate the ability of the Residual Heat Removal (RHR) System to: 1) remove heat from the reactor system so that the refueling and nuclear system servicing can be performed and 2) condense steam while the reactor is isolated from the main condenser.

14.2.12.3.37.2 Prerequisites

The preoperational tests have been completed, the POC has reviewed and approved the test procedures and initiation of testing. Instrumentation has been checked or calibrated as appropriate.

14.2.12.3.37.3 Description

With the reactor at ~~[illegible]~~ power, the condensing mode of the RHR system will be ~~tuned and~~ demonstrated. Condensing heat exchanger performance characteristics will be demonstrated. Final demonstration of the condensing mode will be done from an isolated condition. This test will optimize the controls for this mode of operation.

~~reactor cooldown, the shutdown cooling mode~~ During the first suitable reactor cooldown, the shutdown cooling mode of the RHR system will be demonstrated. Unfortunately the decay heat load is insignificant during the startup test period. Use of this mode with low core exposure could result in exceeding the 100°F/hr cooldown rate of the vessel if both RHR heat exchangers are used simultaneously, therefore ~~the~~ the demonstration is limited by the cooldown rate.

## 14.2.12.3.37.4 Criteria

Level 1

The transient response of any system-related variable to any test input must not diverge.

Level 2

The RHR system shall be capable of operating in the steam condensing, suppression pool cooling and shutdown cooling modes (with both one and two heat exchangers). ~~[illegible]~~ System-related variables may contain oscillatory modes of response. In these cases, the decay ratio for each controlled mode of response must be less than or equal to 0.25.

The time to place the RHR heat exchangers in the steam condensing mode ~~with~~ and COMMENCE ~~[illegible] stable~~ operation shall be one half hour or less.

JR 8/5/79
RoW 8/5/79

Question 423.24: Indicate the status or mode of operations of the plant control systems (i.e., either automatic or manual) for all transient tests. Provide acceptance criteria for the performance testing of the plant control systems. For example, modify the test descriptions of Items (f), (g), (l) and (m) of the preceding questions.

Response:

The response to questions 423.23 items (f), (g), (h), (l) and (m) and the associated revised test abstracts listed in the response discusses the mode of operations of the plant control systems and acceptance criteria.

Q. 423.25

In Section 5.2.2.4.1 of the FSAR, you state that the preoperational and startup testing of the safety/relief valves will include monitoring of the discharge line movement. Modify the startup test description to reflect this commitment.

Response

The statement in Section 5.2.2.4.1 refers to monitoring of thermal growth of the discharge lines. A description of Startup Test Number 17-System Expansion is contained in revised FSAR Section 14.2.12.3.17.* During the development of the detailed Startup Test Procedure representative locations on the discharge lines will be selected and methods of monitoring movement will be determined.

*See the revised Section 14.2.12.3.17 attached to question 423.23.

Question 423.26: In Section 5.2.5.5.5 of the FSAR, you state that alarm points for the leak detection system will be determined analytically or will be based on measurements of appropriate parameters which will be monitored during the startup or preoperational tests. Modify the test description to identify these parameters and to indicate how you will establish the alarm points.

Response:

The preoperational procedure as described in section 14.2.12.1.16 will verify the alarm setpoints as specified for each instrument. However certain alarm setpoints are based on estimates of system operating characteristic and are initially set to conservative values. Therefore, these values may be revised once nominal operating conditions are established. Such setpoints and values will be identified during the development of the preoperational test and instructions will be included in the appropriate startup tests to verify these setpoints.

Q. 423.27

In Section 6.3.2.2.3 of the FSAR, you state that during preoperational testing of the low pressure core spray (LPCS) system, the size of the discharge flow orifice will be established to limit system flow to acceptable values as described in the LPCS process diagram. Modify the test description to reflect this commitment.

Response:

Revised Section 14.2.12.1.13 indicates that proper flow orifice sizing is verified during the preoperational test of the Low Pressure Core Spray System.*

*See attached draft page.

c. General Test Methods and Acceptance Criteria

Verification of low pressure core spray system capability is demonstrated by the proper integrated operation of the following:

(1) Logic and interlocks

(2) Low Pressure Core Spray system pumps, including auto initiation

(3) Flow path verification, including determination of system hydraulic performance to verify proper sizing of restricting orifice in LPCS discharge line to vessel, see section 6.3.2.2.3.

(4) Annunciators

(5) The time for initiation signal to full flow should be verified

(6) Photographs to prove acceptability of core spray patterns

14.2.12.1.14 High Pressure Core Spray System Preoperational Test

a. Purpose

To verify the operation of the High Pressure Core Spray (HPCS) System, including diesel generator and related auxiliary equipment, pumps, valves, instrumentation and control.

b. Prerequisites

The system lineup tests have been completed and the TWG has reviewed and approved the test procedure and the initiation of testing. The HPCS diesel generator must be installed and be operational.

c. General Test Methods and Acceptance Criteria

Verification of HPCS System capability is demonstrated by the proper integrated operation of the following:

Q. 423.28

Provide a preoperational test description for the various modes and systems for the fire protection system.

Response:

FSAR section 14.2.12.1.3 has been revised to be a description of the Fire Protection System Preoperational test.*

*See attached draft changes.

2. Reactor feedwater pumps, turbines and auxiliaries

3. Control logic

4. Annunciators and protective devices

14.2.12.1.2 Deleted

~~14.2.12.1.3 Deleted~~ [Insert attached]

14.2.12.1.4 Reactor Water Cleanup System Preoperational Test

a. Purpose

To verify the operation of the Reactor Water Cleanup (RWCU) System, including pumps, valves, and filter/demineralizer equipment.

b. Prerequisites

The system lineup tests have been completed and the TWC has reviewed and approved the test procedure and the initiation of testing. Filter aid, and anion and cation resin should be available. Closed Cooling Water (CCW) System and instrument air system must have readiness verification.

c. General Test Methods and Acceptance Criteria

Verification of the RWCU System capability is demonstrated by the proper integrated operation of the following:

(1) Drain flow regulator flow interlocks

(2) System isolation and logic

(3) Valve-operating sequence

(4) Pump operation and related control and logic

### 14.2.12.1.3 Fire Protection System Preoperational Test

a. Purpose

To verify the operation of the Fire Protection System, including the diesel engine, pumps, valves, detection and alarm circuits and control and instrumentation circuits. To verify the location and status of all portable equipment.

b. Prerequisites

The system lineup tests have been completed and the TWG has reviewed the procedure and the Startup and Operations Manager has approved the procedures and the initiation of testing. The circulating water system, control and service air system, and electrical distribution system are available to support operation.

c. General Test Methods and Acceptance Criteria

Verification of the Fire Protection System capability is demonstrated by the proper integrated operation of the following:

(1) Diesel engine and pump operation and related control and logic.

(2) Fire alarm and detection circuits.

(3) Fire control panel in the Main Control Room.

(4) Deluge, wet pipe and pre-action sprinkler systems.

(5) Carbon dioxide and Halon systems.

In addition portable equipment and hose station capability will be verified.

TABLE OF CONTENTS (Continued)

TABLE 14.2-1

PREOPERATIONAL TESTS

| Subsection Reference | Test Title | Page Reference |
|---|---|---|
| 14.2.12.1.1 | Reactor Feedwater System Preoperational Test | |
| 14.2.12.1.2 | Deleted | |
| 14.2.12.1.3 | ~~Deleted~~ Fire Protection System Preoperational Test | |
| 14.2.12.1.4 | Reactor Water Cleanup System Preoperational Test | |
| 14.2.12.1.5 | Standby Liquid Control System Preoperational Test | |
| 14.2.12.1.6 | Nuclear Boiler System Preoperational Test | |
| 14.2.12.1.7 | Residual Heat Removal System Preoperational Test | |
| 14.2.12.1.8 | Reactor Core Isolation Cooling System Preoperational Test | |
| 14.2.12.1.9 | Reactor Recirculation System and Control Preoperational Test | |
| 14.2.12.1.10 | Reactor Manual Control System Preoperational Test | |
| 14.2.12.1.11 | Control Rod Drive Hydraulic System Preoperational Test | |
| 14.2.12.1.12 | Fuel Handling and Vessel Servicing Equipment Preoperational Test | |
| 14.2.12.1.13 | Low Pressure Core Spray System Preoperational Test | |

Question 423.29: Provide a schedule relative to the fuel loading date, which identifies the scheduled time for performing the low power testing and the power ascension testing.

Response:

The present schedule indicates the following sequence time.

| Event | Time After Start of Fuel Load (Weeks) |
|---|---|
| Start fuel load | 0 |
| Complete fuel load | 2 |
| Initial critical/shutdown margin demonstration | 3 |
| Start of initial heatup | 5 |
| Start of testing above 25% power | 8.5 |
| Start of testing above 50% power | 10.0 |
| Start of testing above 75% power | 12.5 |
| Start of testing at 100% power | 14.5 |

Round One, Set Six

Operating License Branch

441.01 - 441.05

Q. 441.01 Provide a detailed description of the fire protection training and the subsequent retraining for the initial plant staff and for the replacement personnel in the manner described in paragraphs II.6.A through II.6.E of Section 13.2 (Revision 1) of the Standard Review Plan (SRP), NUREG-75/087.

Response:

## 1.0 FIRE BRIGADE TRAINING

### 1.1 Instruction

Each assigned member of the fire brigade will complete a basic fire brigade training course. The training will be conducted by knowledgeable and experienced personnel. The scope of the course will include the following:

a. Identification of the hazards and types of fires that could occur.

b. Identification, location, and operating features of manual, automatic, and portable fire protection equipment.

c. Indoctrination of the plants's fire fighting plan and responsibilities.

d. Proper use of all fire fighting equipment and proper procedures for postulated fires and conditions.

e. Access and egress routes in the plant.

f. Proper methods of fire control in confined areas, such as buildings and tunnels.

g. Changes in fire fighting plans, plant modifications which affect fire fighting, and familiarization with new fire fighting equipment.

One individual per shift will be designated responsible for directing the actual fire fighting forces. These individuals will receive training as necessary to carry out that function.

### 1.2 Practice

Each brigade member will attend a practice session at least once per year. Practice sessions will include:

a. Practice in extinguishing actual fires.

b. The use and wearing of self-contained breathing apparatus.

### 1.3 Drills

At least four times per year, per shift, fire brigade drills will be conducted. Each brigade member must participate in at least two of the four drills each year. At least one drill per year shall be unannounced. The drills may, or may not be, held in conjunction with other fire brigade practice and/or classroom training.

The offsite fire department will be requested to participate in at least one drill each year.

The drills will be designed so the brigade can practice as a team and to provide an assesment of the effectiveness, knowledge, ability, and response time of the fire brigade team.

## 2.0 OTHER PLANT EMPLOYEES

### 2.1 Instruction

Employees receiving an unescorted security clearance will be provided training to include orientation to the fire protection plans, evacuation signals and procedures, and the procedure for reporting a fire.

### 2.2 Drills

An annual evacuation drill will be held in which all employees will take part.

## 3.0 FIRE PROTECTION STAFF

Fire protection staff members will receive training, as appropriate, to supplement their experience. Such training may include:

a. Design and maintenance of protection systems.

b. Fire prevention techniques and procedures.

c. Training in manual fire fighting techniques and procedures for personnel, and fire brigade personnel.

## 4.0 OFF-SITE FIRE DEPARTMENT

The off-site fire department will be requested to attend and take part in a training practice session which covers basic radiation concerns and typical radiation hazards which may be encountered at the plant.

## 5.0 CONSTRUCTION PERSONNEL

Training will be provided to construction personnel in evacuation, alarm signals and procedures, fire reporting procedure and fire prevention techniques applicable to the task(s) they are to perform.

6.0 SECURITY PERSONNEL

Training will be provided to plant security personnel that address:

a. Entry procedures for off-site fire department.

b. Personnel control during emergency evacuation.

c. Basic fire hazard recognition.

Q. 441.02
(13.2.1)

Discrepancies, possible in terminology, exist between the training programs described in Section 13.2.1.1 of the FSAR and those cited in Figure 13.2-1. Correct the following discrepancies:

a. The Nuclear Energy training cited in Figure 13.2-1 is not described in Section 13.2.1.1.

b. The course entitled, "Basis Nuclear Theory" described in Section 13.2.1.1.1 is not cited in Figure 13.2-1.

c. The course entitled, "BWR Fundamentals," cited in Figure 13.2-1 is not described as such in Section 13.2.1.1.

d. The two week turbine-generator training course cited in Figure 13.2-1 is not described in Section 13.2.1.1.

Response:

a) & b) The title of Section 13.2.1.1.1 has been changed to correct the discrepancy.*

c) & d) See revised Sections 13.2.1.1.6 and 13.2.1.1.7.**

*Draft page changes are attached.
**Draft page changes attached to Question 441.03.

13.2.1.1 Cold License Program Description

The formal training program for Licensed Senior Reactor Operators (SRO) and Licensed Reactor Operators (RO) candidates will consist of the following courses. Successful completion of the requisite courses by each license candidate will be fully documented to substantiate his/her compliance with license prerequisites.

13.2.1.1.1 ~~Basic Nuclear Theory~~ Nuclear Energy Training

This is a ten week course conducted by WPPSS utilizing a video tape program produced by NUS Corporation. Course topics presented in this program include "Plant Mathematics and General Science", "Basic Nuclear Physics", "Reactor Operations", "Core Performance", "Radiation Protection", "Plant Chemistry" and "Instrument and Control". These presentations are supplemented by related lectures and problem working sessions.

Candidates with no previous nuclear experience shall complete this course satisfactorily. Candidates who have had nuclear experience at facilities not subject to licensing or who hold, or have held, a license for a comparable facility* may obtain waivers for all or part of the course by taking a precourse validation exam.

13.2.1.1.2 Research Reactor Training

This is a one week course conducted by university personnel at the Washington State University Nuclear Radiation Center. During this program, each student operates a TRIGA reactor through at least 10 startups and shutdowns. Other subjects covered, both through theoretical instruction and practical experience are:

a. reactor kinetics

b. rod calibration

c. health physics practices

*A comparable facility in this case is defined as any large, light water reactor power plant.

Q. 441.03

Provide the contingency plans for additional training for individuals to be licensed prior to criticality, should the fuel loading be delayed from the date presently indicated in the FSAR (i.e., March 1980). Revise the FSAR, as required, to indicate your best estimate of the fuel load date.

Response:

As stated in Section 13.2.1.3, a simulator refresher training course is provided in the event fuel loading is delayed. See revised Section 13.2.1.1.10 also.* No update will be provided here of the fuel load date. See Section 1.1.9 for the current fuel load date.**

*Draft page changes attached.
**For your information the revised fuel load date is March 1981.
Section 1.1.9 is currently being revised.

Candidates with no previous experience or with experience on a facility not subject to licensing shall attend this course. Those individuals who hold, or have held, licenses for a comparable facility or who have had extensive in-plant operating experience on a large power reactor will not be required to attend.

[Insert new 13.2.1.1.6 and .7 attached]

13.2.1.1.~~6~~8 NRC Exam Refresher

This is a three week training course conducted on site by General Electric just prior to the NRC Exam Analysis (13.2.1.1.7). The curriculum includes a review of reactor theory, features of facility design, general and specific operating characteristics, instrumentation and control, safety and emergency systems, operating procedures, radiation control and safety and administrative procedures. All license candidates shall attend this course.

13.2.1.1.~~7~~9 NRC Exam Analysis

This is a ten day course conducted on site by General Electric Training Engineers just prior to the scheduled NRC exam. All license candidates shall take typical NRC-type written exams plus oral plant and control room walk throughs. An analysis of the results will indicate specific individual areas of weakness and allow the license candidate to concentrate his/her efforts on those areas prior to the NRC exam.

13.2.1.1.~~8~~10 Simulator Refresher Training

This is a seven day course conducted by GE at the ~~BWR~~ TVA Training Center in ~~Morris, Illinois~~ Chattanooga, Tenn. The program is taught entirely in the control room and consists of BWR physics, transient response characteristics and application of WPPSS procedural requirements to an operating environment. Any licensed operator or senior operator candidate who has not attended simulator training, or who has not functioned as a licensed operator or senior operator at a comparable operating plant within twenty-four (24) months of WNP-2 fuel load shall attend this course. It is intended that this course will provide a refresher for license candidates in the event fuel load is delayed from the predicted date.

13.2.1.2 On-Site Training Program

13.2.1.2.1 BWR Systems Analysis

This is a five week course developed and conducted by WPPSS and designed to provide in-depth analysis of WNP-2 NSSS systems including design bases, flow paths, components, instrumentation, interlocks, system interactions and technical specifications.

### 13.2.1.1.6 BWR Fundamentals

This six day course conducted by the WNP-2 Training Coordinator provides previously licensed PWR Operators with experience in the operational characteristics of a BWR. The Fundamentals Class is divided into two sections. The first three day section is conducted on-site and consists of slides and lectures on the Dresden Nuclear Station Systems and prepares the students for participation in the second three day section which is conducted at the GE BWR Simulator in Morris, Illinois. The simulator program consists of twenty-four hours of control room exercises in which each student pulls at least two criticals, heats up the reactor, transfers to the run mode and manipulates the recirculation system controls to vary the reactor power levels. The curriculum also includes a demonstration of various transient conditions including turbine trips, scrams, feedwater casualties and loss-of-coolant conditions. It is intended that the BWR Fundamentals Class be taught after the students attend the BWR Systems Analysis Course (13.2.1.2.1) so the maximum training benefit will be realized.

13.2.1.1.7 Turbine Generator Operation and Maintenance

This three week course conducted on-site either by WPPSS or Westinghouse provides license candidates with approximately one week of classroom and hands on experience operating the turbine DEH control system. The training is accomplished prior to turbine operation by the use of a simulator which is hooked up to DEH panel. Operations which can be performed include turbine start up, synchronization and loading.

Q. 441.04
(13.2.2)

The reactivity manipulations performed to satisfy the requalification requirements of Section 3.a of Appendix A to 10 CFR Part 55 must be approved by the NRC staff. Manipulations for boiling water reactors which we find acceptable can be found in the American National Standard ANSI/ANS 3.1.1978, paragraph 5.5.1.2.1, "Control Manipulations."

Response:

Section 13.2.2.1.2 has been revised to identify the manipulationsin . the identified ANSI standard.*

*See revised draft page attached.

g. Radiation control and safety

h. Technical specifications

i. Applicable portions of Title 10, Chapter 1, Code of Federal Regulations.

The lecture series will be conducted by WNP-2 personnel using a combination of live lectures and videotape material. Live presentations will account for 50 percent or more of the lecture series. A minimum of six lectures per year will be scheduled. A topical examination on each section of the lecture series will be given in which a grade of 70 percent is considered passing.

13.2.2.1.2 Reactivity Manipulations

Each licensed operator shall, during each two year term of his/her license, perform a minimum of ten reactivity control manipulations which demonstrate skill and/or familiarity with reactivity control systems. Each licensed senior operator shall either manipulate the controls or direct the activities of others during at least ten reactivity control manipulations. These may consist of any combination of the following but reasonable effort will be made to provide a variety of reactivity changes for each operator. If necessary, an appropriate simulator may be used to meet the reactivity control requirements of the retraining program.

a. Plant or reactor startups to include a range such that reactivity feedback from nuclear heat addition is noticable.

b. Plant shutdowns

c. Control rod pattern changes

d. Shutdown margin checks

e. Control rod scram insertion time tests

f. Any manual reactor power change of 10% of full power or greater in manual speed control or control rod motion

g. Plant and reactor operation that involves emergency or transient procedures where reactivity is changing

h. Refueling operations where fuel is moved in the core

i. Operation of turbine controls in manual during startup

j. Manual feedwater control during startup or shutdown.

These manipulations are consistent with ANSI Standard ANS 3.1, 1978.

Q. 441.05
(13.2.2)

A statement should be included in the description of your program which indicates that the individuals who prepare and grade the Annual Retraining examination are exempt from taking the examination. A maximum of three licensed personnel may be exempted in this manner.

Response:

See revised Section 13.2.2.1.6.*

*Draft change attached.

13.2.2.1.5 Apparatus and Mechanism Operation

Each RO licensed operator performs and each SRO licensed operator directs or performs the operation of all apparatus and mechanisms, demonstrating knowledge of the operating procedures, in each area for which he is licensed during the term of his license. This may be accomplished by:

a. Actual manipulation of the apparatus and mechanism, or

b. Simulated walkthrough of the procedural steps required to start, stop or change conditions of the apparatus or mechanisms.

13.2.2.1.6 Retraining Evaluation Techniques

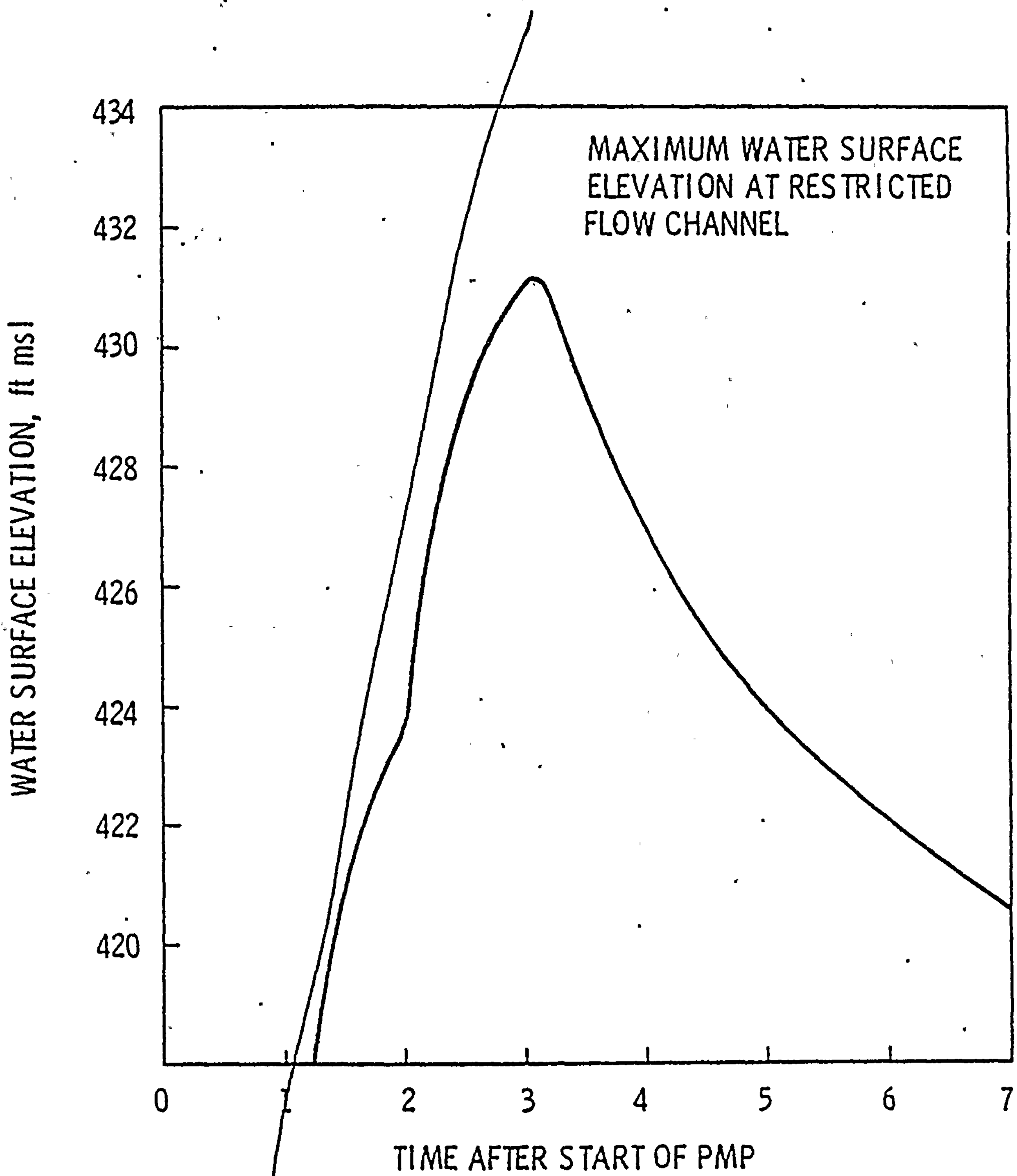
Annual written examinations are given to all RO and SRO licensed personnel to evaluate their proficiency and to determine areas in which retraining is required. The examinations are prepared and evaluated under the direction of the Training Coordinator. The examination will be sectioned in accordance with the nine requalification lecture topics required by Appendix A of 10CFR55. A minimum grade of 80 percent correct on any section will exempt an individual from required attendance at the retraining lectures pertinent to that section. Individuals who are licensed and who prepare and grade the annual retraining exam are exempt from taking the exam. A maximum of 3 licensed personnel may be exempted in this manner. An average grade of less than 70 percent correct on the overall annual examination requires an individual to be relieved of all licensed duties so he/she may participate in an accelerated retraining program. Reassignment to duties shall be allowed upon completion of remedial training in the individual's deficient areas and achievement of a grade of not less than 70 percent correct on examinations given over such areas.

Written examinations shall be given covering material presented in each retraining lecture series section. These examinations will stress abnormal and emergency operating procedures and changes to plant design and procedures. A grade of less than 70 percent will require the individual to attend the session again.

Annual on-the-job evaluations of the performance and competency of each licensed reactor operator and senior reactor operator shall be made by supervisors and/or training staff personnel. Included shall be a systematic observation and evaluation of actions taken or to be taken during actual or simulated abnormal and emergency conditions. It is intended that this

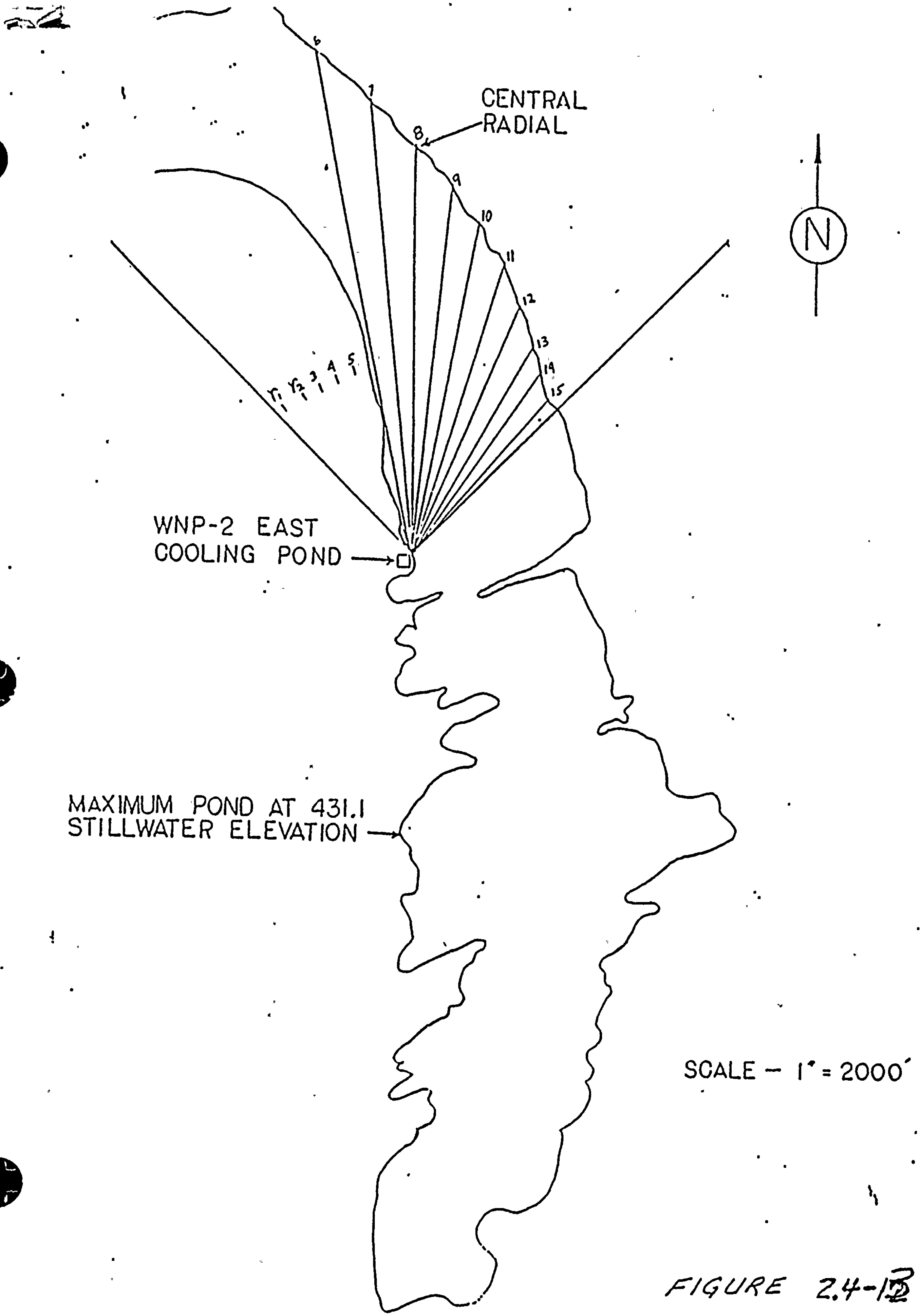
Figure 2.4-13

Deleted from March 21, 1979 submittal
G02-79-45




replace figure with attached

| WASHINGTON PUBLIC POWER SUPPLY SYSTEM<br>NUCLEAR PROJECT NO. 2 | MAXIMUM POND SURFACE ELEVATION<br>VS. TIME AFTER START OF PMP | FIGURE<br>2.4-13 |
|---|---|---|



CENTRAL RADIAL
WNP-2 EAST COOLING POND
MAXIMUM POND AT 431.1 STILLWATER ELEVATION
N
SCALE – 1" = 2000'


FIGURE 2.4-12