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SCHWENCER,A. Licensing Branch 2

SUBJECT: Forwards responses to Round 2 questions.Remaining responses will be completed by 810904.Responses & applicable FSAR changes will be incorporated into Amend 19 of FSAR.

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Docket No. 50-397

August 7, 1981
G02-81-226
NS-L-02-CDT-81-027

Director, Nuclear Reactor
Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. A. Schwencer, Chief
Licensing Branch, No. 2
Division of Licensing

Subject: SUPPLY SYSTEM NUCLEAR PROJECT NO. 2
RESPONSES TO ROUND TWO QUESTIONS
REACTOR SYSTEMS BRANCH

Gentlemen:

Attached are sixty (60) copies of the responses to twelve of the remaining Reactor Systems Branch questions. These responses, and the applicable FSAR page changes will be incorporated into Amendment 19 of the FSAR.

The remaining Reactor Systems Branch questions responses are scheduled to be completed by September 4, 1981.

Very truly yours,

GD Bouchey

G. D. BOUCHEY
Director, Nuclear Safety

GDB/CDT/ldm

Enclosures

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WNP-2 Files

B001
S 1/1

Q. 211.110
(5.2.2)

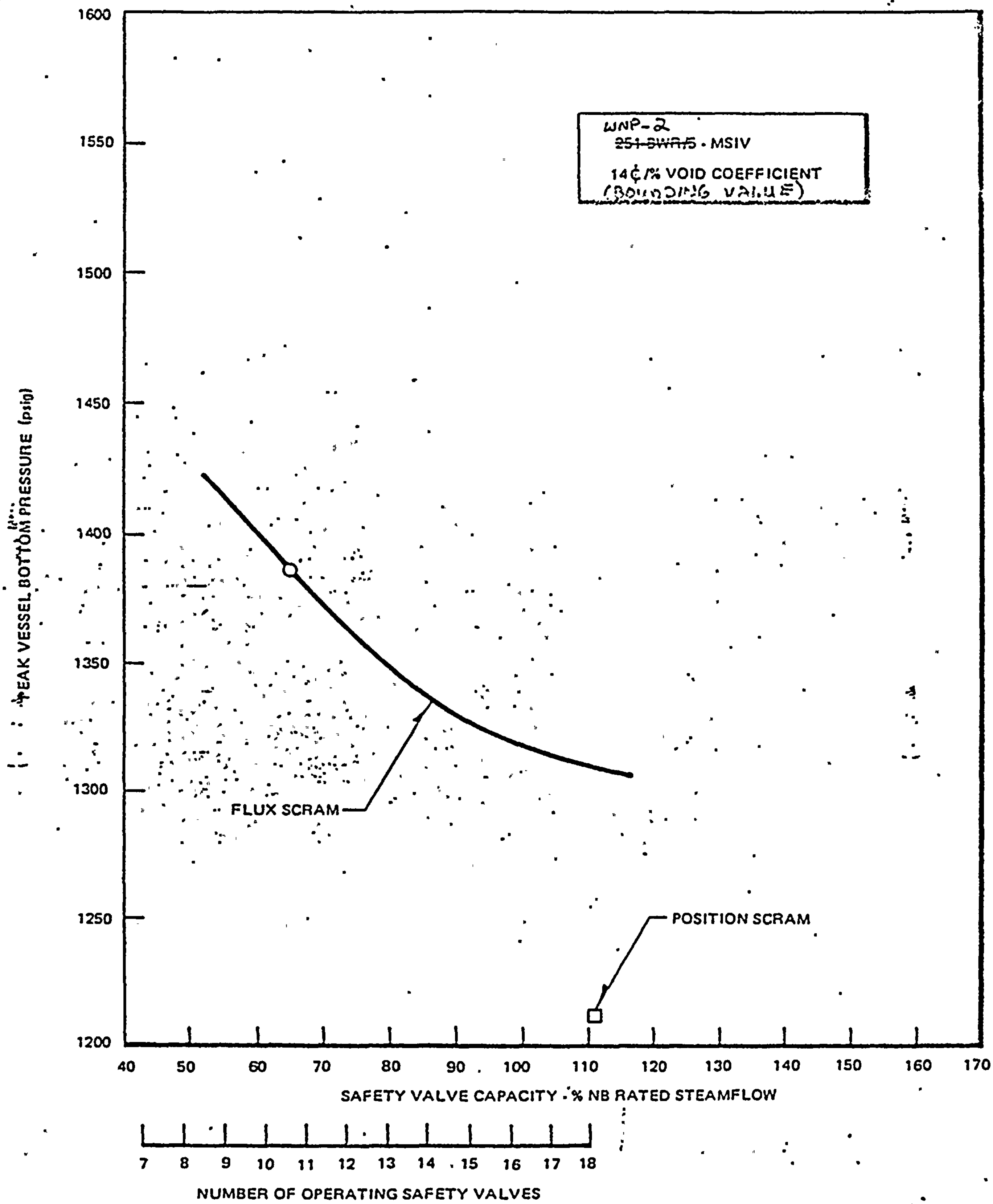
The notations, "251 BWR/5-MSIV, 140/% Void Coefficient" on Figures 5.2-4 and 5.2-5 indicate that these figures may be generic and not specifically for WNP-2. Confirm that these figures are applicable to WNP-2. If these curves are not applicable to WNP-2, complete the necessary analyses to provide data similar to that now presented on Figures 5.2-4 and 5.2-5.

Response:

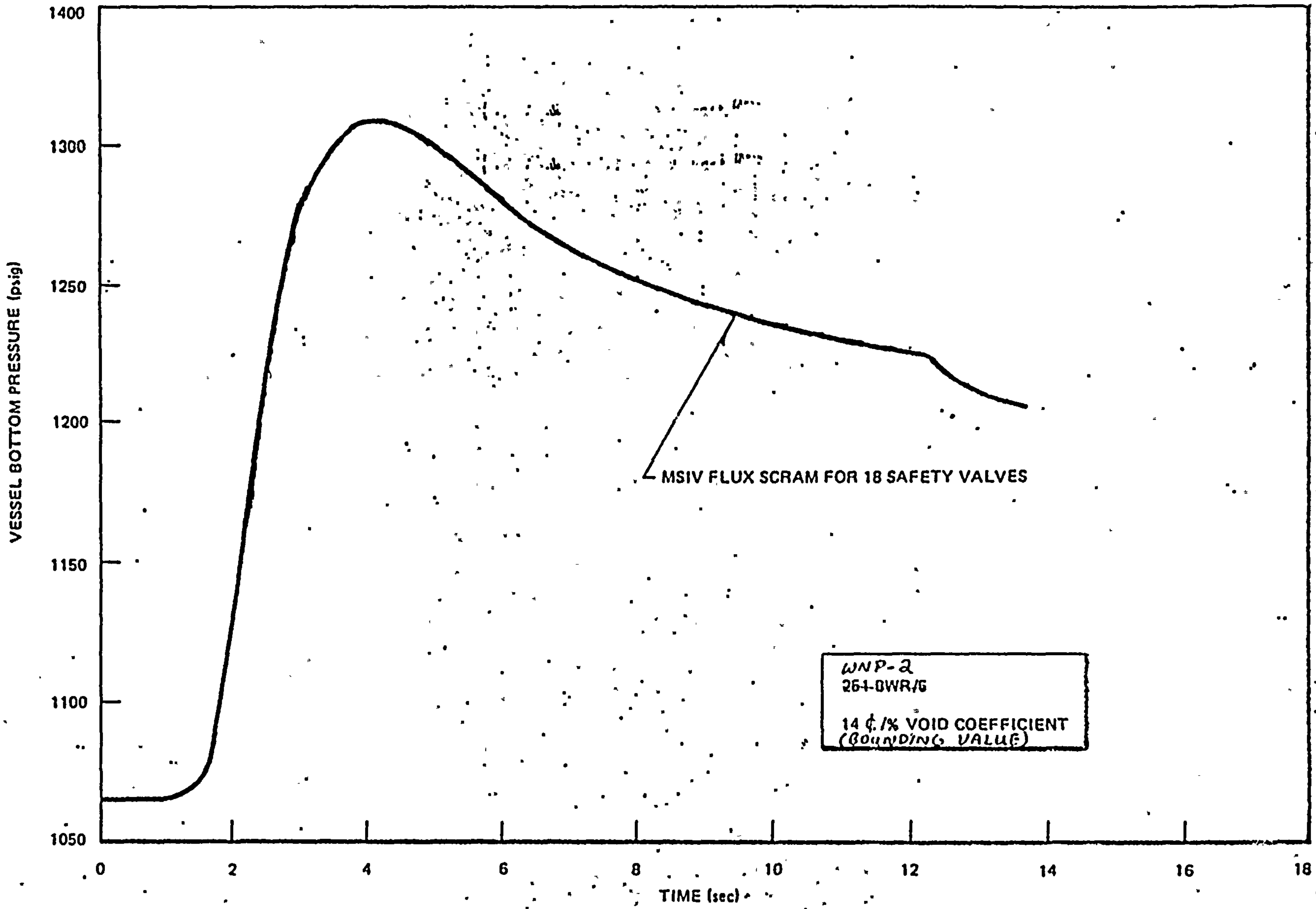
Figures 5.2-4 and 5.2-5 are not generic but do apply to WNP-2 and several other BWR/5 projects. The void reactivity coefficient of 140/% is a bounding value which is conservatively applied to the overpressure protection analysis. These figures have been changed to reflect their applicability to WNP-2.*

*Draft FSAR page change(s) attached.

80-19



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PEAK VESSEL PRESSURE VERSUS SAFETY/RELIEF CAPACITY	FIGURE 5.2-4



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

PEAK VESSEL PRESSURE VERSUS TIME FOR SAFETY VALVE CAPACITY SIZING TRANSIENT

FIGURE 5.2-5

AMENDMENT NO. 11
September 1980

Q. 211.112
(5.2.2)

Section 5.2.2.4.2.1 of the FSAR states that cyclic testing has demonstrated that the safety/relief valves are capable of at least 60 actuation cycles between required maintenance. Will the actuations of the safety/relief valves be recorded? If so, how will these data be recorded and reported to the NRC?

Response:

Our response to TMI requirement II.K.3.3, FSAR Appendix B, commits to implementing an administrative procedure for reporting all safety/relief valve failures promptly and all SRV challenges annually.

Q. 211.117
(5.2.2)

Resolve the following inconsistencies:

a) Figure 3.2-2 of the FSAR indicates in details B and C that the instrument air supply lines to the safety/relief valve air accumulators are safety class G (non-safety grade). Figure 9.3-2 shows these lines as safety class 2 or 3 (safety grade).

b) Figure 5.2-6 shows the safety/relief valves assigned to the automatic depressurization function are F013-M, -N, -P, -R, -S, -U, and -V. Figure 9.3-2 shows the dual accumulators used for the ADS valves assigned to safety/relief valves F013-D, -E, -H, -J, -M, -P, and -S.

Response:

a) There is no inconsistency between Figures 3.2-2 and 9.3-2. Details B and C on Figure 3.2-2 were not colored to show safety class, however, note 7d does indicate this piping is safety class 2 (code group B). The correct figure reference for safety classification for this piping is Figure 3.2-21. In addition, the colored version of Figure 9.3-2 has been replaced by a black and white figure which is periodically updated with the current construction drawings.

b) Figure 9.3-2 was updated in Amendment 11 and is consistent with the information provided in Figure 5.2-6. For purposes of reducing reproduction costs, the multi-colored drawings are revised in black and white only. The multi-colored drawings issued with a copy of the FSAR also include the following disclaimer on the page before the drawing:

Figure XX.XX-XX (Multi-colored)

For General Safety Class Reference Only

See Figure XX.XX-XX, Chapter XX

Q. 211.125
(5.2.5)

Standard Review Plan 5.2.5 specifies that unidentified leakage should be collected separately from the identified leakage so that a small unacceptable unidentified leak is not masked by larger acceptable identified leakage. Section 5.2.5 of the FSAR does not clearly indicate that separate collection of identified and unidentified leakage is provided.

Provide assurances that identified and unidentified leakage will be collected separately. If separate collection is not to be provided, provide justification for use of a common collection reservoir and show that a small unidentified leak of about 1 gpm would be recognized within one hour.

Response:

Identified and unidentified leakage are collected separately. Identified leakage is collected, monitored, and indicated by the Equipment Drain System (see FSAR Figure 3.2-9) while the unidentified leakage is collected, monitored, and indicated by the Floor Drain System (see FSAR Figure 3.2-10). Section 5.2.5.6 refers, in part, to 7.6.1.3 for further explanation. In subsections 7.6.1.3.4, 7.6.1.3.5 and 7.6.1.3.6 of this section, the two separate collection systems are described.

Q. 211.128
(5.4.7)

Subsection 5.4.7.1 of the FSAR states that spoolpiece interties are provided to permit the RHR heat exchangers to be used to supplement the fuel pool cooling system.

Describe the administrative controls that will be exercised for the use of these spoolpieces. What would be the effects if the spoolpieces were left in place and the RHR system were operated in any or all of the RHR modes of operation? Similarly, a spoolpiece is shown on Drawing M521 that connects the low pressure core spray (LPCS) system to the RHR loop A suction pipe. Describe the purpose of this intertie and, also, describe the effects on both the LPCS and RHR systems if the spoolpiece were inadvertently left in place. Are the same administrative controls used for the fuel pool cooling system spoolpiece used for the LPCS spoolpiece?

Response:

The fuel pool cooling and cleanup system is interconnected with the RHR system only in a shutdown condition. This cross-tie is used to supplement the pool cooling system during refueling in the event that a larger than normal batch of fuel is removed from the reactor. As stated in 9.1.3.3 of the FSAR, the RHR system will not be initiated to operate in parallel with the pool cooling and cleanup system unless the reactor is in a cold shutdown condition.

Administrative procedures which allow the operator to use this mode of operation require him to physically insert spool pieces in supply and discharge piping and open normally chain-locked closed valves (RHR-V-104, Figure 3.2-6 and FPC-V-141, Figure 9.1-4). This same procedure requires the operator to remove these spool pieces and chain-lock closed valves RHR-V-104 and FPC-V-141 before returning to the normal mode of operation.

Accordingly, it would require three independent violations of this procedure to maintain the lineup upon leaving this mode of RHR.

The purpose of the LPCS system crosstie with the RHR system is to provide a flowpath from the Condensate Storage Tanks to the LPCS system via RHR. This alignment is necessary to provide clean water to the LPCS system during initial flushing. It is also a source of supply for the LPCS preoperational test to provide a flowpath for suction from the reactor vessel

to the LPCS system and return to the vessel for the core spray sparger test. The flushing and testing requiring the use of this spoolpiece is complete before fuel load and the spoolpiece is expected to be used only once in the lifetime of the plant. The preoperational test procedure provides verification that the spoolpiece is removed after the core spray sparger test.

The administrative controls used for these spoolpieces are different but are both procedurally regulated to assure proper system function.*

*Draft FSAR page change(s) attached.

5.4.6.3 Performance Evaluation

The analytical methods and assumptions used in evaluating the RCIC system are presented in Chapter 15, "Accident Analyses," and Appendix A to Chapter 15, "Plant Nuclear Safety Operational Analyses." The RCIC system provides the flows required by the analysis (see Figure 5.4-10) within a 30 second interval based upon considerations noted in 5.4.6.2.4.

5.4.6.4 Preoperational Testing

The preoperational and initial startup test program for the RCIC system is presented in Chapter 14, "Initial Test Program."

5.4.6.4 Safety Interfaces

The Balance-of-Plant/GE Nuclear Steam Supply System safety interfaces for the Reactor Core Isolation Cooling System are: (1) preferred water supply from the condensate storage tanks; (2) all associated wire, cable, piping, sensors, and valves which lie outside the Nuclear Steam Supply System scope of supply; and (3) air supply for testable check and solenoid actuated valve(s).

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

5.4.7.1 Design Bases

The RHR system is comprised of three independent loops. Each loop contains its own motor-driven pump, piping, valves, instrumentation and controls. Each loop has a suction source from the suppression pool and is capable of discharging water to the reactor vessel via a separate nozzle, or back to the suppression pool via a full flow test line. In addition, the A and B loops have heat exchangers which are cooled by standby service water. Loops A and B can also take suction from the reactor recirculation system suction, and can discharge into the reactor recirculation discharge or to the suppression pool and drywell spray spargers. Spool piece interties are provided to permit the RHR heat exchangers to be used to supplement the cooling capacity of the fuel pool cooling system. ~~Spool piece interties are provided to permit the RHR heat exchangers to be used to supplement the cooling system.~~ remove The A and B loops also have connections to reactor steam via the RCIC steam line and can discharge condensate to the RCIC pump suction or to the suppression pool. LaSalle 1 and 2, and Zimmer 1 are nuclear plants which employ similar RHR systems and which are in the process of being licensed.

5.4.7.1.1 Functional Design Basis

The RHR system has five subsystems; each of which has its own functional requirements. Each subsystem will be discussed

Q. 211.143
(5.4.6.4)

Show how the preoperational initial startup test programs for the RCIC system in Section 14.2.12.1.8 meet the intent of applicable sections in Regulatory Guide 1.68.

Response:

The applicable sections of Regulatory Guide 1.68 which delineate requirements for tests of RCIC include sections 1.d (5) and (6); 1.j (19); 4.k and q; 5.l, dd and mm of Appendix A.

The specific areas of concern that these sections address are, respectively: verification of operability and design features of the RCIC system and the RHR/RCIC system interface in the steam condensing mode during the preoperational phase of the WNP-2 initial startup test program; operability and design verification of the RCIC control instrumentation on the remote shutdown panel again during the preop program; demonstration of RCIC and RHR steam condensing mode operability during low power operation when sufficient steam exists to utilize these plant design features; and finally to demonstrate the design capability of RCIC during major plant transients such as the remote shutdown capability demonstration and the main steam line isolation valve (MSIV) full isolation test.

The WNP-2 initial startup test program provides for extensive tests in each of these areas. FSAR sections 14.2.12.1.8 and 26, 14.2.12.3.14, 25, 28 and 37 briefly describe, in general terms, the tests which will be performed to provide assurance that the RCIC system is fully operational in each of its modes or conditions in which it is expected to perform. Specifically, during the preop phase such RCIC component tests as valve operability, initiation/interlock/trip logic checks, flow path verification, control and instrumentation calibration and pump/turbine vibration measurements are conducted. In addition, the control and instrumentation calibration on the remote shutdown panel and the system interface with RHR in the steam condensing mode are checked for proper operation. During low power operation the ability of the RCIC system to initiate, then deliver rated flow within 30 seconds is demonstrated at three points within the range of 150 psig to rated reactor pressure. Also, following tune-up of the

RHR heat exchanger level and inlet pressure controllers, the adequacy of the RCIC control system is confirmed when the system is coupled with the RHR system in the steam condensing mode. The final confirmation of proper RCIC system performance is achieved by challenging the system to perform during anticipated transients. The ability of RCIC to maintain reactor water level when controlled from the remote shutdown panel is demonstrated by actual testing. The ability of the system to meet its primary design function is demonstrated during the MSIV full isolation test when it is the main source of water for maintenance of vessel inventory.

The combination of component tests during the preop phase and the control system tune-up/overall operability demonstrations during the power ascension phase of the startup test program satisfy the requirement of Regulatory Guide 1.68.

Q. 211.144
(5.4.6)

The ASME Boiler and Pressure Vessel Codes, Section III, Article NB-7000 requires that individual pressure relief devices be installed to protect lines and components that can be isolated from normal system overpressurization protection. With reference to appropriate P&ID, identify those portions of the RCIC system that can be isolated from normal system overpressure protection. Discuss the relief devices provided or provide the basis for deciding that relief devices are not required.

Response:

Referring to FSAR Figures 5.4-9a and 5.4-9b, there are five RCIC pipe lines that have a low design pressure and, therefore, require relief devices or some other basis for addressing overpressure protection. They are:

1) RCIC Pump Suction Line
2) RCIC Turbine Exhaust Line
3) RCIC Steam Condensing Supply Line Downstream of F064
4) Portions of the RCIC Minimum Flow Line Downstream of F019
5) Portions of the RCIC Cooling Water Line Downstream of PCV-F015

The design pressure of the other major pipe lines is equal to the vessel design pressure and subject to the normal overpressure protection system. Below are the overpressure protection basis for the low pressure piping lines.

1) <u>RCIC Pump Suction Line</u>

A relief valve (F017) is located on the pump suction line on Figure 5.4-9b to accommodate any potential leakage through the isolation valves (F013 and F066). A high pump suction pressure alarm is provided in the control room. Also, the pump suction pipe is protected from overpressurization from the RHR system during steam condensing mode by F036 (reference FSAR Figure 5.4-13a) should both the RHR heat exchanger level control valves F065A and F065B (Figure 5.4-13a) fail open while dumping condensate to the RCIC pump suction.

2) RCIC Turbine Exhaust Line

This line is normally vented to the suppression pool and is not subject to reactor pressure during normal operation. Rupture discs D001 and D002, as shown on Figure 5.4-9b, are installed on this line to prevent exceeding piping design pressure should the exhaust line isolation valve F068 be closed when the RCIC turbine is operating. The RCIC system will automatically isolate if the rupture discs were to blow open.

3) RCIC Steam Condensing Supply Line Downstream of F064

In the steam condensing mode, high pressure steam is routed to the RHR heat exchangers via F064. The RHR piping is protected from overpressurization by relief valves F055 and F095 as discussed in Question 271.027.

4) Portions of the RCIC Minimum Flow Line Downstream of F019

This line is normally vented to the suppression pool and is separated from reactor pressure by the pump discharge isolation valves (F013, F065, and F066) and one additional normally closed isolation valve in the minimum flow line (F019) as shown on Figure 5.4-9a.

5) Portions of the RCIC Cooling Water Line Downstream of F019

This line is normally vented to the suppression pool and is separated from reactor pressure by the pump discharge isolation valves (F013, F065, and F066) and one additional normally closed isolation valve in the minimum flow line (F019) as shown on Figure 5.4-9a.

5) Portions of the RCIC Cooling Water Line Downstream of PCV-F015

In the standby condition this line is separated from reactor pressure by the pump discharge valves (F013, F065 and F066) and one additional normally closed shut-off valve in the cooling water line (F046) as shown on Figure 5.4-9b. During system operation a relief valve (F018) is provided to prevent overpressurizing piping, valves, and equipment in the coolant loop in the event of failure of pressure control valve PCV-F015 as shown on Figure 5.4-9b.

Q. 211.173
(15.1.3.2.1)

Add the "initial core cooling" safety action indicated in NSOA Figure 15.A.6-23 for the "pressure regulator failure-open" transient to event Table 15.1-4 for consistency.

Response:

The "initial core cooling" safety action on Figure 15.A.6-23 is the initiation of HPCS or RCIC. This is indicated on event Table 15.1-4.

Q. 211.174
(15.2)

The treatment of uncertainties associated with SRV setpoints appears to be handled in three different ways for the events associated with the sections shown below:

Section	Treatment of SRV Setpoint Uncertainties
15.2.3.3.4	Setpoints include errors (high) for all valves
15.2.4.3.4	Setpoints are assumed 15 psi higher than the valves nominal setpoint
15.2.5.3.4	Setpoints are assumed at upper limit of Technical Specifications for all valves

Explain this apparent discrepancy. If no discrepancy exists, standardize the wording between these sections for consistency.

Response:

There is no discrepancy - the setpoints are the same for each transient. Values used in the analyses are specified in Table 15.0-2, Item 25.

The wording has been standardized in each of the sections to read: "Setpoints are assumed 15 psi higher than the valves nominal setpoint."*

*Draft FSAR page change(s) attached.

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rod-out conditions is assumed.

c. Minimum specified valve capacities are utilized for overpressure protection.

d. Set points of the safety/relief valves include errors (high) for all valves.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip

Peak pressure in the bottom of the vessel reaches 1163 psig, which is below the ASME code limit of 1375 psig for the reactor cooling pressure boundary. Vessel dome pressure does not exceed 1136 psig. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

REPLACE

15.2.3.4.2 Turbine Trip with Failure of the Bypass

The safety/relief valves open and close sequentially as the stored energy is dissipated and the pressure falls below the set points of the valves. Peak nuclear system pressure reaches 1191 psig at the vessel bottom, therefore, the overpressure transient is clearly below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Peak dome pressure does not exceed 1163 psig.

15.2.3.4.2.1 Turbine Trip with Failure of Bypass at Low Power

Qualitative discussion is provided in 15.2.3.3.3.3.

15.2.3.5 Radiological Consequences

While the consequence of this event does not result in fuel failure it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since

Set points of the safety/relief valves are assumed to be 15 psi higher than the valve's nominal set point.

211.174-2

Set points of the safety/relief valves are assumed to be 15 psi higher than the valve's nominal set point.

A faster rate of loss of the condenser vacuum would reduce the anticipatory action of the scram and the overall effectiveness of the bypass valves since they would be closed more quickly.

Other uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rod-out conditions is assumed.

c. Minimum specified valve capacities are utilized for overpressure protection.

d. Set points of the safety/relief valves are assumed to be at the upper limit of technical specifications for all valves.

REPLACE

15.2.5.4 Barrier Performance

Peak nuclear system pressure is 1162 psig at the vessel bottom. Clearly, the overpressure transient is below the reactor coolant pressure boundary transient pressure limit of 1375 psig. Vessel dome pressure does not exceed 1135 psig. A comparison of these values to those for Turbine Trip with Bypass, 15.2.3, at high power shows the similarities between these two transients. The prime differences are the loss of feedwater and main steam line isolation, and the resulting low water level trips.

15.2.5.5 Radiological Consequences

While the consequence of this event does not result in fuel failures, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is contained in the primary containment, there will be no exposure to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under optimum meteorological and release conditions. If purging of the containment is chosen, the release will be in accordance with established technical specifications; therefore, this event, at worst, would only result in a small increase in the yearly integrated exposure level.

211.174-3

Q. 211.182
(15.2.9.2.1)

Revise Table 15.2-12 to indicate the time that suppression pool alarms are received, the Technical Specification limit is exceeded, and the maximum value of the suppression pool temperature is attained.

Response:

The Technical Specification limit and alarm set point relative to suppression pool cooling initiation at WNP-2 is 90°F. For analysis purposes only, the initial suppression pool temperature was assumed to be 95°F. The scenario for this transient is described in Table 15.2-12. A note has been added to Table 15.2-12 indicating that at time zero the suppression pool temperature is 95°F.* A note stating that the tech spec limit is 90°F has also been included to further substantiate the 10-minute operator action time. The time that the maximum suppression pool temperature is attained is given on Figure 15.2-18 and is approximately seven hours.

*Draft FSAR page change(s) attached.

TABLE 15.2-12

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING

Approximate Elapsed Time	Event
0	Reactor is operating at 105% NBR steam flow when LOP transient occurs initiating plant shutdown.
0	Concurrently loss of Division power (i.e., loss of one diesel generator) occurs.
0	Initial suppression pool temperature at 95°F*
10 min.	Suppression pool cooling initiated to prevent overheating from SRV actuation.**
10 min.	Controlled blowdown initiated.
2-3 hrs.	Blowdown to 100 psi completed.
2-3 hrs.	Personnel are sent in to open RHR shutdown cooling suction valve; this fails.
2½-3½ hrs.	Actuate ADS and complete blowdown to suppression pool.
2½-3½ hrs.	Redirect RHR pump discharge from pool to vessel via LPCI line. Alternate cooling path now established.
7 hrs.	Maximum suppression pool temperature attained. Refer to Figure 15.2-18.

* suppression pool tech spec limit is 90°F. The analysis assumes that the transient is initiated at 95°F for conservatism.

**See 15.2.6 for detailed sequence of events for loss of AC power transient.

Q. 211.201
(6.3)

Several plants have used sandbags or sand-filled tanks as biological shielding inside containment. In the event of a LOCA, these tanks or bags could be damaged and sand could be released. Release of sand inside containment could result in damage to the ECCS pumps. Identify any areas where sandbags or sand-filled tanks are used for biological shielding. What precautions would be taken to prevent ECCS damage if sand or similar material were released within containment?

Response:

WNP-2 does not use sandbags or sand-filled tanks (or similar material) for shielding inside containment.

Q. 211.208
(6.3)

Appendix A to Regulatory Guide 1.68, Revision 2, summarizes the systems to be tested and the performance capabilities that should be demonstrated by each BWR applicant during the preoperational and initial test programs.

It is unclear if the ECCS subsystems are tested using normal and emergency power supplies. Provide assurances that both the normal and emergency power supplies are used to verify ECCS operability.

If emergency power is not to be used in the operability tests, justify the exception to the criteria of Regulatory Guide 1.68, Revision 2.

Response:

Paragraph 6.3.4.1, ECCS Performance Tests, states in part, "Finally the entire system (ECCS) is tested for response time and flow capacity taking suction from its normal source and delivering flow into the reactor vessel. This last service of tests is performed with power supplied from both offsite (normal) power and onsite emergency power." In addition, paragraphs 14.2.12.1.7, 14.2.12.1.13, 14.2.12.1.14, 14.2.12.1.43, 14.2.12.1.48 and 14.2.12.1.50 outline individual system preop tests which culminate in the performance of the Loss of Power and Safety Testing preop (14.2.12.1.37). This latter test states as its purpose "to verify the integrated ability of the plant electrical distribution and safety systems to operate on normal and standby power sources during accident conditions" and 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". Accordingly, the Initial Test Program for WNP-2 meets the intent of the guidelines of Regulatory Guide 1.68, Revision 2 for this item. Test procedures defining the program described above will be available for NRC review.