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SUBJECT: Forwards responses to Containment Sys Branch 810316 request for addl info.Responses will be incorporated into FSAR amend within 4 months.Remainder of responses re hydrogen recombiner will be submitted to NRC by 810918.

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Docket No. 50-397

September 4, 1981
G02-81-269
NS-L-02-81-CDT-053

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

RECEIVED
SEP 11 1981
U.S. NUCLEAR REGULATORY COMMISSION

Dear Mr. Schwencer:

Subject: SUPPLY SYSTEM NUCLEAR PROJECT NO. 2 RESPONSES TO CONTAINMENT SYSTEMS BRANCH QUESTIONS

Reference: Letter, RL Tedesco to RL Ferguson, "Request for Additional Information Regarding the WNP-2 Facility (CSB)" dated March 16, 1981.

Enclosed are sixty (60) copies of responses to the Containment Systems Branch questions transmitted to the Supply System by the referenced letter. These responses will be incorporated into the FSAR in an amendment within four months. The remaining four hydrogen recombiner responses will be transmitted to the Nuclear Regulatory Commission by September 18, 1981.

Very truly yours,

G D Bouchey

G. D. Bouchey
Director, Nuclear Safety

GDB/CDT/ldm

Enclosure

cc: WS Chin - BPA
AD Toth - NRC RO
NS Reynolds - Debevoise & Liberman
J Plunkett - NUS Corporation
R Auluck - NRC NY
OK Earle - B&R RO
EF Beckett - NPI
WNP-2 Files

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WNP-2

Q. 211.006

The NRC changed this question number to 010.049.

WNP-2

Q. 211.107

The NRC changed this question number to 010.037.

Q. 211.108

The NRC changed this question number to 010.038.

Q. 211.109

The NRC changed this question number to 010.039.

Q. 211.111
(5.2.2)

Article NB-7200, Overpressure Protection, of the ASME Boiler and Pressure Vessel Code, Section III, requires that an overpressure protection report be provided. No overpressure report could be found in the FSAR. Provide this report.

Response:

Five copies of the Overpressure Protection Report are submitted via separate transmittal. The report is virtually reporduced verbatim in Section 5.2.2 of the FSAR. The response to Question 211.049 addresses the commitment to update this report to conform to a more recent analytical model (ODYN code) and to account for recirculation pump trip. WPPSS committed to reperform the applicable limiting transients in the response, and to update the FSAR.

Q. 211.114
(5.2)

Subsection 5.2.2.4.1 of the FSAR states that each Safety/Relief Valve is provided with a device to counteract the effects of backpressure which results in the discharge line when the valve is open and discharging steam. What type of device is provided? Describe the device and what effects would be anticipated if the device were to fail.

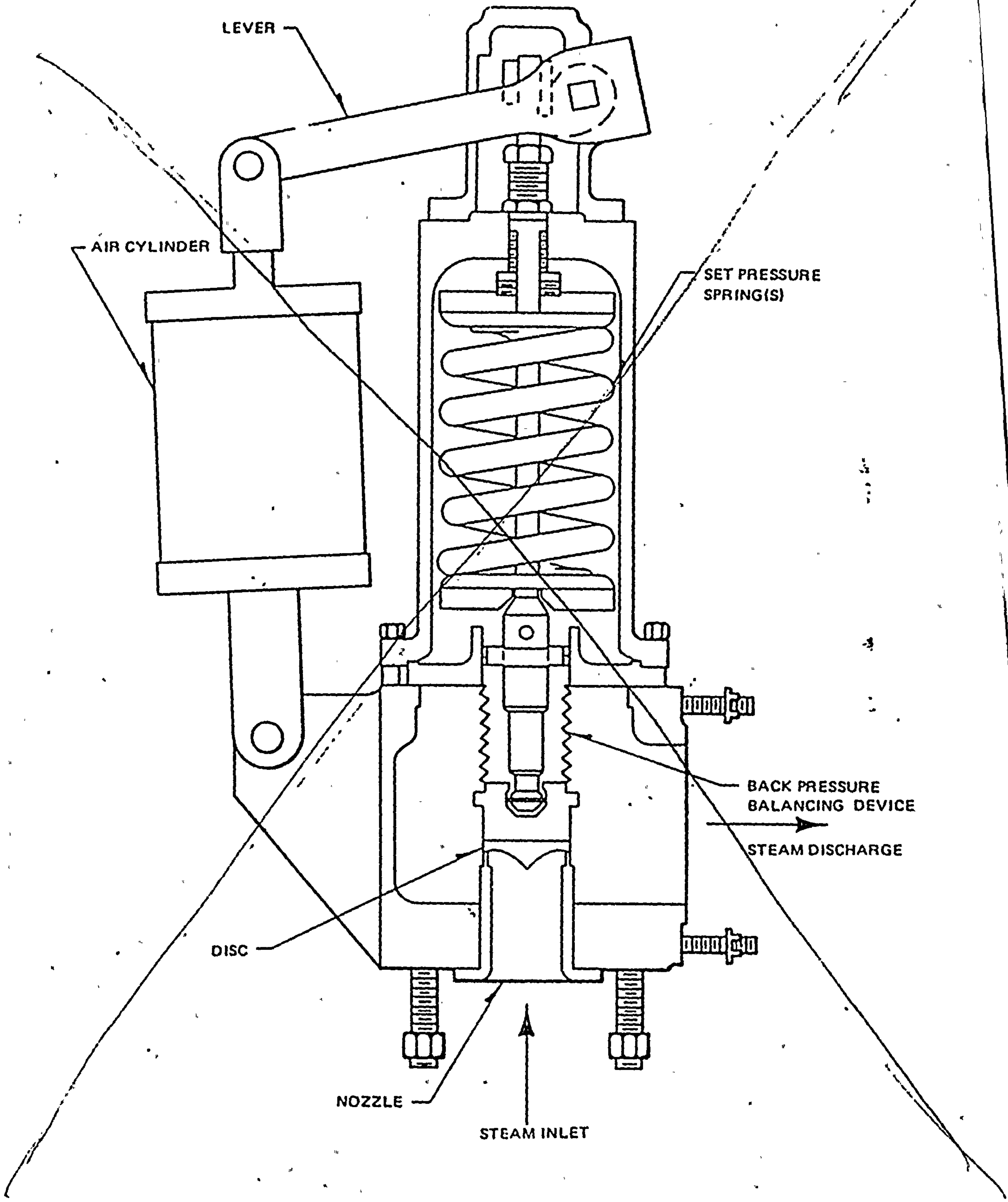
Response:

There is not a singular backpressure balancing device, but there is an integrated feature in each Safety/Relief Valve to counteract the effects of backpressure when the valve is open and discharging steam. To prevent this backpressure from affecting the valve's spring lift set point, each valve has a bellows and a balancing piston. The bellows isolates steam in the valve discharge chamber from the valves's internals, and prevents discharging steam from affecting the valve's set point. If the bellows fails, the balancing piston serves as a functional back-up by presenting an effective piston area to the back pressure equal to the valve seat area. This reduces the acting spring load on the disc insert by the amount of back pressure load additive to the spring set pressure load acting on the disc holder, thus balancing (neutralizing) it so that there is no net back pressure effect on the set (popping) point.

FSAR Figure 5.2-10 has been revised to show the bellows and balancing piston, and the FSAR text in Subsection 5.2.2.4.1 has been revised to include a reference to the bellows and balancing piston arrangement.*

*Draft FSAR page changes attached.

Replace with new Figure
(Attached)



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SCHEMATIC OF SAFETY VALVE WITH AUXILIARY ACTUATING DEVICE	FIGURE 5.2-10

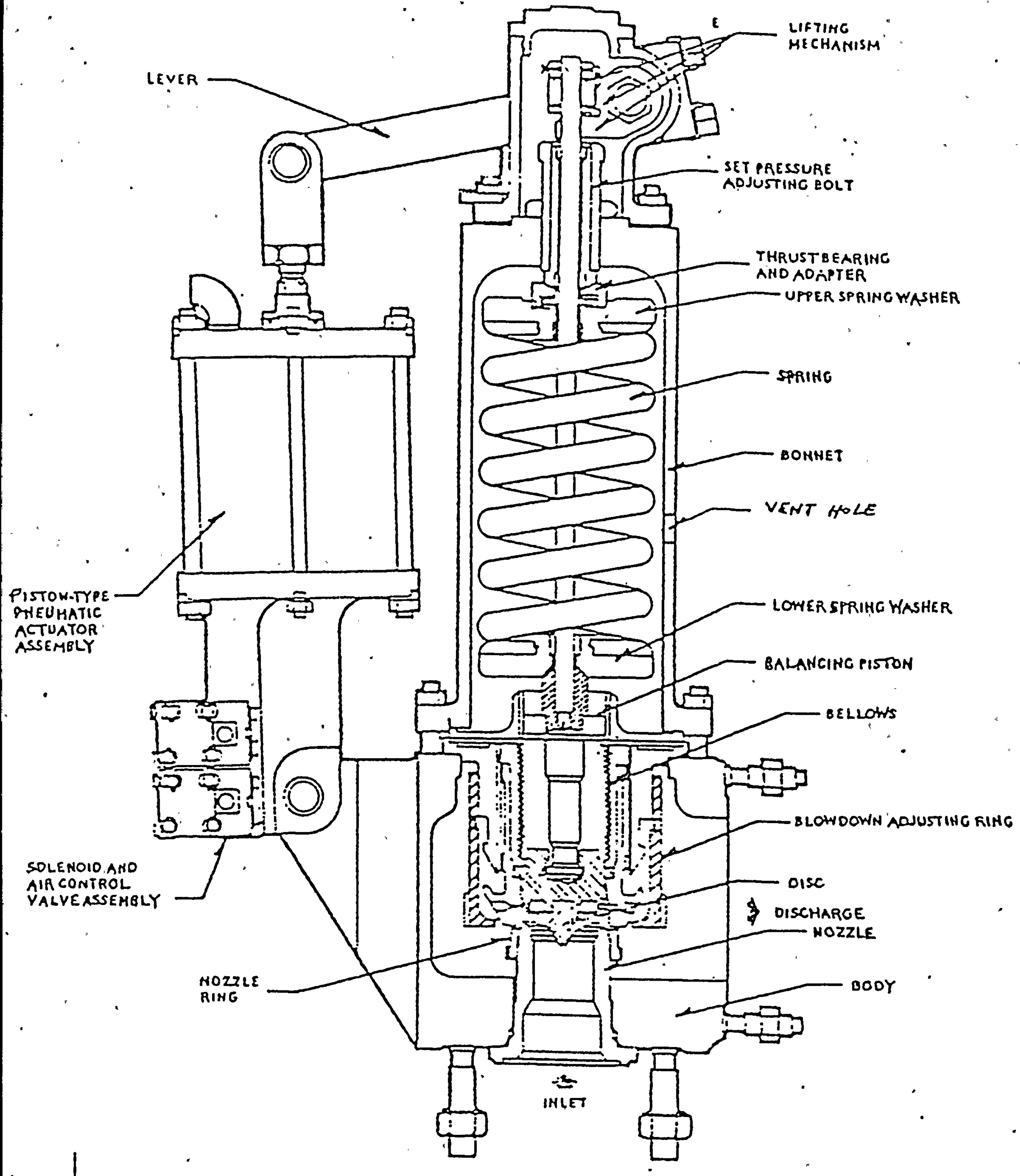


FIGURE 5.2-10

~~911.114-2~~

A schematic of the main safety/relief valve is shown in Figure 5.2-10. It is opened by either of two modes of operation:

a. The spring mode of operation which consists of direct action of the steam pressure against a spring-loaded disk that will pop open when the valve inlet pressure force exceeds the spring force. Figure 5.2-9 diagrams the valve lift vs. time characteristic.

b. The power actuated mode of operation which consists of using an auxiliary actuating device consisting of a pneumatic piston/cylinder and mechanical linkage assembly which opens the valve by overcoming the spring force, even with valve inlet pressure equal to zero psig.

The pneumatic operator is so arranged that if it malfunctions it will not prevent the valve disk from lifting if steam inlet pressure reaches the spring lift set pressure.

For overpressure safety/relief valve operation (self-actuated or spring lift mode), the spring load establishes the safety valve opening setpoint pressure and is set to open at setpoints designated in Table 5.2-2. In accordance with the ASME Code, full lift in this mode of operation is attained at a pressure no greater than 3% above the setpoint.

To prevent backpressure from affecting the spring lift setpoint, each valve is provided with ~~a device~~ a bellows and balancing piston to counteract the effects of backpressure which results in the discharge line when the valve is open and discharging steam. balancing it so there is no back pressure effect on the set point. (Fig 5.2-10)

The safety function of the safety/relief valve is a backup to the relief function described below. The spring-loaded valves are designed and constructed in accordance with ASME III, NB 7640 as safety valves with auxiliary actuating devices.

For overpressure relief valve operation (power actuated mode), each valve is provided with a pressure sensing device which operates at the setpoints designated in Table 5.2-2. When the set pressure is reached, it operates a solenoid air valve which in turn actuates the pneumatic piston/cylinder and linkage assembly to open the valve.

The bellows isolates steam in the valve discharge chamber from the valve's internals. If the bellows fails, the balancing piston serves as a functional back-up by presenting an effective piston area to the back pressure equal to the valve seat area. This reduces the acting spring load on the disc insert by the amount of back pressure load additive to the spring set pressure load acting on the disc holder, thus

Q. 211.121
(5.2.5)

Subsection 5.2.5.5.5 of the FSAR states that the leak detection system will satisfactorily detect unidentified leakage of 5 gpm. Subsection 7.6.2.4.2.1.2 states that the sensitivity and response time for each portion of the leak detection system for detection of unidentified leakage is one gallon per minute in less than one hour (excluding airborne systems). Resolve this inconsistency.

Response:

The information for sensitivity and response time of the leak detection system is now contained in subsection 7.6.2.4.b, rather than subsection 7.6.2.4.2.1.2.

The 5 gpm discussed in Section 5.2.5.5.5 refers to the maximum expected instantaneous leakge rate. The second paragraph on page 5.2-46 of subsection 5.2.5.5.5 has been corrected to read as follows:*

> The leak detection system sensitivity and response time is such that an unidentified leakage rate increase of one gpm in less than one hour will be detected.

*Draft FSAR page change attached.

Q211.121

The unidentified leakage rate limit is based, with an adequate margin for contingencies, on the crack size large enough to propagate rapidly. The established limit is sufficiently low so that, even if the entire unidentified leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before the integrity of the barrier would be threatened with significant compromise.

The leak detection system sensitivity and response time is such that an ~~will satisfactorily detect~~ unidentified leakage ~~of 5 gpm.~~ rate increase of one gpm in less than one hour will be detected.

Sensitivity, including sensitivity testing and response time of the leak detection system, and the criteria for shutdown if leakage limits are exceeded, are covered in Table 7.6-7.

5.2.5.6 Differentiation Between Identified and Unidentified Leaks

Section 5.2.5.1 describes the systems that are monitored by the leak detection system. The ability of the leak detection system to differentiate between identified and unidentified leakage is discussed in 5.2.5.1 and 7.6.1.3.

5.2.5.7 Sensitivity and Operability Tests

Testability of the leakage detection system is contained in 7.6.

5.2.5.8 Safety Interfaces

The Balance of Plant-GE Nuclear Steam Supply System safety interfaces for the leak detection system are the signals from the monitored balance of plant equipment and systems which are part of the nuclear system process barrier, and associated wiring and cable lying outside the Nuclear Steam Supply Equipment. These balance-of-plant systems and equipment include the main steam line tunnel, the safety/relief valves, and the turbine building sumps.

5.2.5.9 Testing and Calibration

Provisions for Testing and Calibration of the leak detection system are covered in Chapter 14, "Initial Tests and Operation".

Q. 211.122

The NRC changed this question number to 010.050.

Q. 211.123

The NRC changed this question number to 010.051.

Q. 211.125

The NRC changed this question number to 010.052.

WNP-2

Q. 211.126

The NRC changed this question number to 010.053.

Q. 211.130

The NRC changed this question number to 010.044.

Q. 211.131

The NRC changed this question number to 010.045.

Q. 211.133

The NRC changed this question number to 010.046.

Q. 211.134

The NRC changed this question number to 010.047.

Q. 211.135

The NRC changed this question number to 010.048.

Q. 211.138
(4.6.4.1)

Provide the common mode failure probability value for the control rod drive and the standby liquid control systems.

Response:

A Fault Tree Analysis was completed for both of these systems, and the calculated unreliabiltiy is less than 10^{-7}/reactor year. This unreliabiltiy is an estimate of the failure* to fully insert the control rods into the core, combined with a failure to inject boron into the vessel by the SLCS.

*Failure is defined to be non-insertion of CRDs in the following manner: >50% in a "checkerboard pattern", >31% in a random pattern, or >4% in a cluster.

Q. 211.150
(15.0)

Provide a listing of the transients and accidents in Chapter 15 for which operator action is required in order to mitigate the consequences. For corrective actions required prior to 20 minutes, provide justification.

Response:

For the design basis accident events (i.e., LOCA's) cited in Chapter 15, the required operator action and its justification are detailed in the responses to the staff Questions 211.59 and 211.65.

For all anticipated transients cited in Chapter 15, no operator action is assumed in less than 10 minutes to mitigate the consequences of the event or to prevent the plant from exceeding safety design limits. Operator action is allowed and utilized after 10 minutes in order to maintain the plant:

a) In a steady state condition;

b) Initiate safe and orderly shut-down;

c) Maneuver plant from condition that would necessitate safety action; or

d) Reduce the impact on plant system operation due to a single operator error or a single equipment malfunction.

In no case would the operator's action or non-action result in an unacceptable effect on the health and safety of the general public. This operator action for transients certainly is justifiable since it is his (or her) normal operational assignment.

Q. 211.152
(15.0)

In relation to Figure 15.0-2, confirm the following items for all transients in Chapter 15.0 which require control rod insertion to prevent or lessen plant damage.

a) The scram curve used in Chapter 15.0 analyses (Figure 15.0-2) has a total reactivity worth of $37.05 and is the nominal conservatism factor of 0.8.

b) The slowest allowable scram insertion speed was for the scram curve applied to Chapter 15.0 analyses.

c) The end of cycle 1 scram curve has a total reactivity worth of $40.21 and is identified incorrectly in Figure 15.0-2.

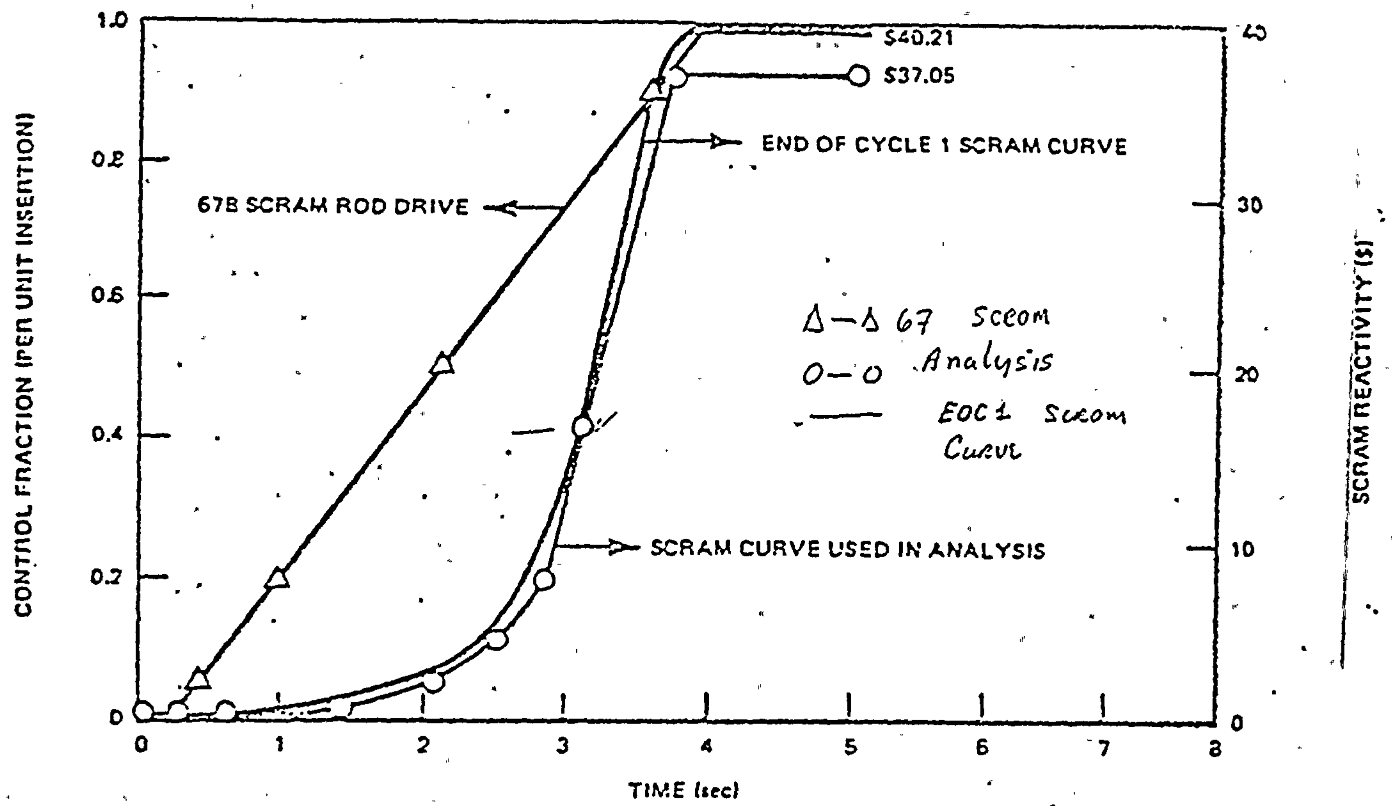
Response:

a) The scram curve used in Chapter 15.0 analyses with a total reactivity worth of $37.05 is quite conservative compared to the nominal scram curve. For any transient, since the neutron flux would drop significantly within 2 to 3 seconds after scram, the excess negative reactivity introduced after this short period of time has negligible effect on the peak values of the important parameters, e.g., neutron flux, surface heat flux or vessel pressure. The initial portion of the scram curve used for analyses bounds the nominal scram curve multiplied by the conservatism factor of 0.8 in order to assure the coverage of the transient effect with the intended conservatism.

b) The scram time characteristic shown in Figure 15.0-2 is derived from the Technical Specification scram time. The slowest allowable scram insertion speed was used for the scram curve applied to Chapter 15.0 analyses.

c) The $40.21 is the correct total reactivity worth for the end of cycle 1 scram curve and is incorrectly labeled in Figure 15.0-2. Figure 15.0-2 has been corrected.*

* Draft FSAR page change attached.



211.152-2

15.0-22

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SCRAM POSITION AND REACTIVITY CHARACTERISTICS	FIGURE 15.0-2

Q. 211.164
(15.1.1.2.2)

On page 15.1-2, it is stated that the thermal power monitor (TPM) is the primary protection system for mitigating the consequences of the transient resulting from loss of feedwater heating. A description of this monitor, which typically involves the flow-weighted APRM scram in conjunction with a 6-second time constant circuit, was not found in the WNP-2 FSAR. Provide this description in sufficient detail to permit evaluation of the TPM for WNP-2.

If the time constant, which affects scram initiation by the TPM, is less than the effective time constant for the WNP-2 fuel for this type of transient, the TPM should provide a conservative measure of the time variation is surface heat flux. However, if the time constant is appreciably larger than that for the fuel, the fixed APRM trip without a time constant would provide the scram protection. The resulting MCPR would then be less than that predicted for the TPM scram which has a lower setpoint.

There is no current provision in the Technical Specifications for surveillance of this time constant circuit. It is the staff's position that credit be taken only for the fixed APRM scram in Chapter 15 unless the TPM is approved by the staff and appropriate limiting conditions for operation and surveillance requirements are incorporated in the Technical Specifications for WNP-2.

a) Provide an analysis of the "loss of feedwater heating" transient assuming credit only for the fixed APRM scram. This is a more conservative approach because it will result in a more severe transient due to a higher fixed APRM scram setpoint.

b) Revise NSOA Figure 15A.6-21 to indicate the high flux scram signal occurs from the fixed APRM scram instead of the TPM.

c) Re-evaluate single failure criteria in Section 15.1.1.2.3 without taking credit for the TPM.

Response:

Sections 7.6.1.4.3 and 7 2.1.1.b.1.b of the WNP-2 FSAR describe the thermal power monitor function of the Neutron Monitoring System (NMS). Table 7.6-3 APRM System Trips and Figure 7.6-10, APRM Circuit Arrangement for RPS Input, provide additional information on the TPM setpoints and trip actions. These descriptions permit the evaluation of the TPM's application to WNP-2

In a response to an earlier question, (211.089) the Supply System outlined its intentions with respect to technical specifications surveillance requirements and limiting conditions for operation for the TPM. The TPM components are safety grade qualified (SC-2, quality class I, seismic categoryI) and the system is designed to be single failure proof. For these reasons, it is appropriate to take credit for the TPM scram during the "Loss of feedwater heating" (LFWH) event. Re-analysis of the LRWH event and revision of the appropriate NSOA figure without TPM is unjustified. The evaluation in Section 15.1.1.2.3 remains accurate.*

*Draft FSAR page change attached.

 Revise Q 211.164

15.1.1.2 Sequence of Events and Systems Operation

15.1.1.2.1 Sequence of Events

Tables 15.1-1 and 15.1-2 lists the sequence of events for this transient and its effect on various parameters is shown in Figure 15.1-1 and 15.1-2.

15.1.1.2.1.1 Identification of Operator Actions

In the automatic flux/flow control mode, the reactor settles out at a lower recirculation flow with no change in steam output. An average power range monitor (APRM) neutron flux or thermal power alarm will alert the operator that he must insert control rods to get back down to the rated flow control line, or that he must reduce flow if in the manual mode. The operator must determine from existing tables the maximum allowable T-G output with feedwater heaters out of service. If reactor scram occurs, as it does in manual flow control mode, the operator must monitor the reactor water level and pressure controls and the T-G auxiliaries during coastdown.

15.1.1.2.2 Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems.

The thermal power monitor (TPM) is the primary protection system trip in mitigating the consequences of this event. A description of the TPM is provided in 7.2.1.1.6.1.6 and 7.6.1.4.3. Required operation of engineered safeguard features (ESF) is not expected for either of these transients.

PTM

15.1.1.2.3 The Effect of Single Failures and Operator Errors

These two events generally lead to an increase in reactor power level. The TPM mentioned in 15.1.1.2.2 is the mitigating system and is designed to be single failure proof.

Q. 211.168
(15.14.2.1.1)

For the "inadvertent opening of a safety/relief valve" transient, include the time at which suppression pool temperature alarms and Technical Specification limit are attained in event Table 15.1-5.

Response:

WNP-2 is currently in the process of analyzing the Suppression Pool (SP) temperature response for various transients, including a stuck open relief valve. The specific transients to be analyzed along with the methodology, assumptions and initial conditions are outlined in NUREG-0783. Among the initial conditions assumed in these analyses is that the suppression pool temperature is at the technical specification limit. For WNP-2, this limit is 90° Fahrenheit, which is also the SP high temperature alarm point. Table 15.1-5 has been revised to indicate that at time zero the plant is operating at the maximum technical specification, suppression pool temperature and the high suppression pool temperature alarm is received in the control room.* Upon completion of the SP temperature response analysis, the FSAR will be revised to incorporate the final results.

*Draft FSAR page change attached.

(Q ~~[illegible]~~) 211.168

TABLE 15.1-5

SEQUENCE OF EVENTS FOR INADVERTANT SRV OPENING

Time	Event
0	Initiate opening of 1 safety relief valve which remains open throughout the event.
0	Suppression Pool at tech spec limit *
0	Operator receives suppression pool high temperature alarm
10 Min	Operator attempts to close valve fail. Operator scrams the plant and MSIV closure occurs (worst case).
10.5 Min	RCIC or HPCS initiate.
20 Min	Operator activates RHR in suppression pool cooling mode.
5 Hours	Shutdown and cooldown completed.

* the suppression pool is considered to be at the maximum tech spec temperature for continuous power generation without pool cooling in service

Q. 211.180

Question deleted.

Q. 211.189
(15.2.9.3)

For the "failure of BHR shutdown cooling" event, specific input parameters for the models used to evaluate blowdown rate and suppression pool temperature are shown in Table 15.2-13 along with the analytical results in Figures 15.2-16, -17, -18, and -19. In connection with this, profide the following information:

a) Identify the analytical models used to evaluate blowdown rate and suppression pool temperature.

b) Revise Table 15.2-13 to include all the input parameters for the models to be identified in step a) and provide justification that the input parameters are conservative.

In addition, it is indicated that only a qualitative evaluation of the "failure of RHR shutdown cooling" transient is provided because the core behavior has been analyzed in Section 15.2.6. Update the FSAR to indicate a quantitative analysis has been provided.

Response:

a) The analytical computer codes used to evaluate blowdown rate and suppression pool temperature response are described in NEDO-10320 and NEDO-10320 Supplement 1, "General Electric Pressure Suppression Containment Analytical Model," and in NEDE-20877, "Long Term Containment Response for BWR."

b) Table 15.2-13 has been proveded to show the key parameters which ralate to the transient analysis.* Providing a complete list of inputs would be impractical. The short and long term responses were obtained from the models referenced in a) above. Parameters in which variations might have significant effect upon the results were selected at the most conservative design basis values (e.g., minimum suppression pool mass) to maximize the containment pressure and temperature response. If some area of input is of special

interest, it can be provided upon specific request.

Section 15.2.9 of the WNP-2 FSAR has been revised to remove the reference to the "qualitative evaluation."*

*Draft FSAR page changes attached.

REVISION

c. At 100 psig RPV pressure, actuates ADS to complete blowdown; and the operator establishes a reactor cooling path as described in the notes for Figure 15.2-11.

Time required to initiate the necessary steps to maintain reactor pressure and level control is approximately 10 minutes.

15.2.9.2.2 System Operation

Plant instrumentation and control is assumed to be functioning normally except as noted. In this evaluation credit is taken for the plant and reactor protection systems and/or the ESF utilized.

15.2.9.2.3 The Effect of Single Failures and Operator Errors

The worst case single failure (Loss of Division Power) has already been analyzed in this event. Therefore, no single failure or operator error can make the consequences of this event any worse. See 15A for discussion of this subject.

15.2.9.3 Core and System Performance

An event that can directly cause reactor vessel water temperature increase is one in which the energy removal rate is less than the decay heat rate. The applicable event is loss of RHR shutdown cooling. This event can occur only during the low pressure portion of a normal reactor shutdown and cooldown, when the RHR system is operating in the shutdown cooling mode. The earliest time the shutdown system can be actuated is 2-3 hours after shutdown is initiated. During this time MCPR remains high and nucleate boiling heat transfer is not exceeded at any time. Therefore, the core thermal safety margin remains essentially unchanged. The 10-minute time period approximated for operator action is an estimate of how long it would take the operator to initiate the necessary actions; it is not a time by which he must initiate action. ~~Only a qualitative evaluation is provided below since the transient behavior of the core has been evaluated in 15.2.6.~~ 2

The TRANSIENT behavior of the core during this event has been evaluated in section 15.2.6.

211.189-2

15.2.9.4 Qualitative Results

For most single failures that could result in loss of shutdown cooling, no unique safety actions are required. In these cases, shutdown cooling is simply re-established using the redundant shutdown cooling loop. In cases where the RHR shutdown cooling suction line valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function (Figure 15.2-10). An evaluation has been performed assuming a failure that disables the RHR shutdown cooling suction line valves.

This evaluation demonstrates the capability to safely transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

The alternate cooldown path chosen to accomplish the shutdown cooling function utilizes the RHR and ADS or normal relief valve systems (see Reference 15.2-3 and Figure 15.2-11). The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment using only safety systems. The systems are capable of bringing the reactor to a cold shutdown in approximately 36 hours or less after the transient occurs.

The systems have suitable redundancy in components such that even for onsite electrical power operation (offsite power is not available) the systems' safety function can be accomplished assuming an additional single failure. The systems can be fully operated from the main control room.

The design evaluation is divided into two phases: (1) full power operation to approximately 100 psig vessel pressure, and (2) approximately 100 psig vessel pressure to cold shutdown (14.7 psia 200°F) conditions.

15.2.9.4.1 Full Power to Approximately 100 psig

Independent of the event that initiated plant shutdown (whether it be a normal plant shutdown or a forced plant shutdown), the reactor is normally brought to approximately 100 psig using either the main condenser or, in the case where the main condenser is unavailable, the RCIC/HPCS systems together with the nuclear boiler pressure relief system and the RHR heat exchanger in the suppression pool cooling mode.

Q. 211.198
(6.3)

Expand the discussion in Section 6.3 to describe the design provisions that are incorporated to facilitate maintenance (including draining and flushing) and continuous operation of the ECCS pumps, seals, valves, heat exchangers and piping rusn in the long-term LOCA mode of operation considering that the water being recirculated is potentially very radioactive.

Response:

In response to items II.B.2 and II.D.1 of NUREG-0737, WNP-2 is currently evaluating the ECCS, as well as RCIC and other systems required for long-term cooling, considering that the water being recirculated may be potentially very radioactive. One of the objectives of this evaluation of to determine that the release of large amounts of radioactive material will not limit personnel occupancy or degrade safety equipment by the radiation fields that may exist during and following the accident to the extent that required safety functions cannot be accomplished.

The evaluation assumes the source terms recommended in Regulatory Guide 1.3 and 1.7 and Standard Review Plan 15.6.5. It further assumes that these source terms are released instantaneously at the start of the accident. No one particular accident scenario is used; however, all systems which receive automatic initiating signals are assumed to be running. If these systems take suction from the containment atmosphere or the suppression pool, they are assumed to be contaminated. The systems assumed to be contaminated are as follows:

Reactor Core Isolation Cooling (RCIC)

Residual Heat Removal (RHR)

Low Pressure Core Spray (LPCS)

High Pressure Core Spray (HPCS)

Containment Atmospheric Control (CAC)

Main Steam (up to the second isolation valve) (MS)

Main Steam Isolation Valve Leakage Control System (MSIVLC)

Primary Containment

Secondary Containment (due to leakage from primary containment and systems in secondary containment)

Standby Gas Treatment (SGT)

The reactor building is separated into radiation zones. Within each radiation zone, all the significant contaminated fluid piping is located. In addition, all safety related equipment is identified and located in each radiation zone. A "worst case" target is chosen and located by inspection or by order-of-magnitude calculations for each potential "worst case" target. Next, the total integrated accident dose is calculated, taking into account direct shine from the contaminated system piping and from the primary containment building. The accident dose is calculated using a time period of six months because the integrated dose from the contaminated fluid systems does not increase significantly after six months. The accident dose is acced to the 40 year integrated dose from normal plant operation. It is this total integrated dose for the "worst case" total integrated dose, then specific calculations are performed. If necessary, the equipment is relocated, replaced, or shielded.

In addition to determining that the equipment needed to mitigate an accident is not unduly degraded by the resulting radiation fields, the evaluation also identifies vital areas needed for post-accident operations (e.g., Control Room, Sampling Station, Technical Support Center, etc.) and provides assurance that there is access to these vital areas.

An interim report will be submitted to the NRC by the end of 1981 identifying all the assumptions and methodologies used to do the evaluation. Results of the evaluation in terms of accessibility and equipment reliability will also be presented. As required by NUREG-0737, Item II.B.2, the final report, including an evaluation of all safety-related equipment applicable, will be completed and submitted at least four months prior to issuance of an operating license.

Q. 211.202

A timer is used in each ADS logic. The time delay setting before actuation of the ADS is long enough that the HPCS system has time to operate, yet not so long that the LPCI and core spray systems are unable to adequately cool the fuel if the HPCS system fails to start. Manual reset circuits are provided for the ADS initiation signal and primary containment high pressure signals. By resetting these signals manually, the delay timers are recycled. The operator can use the reset pushbuttons to delay or prevent automatic opening of the relief valves if such delay or prevention is necessary. The operator may also interrupt the depressurization at any time by the same action. The operator would make this decision based on an assessment of other plant conditions.

Discuss in detail any criteria to be given to the operator (e.g., in emergency procedures, or operator training) that would form the bases for the operator's decision. Discuss the consequences of interrupting ADS depressurization prior to reaching the injection pressure for low pressure systems.

Response:

At the present time, WNP-2 is still formulating the emergency procedures. This effor will involve incorporating BWR Emergency Procedure Guidelines (EPG) which, in part, provide the following criteria:

1) Monitor reactor pressure vessel water level and pressure and primary containment temperatures and pressures from multiple indications.

2) If a safety function initiates automatically, assume a true initiating event has occurred unless otherwise confirmed by at least two 2) independent indications.

3) Do not secure or place an emergency core cooling system in manual mode unless by at least two (2) independent indications, a) missoperation in automatic mode is confirmed, or b) adaquate core cooling is assured. If an emergency core cooling system is placed in the manual mode, it will not initiate automatically. Make frequent checks of the initiating or controlling parameters. When manual operation is no longer required, restore the system to automatic/standby mode if possible.

The Emergency Procedures as they now exist specifically mention resetting the ADS timers in two precedures: 4.8.1 HPCS Failure Step 4.8.1.4 Step A.9 and 4.8.0.2.4 Total Loss of All Feedwater Flow Step 3.C.

Step 4.8.1.4, A-9 HPCS Failure-
"If reactor water is being restored and ADS timers have initiated, reset the timers at panel P.601.
"Note: ADS timers are not to be reset without permission of the Shift Supervisor."

Step 4.8.0.2.4, 3.C Total Loss of All feedwater Flow-
"If ADS timers have been initiated, monitor reactor water level.
If level can be restored and maintained with HPCS and RCIC, reset the ADS timers on Panel P.601."
"Note: ADS timers are not to be reset without permission of the Shift Supervisor."

Resetting of the ADS timer as well as the interruption of ADS depressurization are both covered under the EPG as described above. The operator would not reset ADS unless he would confirm either adequate core cooling or ADS system messoperation by at least two independent indications.

In addition, the ADS function would only be required during the course of a plant transient in the event that the High Pressure (HP) makeup systems failed.

Q. 211.203
(6.3)

Restricting orifices are commonly installed downstream of a pump to limit the maximum flow rate that could occur and prevent pump damage if the pump discharge line were to fail (i.e., pump runout protection). It is not clear whether or not restricting orifice plates will be used for the LPCI system at WNP-2. Figures 5.4-13a and 5.4-13b show a restricting orifice in the injection piping of each LPCI loop. However, note 9 on Figure 5.4-13a states that these orifices are recommended but not required.

Describe precautionary measures taken to reduce the potential for LPCI pump damage due to runout conditions.

Response:

The metering orifice in the discharge line does not serve as a restricting orifice.

The piping for each mode of RHR operation has been investigated to ensure that the resistance is low enough to allow the rated flows given in Figure 5.4-14b yet high enough to prevent pump runout. Restricting orifices are necessary in the system test lines to prevent excessive runout during suppression pool cooling and test modes and in the main discharge line to prevent excessive runout for LPCI and all other RHR modes. Engineering changes are currently being processed which will add these restricting orifices. Figure 3.2-6 will be revised to indicate the location of restricting orifices in the main discharge line.

Q. 211.204
(6.3)

Figures 6.3-53a, -53b, -54a, and -54b show the results of a break in a core spray line from the "lead plant" analyses. The assumed single failure shown on the figures does not appear to be the most limiting. It would appear that the LPCI diesel-generator failure (division 2) would be more restrictive than the LPCS diesel-generator failure (division 1), i.e., only LPCI Loop A wuld be available to reflood the core. Explain why failure of the LPCI diesel-generator (division 2) does not result in a higher peak cladding temperature than that shown on Figure 6.3-54b.

Response:

Assuming a high pressure core spray (HPCS) line break, the worst single failure is the LPCS diesel-generator (DG failure (division 1). With this failure only 2 LPCI loops are available for reflooding the core. The LPCI is injected into the core bypass region and drains into the lower plenum through specified leakage paths (refer to NEDE 20566, Section 3.3 p. II-14 for further details). The flow allowed through these leakage paths is insufficient to completely drain the flow from 2 LPCI loops. Therefore for the LPCS DG failure case there is a buildup of LPCI flow in the core bypass region. This water that builds up in the bypass region does not directly contribute to reflooding the core. These factors combine to produce correspondingly longer reflooding times (and hence higher peak cladding temperatures) for the LPCS DG failure case when compared to the LPCI DG failure case.

For the LPCI DG (division 2) failure case which leaves 1 LPCI and 1 LPCS available for reflooding the core from division 1, the flow of the 1 LPCI is allowed to drain through the leakage paths. Also, although limited by counter current flow limiting (CCFL) considerations, STET the LPCS flow passes through the fuel bundles and into the lower plenum thereby providing core spray heat transfer and directly contributing to the reflooding of the core.

Q. 211.211
(15.3.3)

The response to question 211.092 is unacceptable. Explain why the DBA-LOCA event is indicated as conservatively bounding the pump seizure event when different acceptance criteria are used for each. The pump seizure event is evaluated based on exceeding 10CFR100 guidelines whereas the main criterion for evaluating the DBA-LOCA event is a peak cladding temperature of 2200°F. Coordinate this request with question 211.185.

Response:

See response to question 211.185.

Q. 211.212

Our position on the emergency core cooling systems (ECCS) is that these systems should be disigned to withstand the failure of any single active or passive component without adversely affecting their long-term cooling capabilities. In this regard, we are concerned that the suppression pool in boiling water reactors (BWR's) may be drained by leakage from isolation valves which may be rendered inaccessible by localized radioactive contamination following a postulated loss-of-coolant accident (LOCA). Accordingly, indicate the disegn features in the WNP-2 facility which will contain leakage from the first isolation valve in the ECCS lines taking water (suction lines) from the suppression pool during the long-term cooling phase following a postulated LOCA.

Response:

During normal operation, leakage is collected in sumps located in the ECCS and RCIC pump compartments in the reactor building and pumped to the radwaste building for processing. The ECCS and RCIC pump compartments are watertight structures, the walls of which rise to a level above the suppression pool water level.

For the worst conceivable leak from the suppression pool in which the water level in the largest of these pump compartments equalizes with the suppression pool water level, at least 12 feet of NPSH over NPSH requirements on the ECCS pump performance reports is maintained for each pump. The suppression pool water temperature was taken at 212°F.

Additionally, the concerns in this question have been previously responded to in NRC Question 212.003, in which WNP-2 committed to installing Class IE level instruments in each ECCS pump room. These instruments will be mounted just above floor level to allow sufficient time for the operator to identify and isolate the faulted ECCS line. Any ECCS leak can be isolated, including any packing failure on any ECCS pump suction valve.

Q. 022.069
(RSP)

Based on our review of the information presented in Section 6.2.1.1.5 of the FSAR and your responset to item 022.018 which references your response to Item 031.070, we find that your discussion of steam bypass from the drywell to the wetwell for postulated small steam line breaks, is unaccptable. Specifically, the maximum allowable bypass leakage which you calculate (i.e., $A/\sqrt{K}$ = 0.028 sq. ft.), is not acceptable. Accordingly, we require that you design the WNP-2 containment to have a bypass leakage capability which satisfies the provisions of Appendix I to Section 6.2.1.1.C of the Standard Review Plan (SRP); (i.e., $A/\sqrt{K}$ = 0.05 sq. ft.). Provide the appropriate discussions, justifications and analyses to demonstrate how you comply with the provisions of Appendix I cited above.

Response:

Please refer to revised 6.2.1.1.5.4 and Figure 6.2-17b. Also refer to the revised response to Question 031.070.*

*Draft FSAR page changes attached.

6.2.1.1.5.2 Reactor Blowdown Conditions and Operator Response

In the highly unlikely event of a primary system leak in the drywell accompanied by a simultaneous open bypass path between the drywell and suppression chamber, several postulated conditions may occur. For a given primary system break area, the maximum allowable leakage capacity can be determined when the containment pressure reaches the design pressure at the end of reactor blowdown. The most limiting conditions would occur for those primary system break sizes which do not cause rapid reactor depressurization~~, but rather have long leakage duration~~. These ~~This~~ break sizes which ~~less~~ are less than ~~approximately~~ 0.4 square feet, require~~s~~ operator action to terminate the reactor blowdown ~~next~~ if there is a bypass path.

replace with insert A attached

~~Immediately after the postulated conditions given for a small primary system break, there would be a fairly rapid rise in containment pressure as the noncondensable gases in the drywell are carried over to the suppression chamber. During this portion of the transient, it is assumed that the plant operators are unaware that a leakage path exists. Under these circumstances, the maximum pressure that can occur in the suppression chamber is approximately 27 psig. This is the pressure that would result if all of the noncondensable gases initially in the containment are carried over to the suppression chamber free space.~~ For the maximum allowable leakage calculations, it ~~was~~ is assumed that the plant operators realize a leakage path exists ~~only when the suppression chamber pressure reaches 30 psig. For conservatism, an additional ten minute delay is assumed before any corrective action is taken to terminate the transient. The corrective action is also assumed to take five minutes to be effective. At that time, the containment pressure would be equal to the design pressure if the allowable leakage had occurred. The specific corrective action taken after ten minutes is not accounted for in the analysis. The operators have several options available to them. If the source of the leakage is undefined, they would probably depressurize the primary system via either the main condenser or relief valves, or they could activate the containment spray system.~~

Insert B attached

6.2.1.1.5.3 Analytical Assumptions

When calculating the allowable leakage capacities for a spectrum of break sizes, the following assumptions are made:

Insert A to Page 6.2-29:

Following the postulated condition given above, there is an increase in drywell pressure which leads to drywell venting to the wetwell via the downcomers. Both noncondensibles and vapor are vented. If no bypass leakage exists, the maximum suppression chamber pressure would be 28 psig, the pressure resulting from displacing all containment noncondensibles into the suppression chamber.

Insert B to Page 6.2-29:

after an alarm announces that the suppression chamber pressure has reached 30 psig and prepare to take action. Containment sprays must be initiated with a delay no longer than 41 minutes or drywell pressure will exceed 45 psig. Containment pressure decreases immediately upon the starting of drywell spraying. Once the containment pressure is reduced by means of drywell spraying, the operators would proceed to depressurize the reactor vessel by means of either the main condenser or the relief valves.

a. Flow through the postulated leakage path is pure steam. For a given leakage path, if the leakage flow consists of a mixture of liquid and vapor, the total leakage mass flowrate is higher but the steam flowrate is less than for the case of pure steam leakage. Since only the steam entering the suppression chamber free space results in the additional containment pressurization, this is a conservative assumption.

b. There is no condensation of the leakage flow on either the suppression pool surface or the containment and vent system structures. Since condensation acts to reduce the suppression chamber pressure, this is a conservative assumption. For an actual containment there will be condensation, especially for the larger primary system break where vigorous agitation at the pool surface will occur during blowdown.

6.2.1.1.5.4 Analytical Results

The containment has been analyzed to determine the allowable leakage between the drywell and suppression chamber. Figure 6.2-17a shows the allowable leakage capacity ($A/\sqrt{K}$) as a function of primary system break area. A is the area of the leakage flow path and K is the total geometric loss coefficient associated with the leakage flow path.

Figure 6.2-17a is a composite of two curves. If the break area is greater than approximately 0.4 square feet, natural reactor depressurization will rapidly terminate the transient. For break areas less than 0.4 square feet, however, continued reactor blowdown limits the allowable leakage to small values. ~~The~~ maximum allowable leakage capacity ~~is~~ at $A/\sqrt{K}$ = ~~[illegible]~~ square feet. Since a typical geometric loss factor would be three or greater, the maximum allowable flow path would be about ~~0.052~~ [illegible] square feet. This corresponds to a ~~[illegible]~~ 4 inch line size.

150

use 0.1 sq ft

Burns and Roe, Inc. confirmed the results of the above analysis by GE in Reference 6.2-7. Further investigation into the transient nature of the problem was then undertaken at the request of the NRC, which assigned the in Appendix I of SRP 6.2.1.1.C,

A transient analysis using the CONTEMPT-LT (Ref. 6.2-8) computer code was performed. The code was modified to include the mass and energy transfer to the suppression pool from relief valve discharge. The limiting case was a very small reactor system break which would not automatically result in reactor depressurization. For this limiting case, it was assumed that the response of the plant operators was to shut the reactor down in an orderly manner at 100°F/hr cooldown rate. No other operator actions ~~was~~ were accounted for. Heat sinks considered were such items as major support steel inside containment, the reactor pedestal, the diaphragm floor and support columns and the steel and concrete of the primary containment. Based on this analysis, the allowable bypass leakage ($A/\sqrt{K}$) used was ~~0.028~~ 0.050 ft^2. The drywell pressure transient is shown in Figure 6.2-17b along with the corresponding curves of wetwell pressure, wetwell temperature and suppression pool temperature.

The mandated allowable bypass leakage of ~~0.028~~ 0.050 ft^2 is well above the Tech Spec ~~maximum possible~~ containment bypass leakage. Periodic testing will be performed to confirm that the containment bypass leakage does not exceed $A/\sqrt{K}$ = 0.0045 ft^2. Figure 6.2-17c presents the resulting containment transient for $A/\sqrt{K}$ = 0.0045 ft^2. The peak containment pressure shown in Figure 6.2-17c is well below the containment design pressure.

6.2.1.1.6 Suppression Pool Dynamic Loads

A generic discussion of the suppression pool dynamic loads and asymmetric loading conditions is given in Mark II Dynamic Forcing Function Information Report, Reference 6.2-4. A unique plant assessment of these dynamic loads is made in WNP-2 Design Assessment Report, Reference 6.2-5.

6.2.1.1.7 Asymmetric Loading Conditions

See comment in 6.2.1.1.6.

to initiate the drywell sprays when the ~~drywell~~ suppression chamber pressure exceeds ~~54~~ 39 psig, and then to proceed

A point by point discussion summarizing the WPPSS design capabilities to mitigate Bypass Leakage problems based on the above correspondence and with respect to the proposed Branch Technical Position is given below:

1. NRC Proposed Requirement: Allowable bypass capability on the order of 0.05 ft^2 ($A/\sqrt{K}$)

   Response: As ~~documented in~~ required by Reference 7 and the FSAR, the maximum allowable bypass leakage capacity is $A/\sqrt{K}$ = ~~0.028~~ 0.050 ft^2 ~~using conservative calculational techniques and assumptions.~~* ~~WPPSS, therefore, believes the existing calculations meet the intent of $A/\sqrt{K}$ = 0.05 ft^2.~~

2. NRC Proposed Requirement: An automatic system should be provided to initiate automatic wet-well sprays. The system should meet the standards of an Engineering Safety Feature including redundancy and diversity and be actuated automatically ten minutes following a LOCA. If the RHR system is used for this purpose, it must be analyzed to assure no degradation of its ECCS function.

   Response: WPPSS asserts that manual initiation is sufficient since the drywell floor will be routinely tested and evaluated against a Tech Spec limit of $A/\sqrt{K}$ = 0.0045 ft^2, a level at which no operator action is required for the spectrum of small break sizes. (Reference 5 - see #3 below for testing details).

   The construction, design, quality control, and surveillance requirements on the drywell floor give it the same level of safety as the containment itself. Reference 4 and Part VI of Reference 2 showed that through-wall cracks will not develop through the concrete slab under postulated design conditions including

* ~~The FSAR currently lists the capability as 0.026 ft^2. This is from a GE analysis and the FSAR is being amended to reflect the latest calculations.~~

the SSE and that leakage in excess of that accounted for due to permeability would not be possible. Reference 6 indicated the NRC Structural Engineering Branch's acceptance of these responses. Accordingly, WPPSS sees no reason to assume that an $A/\sqrt{K}$ of 0.0045 ft^2 is exceeded any more than there would be reason to assume the design containment leak rate of .5% per day is exceeded. Calculations using the methodology documented in Reference 5 ~~using~~ with the CONTEMPT - LT computer code were used in computing the maximum allowable leakage rate of $A/\sqrt{K}$ = ~~0.028~~ 0.050 ft^2, ~~six~~ eleven times the Tech Spec limit. In the calculation ~~over 167~~ 4.1 85 minutes was available after the wetwell press exceeded 30 psig for operator action before drywell design pressure was exceeded. Accordingly, a requirement that an automatic system be provided is unnecessary.

7/21/81

3. NRC Proposed Requirement: A single ~~pe~~ A operational high pressure leakage test should be performed and periodic low pressure tests at each refueling outage with an acceptance criterion of 10% of the bypass capability at the test pressure.

Response: The intent of this proposed requirement has been committed to by WPPSS. A single preoperational leakage test will be conducted with the downcomers capped at 15 psid and 25 psid (the design drywell to wetwell differential pressure). At each refueling outage a low pressure operational test will be performed as a Tech Spec Surveillance Requirement to verify 0.0045 ft^2. Details of the nature of this test are discussed in question 5.22 to the PSAR but will be summarized here since the specific numbers have been since updated.

Routine Leak Testing and Inspection: During entry to the drywell at each refueling outage, accessible drywell to wetwell barrier surfaces will be visually inspected to ascertain any possible leak paths. Vacuum relief valves will be visually inspected to insure they are clear of foreign material. At each refueling

References

1. Letter, WR Butler, NRC, to JJ Stein, WPPSS, "Meeting Summary", October 17-18, 1973, dated November 26, 1973.

2. Letter, WPPSS to NRC, GO2-74-17, dated August 9, 1974.

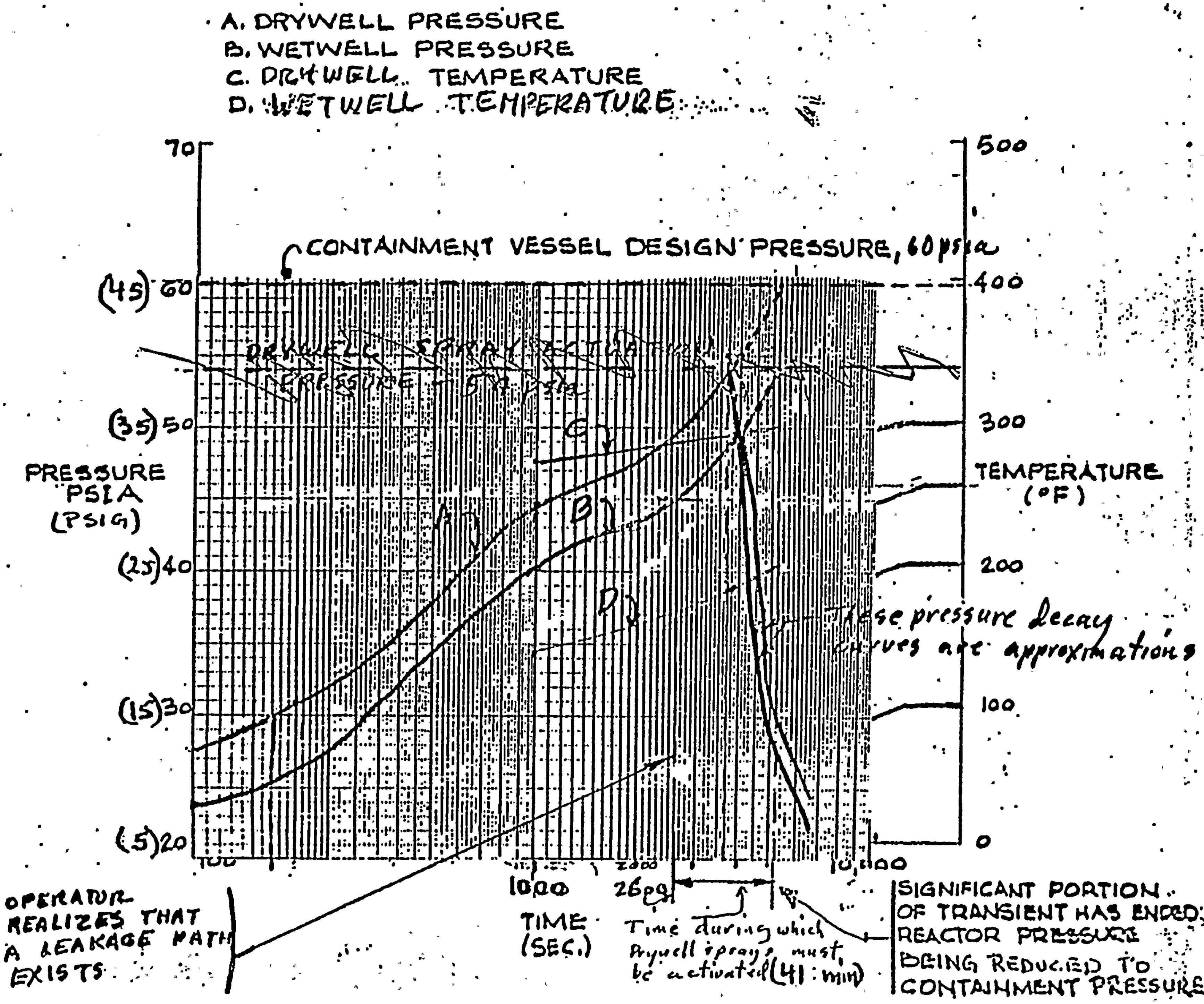
3. Letter, NRC to WPPSS, dated January 14, 1975.

4. Letter, WPPSS to NRC, GO2-75-52, dated February 25, 1975.

5. Letter, WPPSS to NRC, GO2-76-156, dated April 23, 1976.

6. Letter, NRC to WPPSS, dated May 15, 1975.

7. NRC Question 22.069, referencing Appendix I of ~~SRP~~ NRC Standard Review Plan (Section 6.2.1.1).



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

CONTAINMENT TRANSIENT FOR MAXIMUM ALLOWABLE BYPASS CAPACITY, $A/\sqrt{k} = 0.050\ FT^2$

FIGURE 6.2-71

Amendment No. 3
March 1979

Q. 022.074

In note 31 of Table 6.2-16, you indicate that primary containment and reactor vessel isolation signals are not desirable signals for initiating closure of the feedwater block valve. We find this approach accetable provided that the valve can be manually closed from the control room (i.e., remotely) if the control room operator determines that continued makeup from the feedwater system is either unavailable or unnecessary. Discuss the information which will be available to the control room operator to alert him of the need to isolate the feewater system. Indicate the time interval which would then take the operator to complete this action.

Response:

The response to this question is addressed under revised Section 6.2.4.3.2.1.1.1.*

*Draft revised FSAR page changes attached.

FSAR Revision 022.074

Table 6.2-16 contains those influent pipes that comprise the reactor coolant pressure boundary and penetrate the containment.

6.2.4.3.2.1.1.1 Feedwater Lines

The feedwater lines are part of the reactor coolant pressure boundary as they penetrate the drywell to connect with the reactor pressure vessel. The isolation valve inside the drywell is a y-pattern check valve, located as close as practicable to the containment wall. Outside the containment is another y-pattern check valve located as close as practicable to the containment wall and farther away from the containment is a motor operated gate valve. Should a break occur in the feedwater line, the check valves prevent significant loss of reactor coolant inventory and offer immediate isolation. However, in case a loss-of-coolant accident occurs without a seismic event, the design allows the condensate and condensate booster pumps to supply feedwater to the vessel through a bypass line around the reactor feed pumps - which are tripped on a loss of steam supply - as soon as the vessel is partially depressurized. For this reason, the outermost, gate valve does not automatically isolate upon signal from the protection system. The gate valve meets the same environmental and seismic qualifications as the outboard isolation valve. The valve is capable of being remotely closed from the control room to provide long-term leakage protection upon operator judgement that feedwater makeup is unavailable or unnecessary. INSERT No credit is taken for feedwater flow in accessing core and containment response to a loss-of-coolant accident.

6.2.4.3.2.1.1.2 HPCS Line

The HPCS line penetrates the drywell to inject directly into the reactor pressure vessel. Isolation is provided by an air testable check valve, located inside the drywell with position indicated in the main control room, and remote-manually actuated gate valve located as close as practicable to the exterior wall of the containment. Long-term leakage control is maintained by this gate valve. If a loss-of-coolant accident occurred, this gate valve would receive an automatic signal to open.

6.2.4.3.2.1.1.3 LPCI and LPCS Lines

Satisfaction of isolation criteria for the LPCI and the LPCS system is accomplished by use of remote-manually operated gate valves and check valves. Both types of valves are normally closed with the gate valves receiving an automatic

Question 022.074

Insert to 6.2-58

The operator can determine if make-up from the feedwater system is unavailable by use of the feedwater flow indicator in the control room, which will show high flow for a feedwater pipe break or no flow for feedwater pump trip.

The operator can also determine if make-up from the feedwater system is unnecessary by verifying that the ECCS is functioning properly and the reactor water level is being adequately maintained. ECCS operation signals and reactor vessel water level indication are provided in the control room for operator information.

Since due to the check valves it is not necessary to immediately isolate the feedwater system for leakage mitigation, there is no need to alert the operator to initiate the feedwater isolation signal other than as described above. However, for long term isolation purposes, the operator may close the motor-operated gate valve at any convenient time.

Q. 022.076

With regard to Note 45 of Table 6.2-16 of the FSAR, you should note that our acceptance criteria for containment isolation signals is set forth in Section 6.2.4, "Containment Isolation Systems," of the SRP. Indicate whether your design for the WNP-2 facility conforms to our acceptance criteria on this matter.

Response:

The Recirculation pump seal water supply line does not close automatically because of the desirability of maintaining the pump seal water as long as the supply is available or the RRC pump is operating.

The line will be isolated manually from the control room on indication of loss of either the CRD or RRC pumps or a loss of flow as indicated by the seal water inlet and outlet pressure instrumentation. All instrumentation indication is available to the operator in the control room.

This design conforms to the requirements of the SRP 6.2.4 acceptance criteria in paragraph II.11.

Q. 022.080
(RSP)

Your response to question 022.050 in which you state that the hydrostatic or pneumatic test will be repeated every ten years, is not acceptable. It is our position that those systems identified as closed systems and which become an extension of the primary containment, should be leak tested during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years. Accordingly, we require you to provide a commitment consistent with our position on this matter.

Response:

Note 18 of Table 6.2-16 commits the Supply System to meeting the requirements for a closed system as set by SRP-6.2.4, Section II, paragraph 3e which states in part, that "The closed system outside containment should be leak tested unless it can be shown that the system integrity is being maintained during normal plant operation." Of the five systems listed as closed systems outside containment in question 022.050, four (LPCS, HPCS, RHR, RCIC) are equipped with water leg pumps which maintain the system full of water and pressurized to about 85 psig during normal operation. The sections of these pressurized systems outside containment, therefore, are under a continuous leak check at a pressure over twice that required by 10CFR50, Appendix J. The pressurized system combined with periodic visual inspections are considered more than adequate to insure system integrity during normal operation.

The Containment Atmosphere Control System (CAC) is not pressurized during normal operation and will be pneumatically leak tested, as required, during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years. Paragraph 6.2.6.5 of the WNP-2 FSAR will be revised to show the additional test requirements.

6.2.6.5 Special Testing Requirements

The secondary containment shall be subjected to tests prior to initial fuel loading and at each refueling outage to assure the maximum allowable leakage rate of 100% of secondary containment free volume per day at -0.25 inches water gauge pressure with respect to outside atmospheric pressure. The test procedure for determining that the leakage rate does not exceed the maximum allowable is summarized in 6.2.3.4. See Chapter 16, Technical Specifications. Test procedures for the MSIV-LCS are in 6.7.

THE HYDROGEN RECOMBINER SYSTEM, AS A CLOSED SYSTEM OUTSIDE CONTAINMENT, SHALL BE LEAK TESTED DURING EACH REFUELING OUTAGE, OR AT OTHER CONVENIENT INTERVALS, BUT IN NO CASE AT INTERVALS GREATER THAN TWO YEARS. THE LEAK ~~B~~ TESTING SHALL BE IN ACCORDANCE WITH REG. GUIDE 1.141, POSITION C.1.

Q. 022.090

In the event of a small steam line break, the potential exists for steam to bypass the suppression pool via the hydrogen control system. Discuss the capability of the hydrogen recombiner system to condense this superheated steam. Discuss whether the after-cooler could be initiated independently from the recombiner system to condense leaking steam. Discuss the design provisions in the WNP-2 facility to eliminate the potential for steam bypass.

Response:

The hydrogen recombiner system containment isolation valves are normally closed and are opened only if the hydrogen concentration inside primary containment requires operation of the recombiner. Therefore, in the event of a small steam line break without hydrogen generation, the isolation valves remain closed and there is no potential for steam bypass through the hydrogen recombiner system.

The hydrogen recombiner system is designed to take suction from the drywell and discharge the processed stream into the suppression chamber. Existing discharge line valves to the drywell and suction line valves from the suppression chamber are key-locked closed and their electrical interlocks with the recombiner are disconnected. The key locks are located on a control room panel for remote manual operation, when and if another mode of operation (based on hydrogen concentration) is required. Refer to FSAR Section 6.2.5.2.3 and Figure 3.2-17. As dicussed in the response to NRC question 022.089, the hydrogen recombiner system is capable of handling a feed gas containing steam. Therefore, in the event of a small steam line break with hydrogen generation and the recombiner system in operation, there is no potential for steam bypass through the hydrogen recombiner system because any steam in the feed gas is condensed in the recombiner's scrubber.

Based on the above discussion and the response to NRC question 022.089, independent operation of the after-cooler to condense leaking steam is not required.