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SUBJECT: Forwards responses to NRC 810316 request for addl info re OL application review. Responses will be incorporated into FSAR within 4 months. Addl responses will be submitted by 810830.

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Dockett No. 50-397

G02-81-251
August 21, 1981

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

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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 remainder of the responses will be transmitted to the Nuclear Regulatory Commission by August 30, 1981.

Very truly yours,

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
WNP-2 Files

B001
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Q. 022.064

Provide a list, including appropriate drawings, identifying all piping, equipment, instrumentation and structures in the WNP-2 containment which may be subjected to pool dynamic loads. In addition, provide drawings showing the location of access galleys in the wetwell, the vent vacuum breaker configuration, the wetwell grating, the vent bracing configuration, the vent configuration in the pedestal region of the wetwell and any large horizontal structures in the zone affected by the pool swell phenomenon.

Response:

The "Plant Design Assessment Report for SRV and LOCA Loads," Revision 2, identifies piping, equipment, instrumentation and structures subjected to pool dynamic loads. Individually, each item is assessed in Chapter 4 of the report. A summary list of items is provided in Table 2.3-1. Drawings are provided in Chapter 2 as Figures 2.1-1 through 2.1-9.

Q. 022.065

Discuss the applicability of the generic supporting programs, tests and analyses (e.g., those relating to fluid-structure interactions, downcomer stiffeners and downcomer diameters) to the design of the WNP-2 facility.

Response:

Please refer to Section 1.1, Role of DFFR/Mark II Program, of the "Plant Design Assessment Report for SRV and LOCA loads," Revision 2.

Q. 022.066

Provide the time history of plant specific loads and your assessment of responses of plant structures, piping, equipment and components to pool dynamic loads. Identify any significant plant modifications which you made due to considerations of the pool dynamic loads.

Response:

The time history of plant specific loads and the assessment of plant structures, piping, equipment and components are given in the "Plant Design Assessment Report for SRV and LOCA Loads," Revision 2. Significant plant modifications are given in Section 2.3.1 of the report.

Q. 022.067

Provide the analyses which you performed to determine the post-swell wave load and the seismic slosh load. Discuss your analytical model and the assumptions you made in performing these analyses.

Response:

Please refer to the response to Question 022.020.

Q. 022.068

Provide the type, number and location of the temperature instumentation which will be installed in the suppression pool for the suppression pool temperature monitoring systems. Discuss the sampling and/or averaging technique that you will use to arrive at a definitive pool temperature. Provide justification for your approach.

Response:

FSAR section 7.6.1.7 has been revised to clarify and more accurately define the WNP-2 Suppression Pool Temperature Monitoring, Averaging, and Annunciation Techniques.*

*Draft FSAR page change attached.

7.6.1.6 Spent Fuel Pool Cooling and Cleanup System (FPC) - Instrumentation and Control

a. FPC System Function

The function of the FPC system is to remove decay heat from the spent fuel storage pool to insure adequate cooling of irradiated stored fuel assemblies. The FPC system also purifies the storage pool water, maintains water clairty for fuel handling operations, and fills and drains the fuel transfer canal. Refer to 9.1.3.

b. FPC System Operation

Schematic arrangement of the FPC system mechanical equipment is shown in Figure 3.2-12 (FPC P&ID). FPC system component control logic is shown in Figure 7.6-11 (FPC Logic Diag.). Instrument Specifications are listed in Tables 7.6-11 and 7.6-12. Plant layout drawings and electrical schematics are shown in 1.7. Operator information displays are shown in Figure 3.2-12 (FPC P&ID) and Figure 7.6-11 (FPC Logic Diag.)

The FPC System consists of two redundant cooling loops. The system is manually initiated and one loop runs continuously when the pool contains spent fuel.

Instrumentation is provided to monitor the pool temperature, pump suction and discharge pressures, and water conductivity to allow the control room operator to assess system operation.

7.6.1.7 Suppression Pool Temperature Monitoring System - Instrumentation and Controls

a. System Function

The suppression pool temperature monitoring (SPTM) system is designed to monitor suppression pool water temperature and alert the plant operator to the potentially hazardous condition of elevated pool water temperature.

The instrumentation for the SPTM system is shown in Figure 3.2-8. Specifications are identified in Table 7.6-13.

b. System Operation

The suppression pool temperature monitoring system consists of 24 dual element thermocouples. Sixteen thermocouples are arranged near the surface of the pool whereas the remaining 8 are located midlevel in the pool. This arrangement was chosen to track pool stratification. The sensors are separated into two redundant divisions and maintained throughout the system. INSERT #

Q077.068
81-178

The time constant for the thermocouples is no greater than 15 seconds. The time from signal output of sensor to initiation of alarm is no greater than 0.5 seconds. The difference between measurement reading and actual temperature is within ±2°F.

Each division of the Suppression Pool Temperature Monitoring System is provided with ~~temperature readout devices in the main control room consisting of four indicators and a four channel recorder. Each division is provided with an audible and visual annunciator in the control rool~~. ← Insert No 2

7.6.1.8 Design Basis

The safety-related systems described in 7.6 are designed to provide timely protective action inputs to other safety systems to protect against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the reactor coolant pressure boundary. Chapter 15, Appendix 15A, "Accident Analysis," identifies and evaluates events that jeopardize the fuel barrier and reactor coolant pressure boundary. The methods of assessing barrier damage and radioactive material releases, along with the methods by which abnormal events are identified, are also presented in Chapter 15.

The station conditions which require protective actions are described in Chapter 15 and Appendix 15A.

a. Variables Monitored to Provide Protective Actions

The following variables are monitored in order to provide protective action inputs:

1. High Pressure/Low Pressure System Interlocks

   a) Reactor pressure

   b) Differential pressure across the LPCS and LPCI injection valves

2. Leak Detection System

   a) RCIC area temperatures - differential and ambient

   b) RCIC steam line flow rate

   c) RCIC turbine exhaust diaphragm pressure.

Insert #1

... and 24 millivolt to milliamp converters with 4 to 20 milliamp output.

Insert #2

... a multipen recorder for individual thermocouple temperature recording, a microprocessor for averaging the thermocouple outputs of both divisions and providing an average bulk suppression pool temperature, a recorder for recording the average bulk temperature, and ~~visual and audible~~ audio-visual annunciators which alarm on abnormally high suppression pool temperatures output (at 90°F, 105°, 110° and 120°F) of each division and of the bulk temperature averaging system.

TABLE 7.6-13

SUPPRESSION POOL TEMPERATURE MONITORING INST. SPECIFICATIONS.

FUNCTION	INSTRUMENT	INSTRUMENT RANGE (1)	TRIP SETTING (2)	MARGIN (3)	REQUIRED ACCURACY (4)	RESPONSE TIME (4)
Monitor Supp. Pool Temp.	Thermocouple (SPTM-TE-1A, 1B Thru 8A, 8B; 9 thru 16	-300°F to +750°F	—	—	—	—
Change Thermocouple output to 4-20 mamp.	Convertor MV/I-1A, 1B Thru 8A, 8B; 9 thru 16	4 to 20 milliamps.	—	—	—	
Provide S.P. Average Temperature	Summer TΣ-1 TΣ-2	4 to 20 milliamps	90°F 105°F 110°F 120°F	—	—	
Individual Temp Recorder	Multi Pen Recorder TR-1 TR-2.	30°F to 230°F	90°F on any input.	—	—	—
Average Temp Recorder	TR-3 TR-4	30°F to 230°F	—	—	—	—

Add New Page

Note: this to be a new Table

AMENDMENT NO. Q022.068

Page 1 of 1

8/1-128

Q. 022.070

You state in your response to Item 022.6 that closed systems are not relied upon as barriers to eliminate bypass leakage. However, in your response to Item 022.35, you indicate that the reactor feedwater lines are the only lines for which a water seal is assumed to prevent secondary containment bypass leakage. Accordingly, explain your rationale for eliminating some of the penetrations listed in Table 6.2-16 of the FSAR as potential bypass leakage paths.

Response:

Lines with potential bypass leakage are only those that run through the secondary containment and terminate in the rad-waste or turbine buildings (page 6.2-49). All lines which are potential bypass leakage paths are noted in 6.2.3.2 (page 6.2-50).

Q. 022.071

Describe the test which you will perform to verify your assumptions about the amount of inleakage and the drawdown time for reestablishing - 0.25 inches of water gauge in the secondary containment following a postulated loss-of-coolant accident (LOCA).

Response:

The preoperational test for secondary containment leakage will be performed to assure that each train is capable of drawing down the secondary containment to - 0.25 inches of water gauge in 120 seconds and is capable of maintaining this negative pressure at a flow rate not exceeding 2240 CFM.

The leakage test will be performed by using the standby gas treatment filter unit. After system actuation, the air flow through the standby gas treatment filter unit, the secondary containment pressure will be monitored at pre-determined time intervals to demonstrate that the system meets the design intent and complies with the SAR and the Technical Specification.

Q. 022.072

In Table 6.2-16 of the FSAR, you indicate some lines which are connected to the reactor coolant pressure boundary or which connect directly to the containment atmosphere, rely on manual valves for containment isolation following a postulated LOCA. We find this approach unacceptable. Accordingly, we require you to provide containment isolation provisions which satisfy the requirements set forth in General Design Criteria (GDC) 55 and 56 of Appendix A to 10 CFR Part 50; these specific GDC require automatic isolation.

Response:

Both GDC 55 and 56 of Appendix A to 10 CFR 50 list closed isolation valves as an acceptable method for providing containment isolation.

SRP 6.2.4 para. II.3.f further states that: "Sealed closed barriers may be used in place of automatic isolation valves," and further defines sealed closed barriers as "...including blind flanges and sealed closed isolation valves which may be closed manual valves..."

WNP-2 penetrations utilizing manual isolation valves meet the requirements stated in GDC 55 and 56 and the acceptable alternate criteria of SRP 6.2.4.

Q. 022.073

In Table 6.2-16 of the FSAR, you indicate that the reactor recirculation hydraulic lines (X-76 and X-77) conform to the requirement of Criterion 57 of the GDC. It is our position that the isolation provisions for these specific lines should meet the requirements of Criterion 56. Further, in Table 6.2-16 of the FSAR, you indicate that traversing incore probe (TIP) system conforms to the requirements of Criterion 54 of the GDC. (Refer to Note 29 of Table 6.2-16.) However, in Section 6.2.4.3.2.3 of Criterion 57 of the GDC. It is our position that the TIP system should meet the requirements of GDC 56. Accordingly, revise Table 6.2-16 and other appropriate portions of the FSAR to reflect our position. Indicate if the other acceptable alternatives for meeting the requirements of the GDC as noted in Section 6.2.4 of the SRP could be applied to any of these lines.

Response:

The Supply System does not agree that the reactor recirculation hydraulic lines (X-76 and X-77) should conform to the requirements of Criterion 56 of the GDC which covers lines that penetrate the primary containment boundary and "connect directly to the containment atmosphere." These lines meet the requirements of Criterion 57 and SRP 6.2.4 as stated in note 28 of Table 6.2-16 and should remain so classified.

Table 6.2-16 and the FSAR has been revised to show the TIP system meeting the acceptable alternative requirements of Criterion 56 as noted in SRP paragraph 6.2.4.II.d.*

*Draft FSAR page changes(s) attached.

6.2.4.3.2.2.3.5 Reactor Building to Wetwell (RB-WW) Vacuum Relief Lines

The RB-WW vacuum relief penetrations, three in total, are each equipped with a positive closing swing check valve in series with an air-operated butterfly valve. The air operator on the swing check valve is used only for testing. The air operated butterfly valve is controlled by a differential pressure indicating switch which senses the pressure difference between the suppression chamber and the reactor building. When the negative pressure in the suppression chamber exceeds the instrument setpoint, the butterfly valve opens. The arrangement of valves and instruments is shown in Figure 3.2-15. See Table 6.2-15 for differential pressure indicating switch characteristics.

6.2.4.3.2.2.3.6 ← TIP Subsystem Guide tubes

6.2.4.3.2.2.4 Conclusion on Criterion 56

In order to assure protection against the consequences of accidents involving release of significant amounts of radioactive materials, pipes that penetrate the containment have been demonstrated to provide isolation capabilities on a case-by-case basis in accordance with Criterion 56.

In addition to meeting isolation requirements, the pressure retaining components of these systems are designed to the same quality standards as the containment.

6.2.4.3.2.3 Evaluation Against Criterion 57

Lines penetrating the primary containment for which neither Criterion 55 nor Criterion 56 govern comprise the closed system isolation valve group.

Influent and effluent lines of this group are isolated by automatic or remote manual isolation valves located as closely as possible to the containment boundary.

TIP subsystem guide tubes are provided with an isolation valve which closes automatically upon receipt of a proper signal and after the TIP cable and fission chamber have been retracted. In series with this isolation valve is included an additional or backup isolation shear valve. Both valves are located outside the drywell. The TIP system and isolation provisions are discussed in Note 29 of Table 6.2-16.

The RRC system hydraulic control lines to the flow control valve contain an isolation valve located outside the drywell which closes automatically upon receipt of its isolation signal. The hydraulic lines and their isolation valves are discussed in Note 28 of Table 6.2-16.

TABLE 6.2-16 (Continued)

LINE DESCRIPTION	Penetration No.	FSAR Figure No.'s	GDC	Code Gp.(12)	Valve No.	Valve Type	Location	Power to Open (5)	Power to Close (5)	Isolation Signal (9)	Back Up	Normal Position (10)	Shutdown Position	Post LOCA	Failure Position (6)	Valve Size (14)	Closure Time(7)(11)	Distance to Penetration	Leads to ESF System	Process Fluid	Leakage Barrier (13)	Termination Zone (13)	Potential (13) Bypass Leakage (SCFH)	Notes
Air line for testing RHR-V-50A	42d	6.2-31r 3.2-6	56	B	PI-VX-42d	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-216	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-50B	69c	6.2-31r 3.2-6	56	B	PI-VX-69c	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-221	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-41A	61f	6.2-31r 3.2-6	56	B	PI-VX-61f	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-219	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-41B	54Bf	6.2-31r 3.2-6	56	B	PI-VX-54Bf	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-218	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RHR-V-41C	62f	6.2-31r 3.2-6	56	B	PI-VX-62f	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-220	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing LPCS-V-6	78d	6.2-31r 3.2-7	56	B	PI-VX-78d	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-222	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing HPCS-V-5	78c	6.2-31r 3.2-7	56	B	PI-VX-78c	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	valves	R.B.	No	25
					PI-VX-223	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing RCIC-V-66	54Aa	6.2-31r 3.2-8	56	B	PI-VX-54Aa	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7	No	A	Valves	R.B.	No	25
					PI-VX-217	Globe	O	Manual	Manual	-	-	LC	LC	LC	-	1	-	<7						
Air line for testing WW-DW vacuum relief valves	82c	6.2-31r 9.3-1	56,	B	CAS-V-453	SO Globe	O	AC	Spring	44	-	C	C	C	C	1	<5	5	No	A	Valves	R.B.	No	25
					CAS-CVX-82c	Check	O	Process	Process	-	-	C	C	C	-	1	-	7						
Air line for maintenance	93	9.3-1 6.2-31t	56	B	-	Pipe Cap	I	-	-	-	-	C	C	C	-	2	-	-	No	A	Cap & Valve	S.B.	No	
					SA-V-109	Gate	O	Manual	Manual	-	-	LC	LC	LC	-	2	-	1						
Tip lines	27a-e		56	-	C51J004	SO. Ball	O	AC	AC	L,F	RM	C	C	C	C	3/8	<5	2	No	A	Valves	R.B.	No	29
					C51J004	Shear	O	-	Explosive	43	-	O	O	O	O	3/8	-	2						

AMENDMENT NO. 5
August 1979

TABLE 6.2-16 (Continued)

a. do not communicate with either the reactor coolant system or the containment atmosphere,

b. are protected against missiles and pipe whip,

c. are designed to withstand temperatures at least equal to the containment design temperature,

d. are designed to withstand the external pressure from the containment structural acceptance test, and

e. are designed to withstand the loss-of-coolant accident transient and environment.

Even if the failed closed valve were to not shut there will be no leakage of containment atmosphere through the hydraulic control lines since the piping inside the primary containment remains intact. There are no active component failures which would compromise the integrity of the closed system inside the primary containment. Integrity of the closed system inside the primary containment is, essentially, constantly monitored since the system is under a constant operating pressure of 1800 psig. Any leakage through this system would be noticed because operation would be erratic and because of indications provided on the hydraulic control unit. In addition, in order to perform type C tests on these lines, the system would have to be disabled and drained of the corrosive hydraulic fluid. This is considered to be detrimental to the proper operation of the system in that possible damage could occur in establishing the test condition or restoring the system to normal.

These lines and associated isolation valves should therefore be considered to be exempt from type C testing.

29. EVEN THOUGH ~~Since~~ the traversing incore probe (TIP) system lines do not communicate freely with the containment atmosphere ~~or the reactor coolant~~, General Design Criteria ~~55 and~~ 56 WILL BE APPLIED ~~are not directly applicable~~ to this specific class of lines BECAUSE OF AN NRC REQUEST. ~~The basis to which these lines are designed is more closely described by General Design Criterion 54, which states in effect that isolation capability of a system should be commensurate with the safety importance of that isolation. Furthermore, even though the failure of the TIP system lines presents no safety consideration, the TIP system has redundant isolation capabilities.~~

TABLE 6.2-16 (Continued)

~~The safety features have been reviewed by the NRC for BWR/4 (Duane Arnold), BWR/5 (Nine Mile Point) and BWR/6 (GESSAR), and it was concluded that the design of the containment isolation system meets the objectives and intent of the General Design Criteria.~~

Containment Isolation is accomplished by a seismically qualified solenoid-operated ball valve, which is normally closed and ~~To ensure isolation capability,~~ an explosive shear valve ~~is installed~~ in each line. Upon receipt of a signal (manually initiated by the operator), this explosive valve will shear the TIP cable and seal the guide tube.

When the TIP system cable is inserted, the ball valve of the selected tube opens automatically so that the probe and cable may advance. A maximum of five valves may be opened at any one time to conduct calibration, and any one guide tube is used, at most, a few hours per year.

If closure of the line is required during calibration, a signal causes a cable to be retracted and the ball valve to close automatically after completion of cable withdrawal. If a TIP cable fails to withdraw or a ball valve fails to close, the explosive shear valve is actuated. The ball valve position is indicated in the control room.

The WNP-2 TIP system design specifications require that the maximum leakage rate of the ball and shear valves shall be in accordance with the Manufacturer's Standardization Society (Hydrostatic Testing of Valves). The ball valves are 100% leak tested to the following criteria by the manufacturer:

Pressure	0 - 62 psig
Temperature	$340^{\circ}F$
Leak Rate	$10^{-3} cm^3/sec$

A statistically chosen sample of the shear valves is tested by the manufacturer to the following criteria:

Pressure	0 - 125 psig
Temperature	$340^{\circ}F$
Leak Rate	$10^{-3} cm^3/sec$ STP

TABLE 6.2-16 (Continued)

~~The shear valves have explosive squibs and require testing to destruction. They cannot therefore be 100% tested.~~

As stated above, the penetration is automatically closed following use. During normal operation the penetration will be open appproximately eight hours per month to obtain TIP information. If a failure occurred such as not being able to withdraw the TIP cable, the shear valve could be closed to isolate the penetrations. Installation requirements are that the guide tube/penetration flange/ball and shear valve composite assembly not leak at a rate greater than 10^{-4} std cc/sec at 80 psig. Further leak testing of the shear valves is not recommended since destructive testing would be required.

TYPE C Leak testing of the ball valves, HOWEVER, WILL BE PERFOMED IN ACCORDANCE WITH APPENDIX J OF 10CFR50. ~~also is not recommended since the guide tube terminates in a sealed indexer housing which is kept under a positive pressure by a nitrogen or air purge. The purge make-up will be indicative of the system leakage. Note that the TIP ball valve is normally closed and thus is a part of the leakage barrier being monitored. Consequently, the personnel exposure required to conduct type C tests from inside the containment is not warranted.~~

30. System is initiated after a LOCA. Isolation valves will automatically close on the following high leakage conditions:

    a. Five psi between main steam isolation valves, 60 seconds after system initiation

    b. High flow from main steam line to low pressure manifold, 150 seconds after system initiation

    c. Inboard main steam isolation valve opened, after system initiation

31. PCRVIS is not desirable since the feedwater system, although not an ESF system, could be a significant source of make-up after a LOCA which is not concurrent with a seismic event.

    Feedwater check valves on either side of the containment provide immediate leak isolation, if required. The feedwater block valves can, however, be remote-manually closed if there is no indiction of feedwater flow (see 6.2.4.3.2.1.1.1).

Q. 022.075

Provide assurance and/or justification that all valves providing containment isolation and which will be tested in the reverse direction, will be tested on the same basis as those valves for which the manufacturer's data are available to justify testing containment isolation valves in the reverse direction. Discuss your plans for having this information available for inspection on site by Region V of the office of Inspection and Enforcement.

Response:

Of the penetration designs requiring Type C testing, fifty-two incorporate isolation valves where the test pressure is not in the same direction as the pressure existing when the valves are required to perform the safety function. These isolation barriers consist of gate, globe and butterfly valves and either: 1) manufacturer's test data, 2) site test results, or 3) justification that the valve leak rates are equivallent to or greater than the leakage rate which would occur if the test pressure was applied in the normal test direction, will be maintained at the site with the quality compliance package for each subject valve.

Q. 022.077

In Section 6.2.5.2.4 of the FSAR, you state that the WNP-2 containment purge system can be used to perform a controlled purge of the containment atmosphere in the event this is necessary to limit the hydrogen concentration in the containment (e.g., following an accident). We find this approach to be acceptable provided that your purge system is capable of diluting the hydrogen concentration in the containment atmosphere at the conditions existing in the containment following a postulated LOCA (i.e., the pressure and temperature in the containment at the time hydrogen purging is required).

Response:

The containment purge system is not required as a backup to the hydrogen recombiners for post-accident hydrogen control inside containment. However, the containment purge system has the capability for a controlled purge of the containment atmosphere to aid in cleanup, per the guidance provided in Section C.4 of Regulatory Guide 1.7.

FSAR Sections 6.2.1.1.8.3 and 6.2.5.2.4 have been revised to clarify the post-accident function of the containment purge system.*

*Draft FSAR page changes attached.

All containment purge valves, including the 2" bypass valves, are designed to shut within four seconds of receipt of a containment isolation signal and to shut against full containment design pressure, 45 psig. The containment isolation signals and the purge valves are part of the containment isolation system which is an ESF system. Each purge line has two isolation valves. These valves are opened by allowing compressed air to oppose a spring in the valve actuator. On a loss of compressed air, loss of electrical signal, or on a containment isolation signal the valve is shut. If the purge system were operating at the time of a LOCA, the system will automatically be secured. The level of the activity released through the purge system before isolation would be limited to the activity present in the coolant prior to the accident since the purge system will be isolated before any postulated fuel failure could occur.

6.2.1.1.8.3 Post - LOCA

The unit coolers are not required after a LOCA since heat removal is then accomplished by the containment cooling system, a subsystem of the RHR system, as described in 6.2.2. Two 100% redundant hydrogen recombiners are available to be placed in operation to ensure that the hydrogen buildup does not reach a flammable level. Containment purge has the capability for a controlled purge of the containment atmosphere to aid in ~~hydrogen control~~ cleanup, if necessary, per the guidance provided in Section C.4 of Regulatory Guide 1.7. Any equipment located inside the primary containment which is required to operate subsequent to a LOCA has been designed to operate in the worst anticipated accident environment for the required period of time.

6.2.1.1.9 Post Accident Monitoring

A description of the post accident monitoring systems is provided in 7.5.

6.2.1.2 Containment Subcompartments

The two areas within the primary containment considered subcompartments are the area within the sacrificial shield wall and the area above the refueling bulkhead plate at elevation 583'.

The cooling water supplied to the aftercooler is returned to the standby service water system. The cooling water supplied to the scrubber is discharged to the suppression pool.

All components of the containment atmosphere control system are redundant. Controls include the control panel located in the main control room and the local control panel for each recombiner located in environmentally suitable rooms in the reactor building. All of the functions necessary to control the system are located in the main control room.

6.2.5.2.4 Containment Purge

Containment purge, discussed in 6.2.1.1.8, has the capability for a controlled purge of the containment atmosphere to aid in ~~hydrogen control~~ cleanup, if necessary, per the guidance provided in Section C.4 of Regulatory Guide 1.7.

6.2.5.3 Design Evaluation

Based on the assumptions of the model described below, it is calculated that the hydrogen concentration in the drywell eventually reaches 4% by volume approximately 10.0 hours after the postulated LOCA if the hydrogen recombiner is not in operation. The recombiner is started, however, when the hydrogen concentration reaches approximately 3.5% by volume (2.75 hours after the postulated LOCA) to limit the hydrogen concentration below 4% by volume. Figure 6.2-26 shows the drywell and suppression chamber hydrogen concentration as a function of time, with and without operation of the hydrogen recombiner system at design capacity of 150 scfm and at 105 scfm, minimum flow required to maintain the hydrogen concentration below 4% by volume.

The determination of the time dependent hydrogen concentration in the drywell and suppression chamber atmospheres is based on a two-region model of the primary containment, a drywell and a suppression chamber atmosphere.

The drywell and suppression chamber free volumes contain air and water vapor at atmospheric pressure just prior to the postulated LOCA. Gases considered available for hydrogen dilution are the non-condensibles and water vapor present during normal operating conditions. Water vapor generated from blowdown is not considered. The radiolytic generation of free oxygen is added to the total inventory of gases. The pressure in containment is assumed to remain at atmospheric pressure and the temperature history of Figure 6.2-7 curve a, is used. The hydrogen contribution from zinc and organics took no credit for dilution.

Q. 022.078

Your response to item 022.048 cited several references and tests conducted to determine the evolution of hydrogen following a postulated LOCA. We are currently undertaking additional effort to better define the various sources of hydrogen, including zincrich paints and organic materials. The following equation which describes the hydrogen generation rates as a function of temperature is currently used by the staff for its confirmatory analysis. H_2 (SCF/sq. ft. -hr) = 4.6×10^5 exp (-14500/RT) where: R (cal/gm K) = 1.986
T = absolute temperature (degrees Kelvin)
We are currently reviewing the information presented in your response to question 022.048. As an acceptable alternative approach to facilitate the staff review, provide a sensitivity study based on the above equation which shows that hydrogen concentration inside the containment will not exceed our acceptance criterion of 4 volume percent. In responding to this question, indicate the time interval following a postulated LOCA at which the hydrogen recombiner should be turned on and the amount of time needed to heat up the recombiner.

Response:

As stated in Washington Public Power Supply System letter number G02-81-181, G. D. Bouchey to D. G. Eisenhut, "Inerting of the WNP-2 Containment", dated July 16, 1981, the Supply System has committed to inert the WNP-2 containment. Since it is the oxygen concentration rather than the hydrogen concentration that must be controlled in an inerted containment, work is currently in progress to examine post-LOCA oxygen generation and to evaluate the recombiner performance in an inerted atmosphere. A detailed discussion of recombiner performance will be supplied with the January 1983 containment inerting submittal as noted in the referenced letter. For this reason and the fact that the parameters and assumptions concerning hydrogen evolution are the subject of rulemaking, sensitivity study does not need to be provided as requested.

Q. 022.079

State the seismic qualification and quality group of the water leg pumps and the associated piping which are used to maintain the water level in the pipes that you identified in your response to Item 022.049, as being filled with water at all times.

Response:

The water leg pumps and associated piping are Seismic Category I and Quality Group B. Table 3.2-1 has been modified to include those items.

TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
10. RHR System (Figure 3.2-6)							
.1 Heat exchangers, primary side	GE	2	R	B	I	I	
.2 Heat exchangers, secondary side	GE	3	R	C	I	I	
.3 Piping, within outermost isolation valves, reactor coolant pressure boundary	P	1	C,R	A	I	I	
.4 Piping, other	P	2	R	B	I	I	
.5 Pumps	GE	2	R	B	I	I	
.6 Water Leg Pumps	GE/P	2	R	B	I	I	
[.7] .6 Pump motors	GE	2	R	N/A	I	I	
.7 Valves, isolation, Reactor Coolant Pressure Boundary	P	1	C,R	A	I	I	
[.8] .8 Valves, other	P	2	R	B	I	I	(12)
[.9] .9 Mechanical modules	GE	2	R	B	I	I	
[.10] .10 Electrical modules with safety function	GE	2	R	N/A	I	I	
[.11] .11 Cable, with safety function	P	2	C,R,W	N/A	I	I	
11. Low Pressure Core Spray (Figure 3.2-7)							
.1 Piping, within outermost isolation valves to reactor vessel	P	1	C,R	A	I	I	(12)
.2 Piping, beyond outermost isolation valves	P	2	R	B	I	I	(12)
.3 Pumps	GE	2	R	B	I	I	
.4 Water Leg Pumps	GE/P	2	R	B	I	I	
[.5] .4 Pump motors	GE	2	R	N/A	I	I	
[.6] .5 Valves, isolation, Reactor Coolant Pressure Boundary	P	1	C	A	I	I	(12)
[.7] .6 Valves, other	P	2	C,R	B	I	I	(12)
[.8] .7 Electrical modules with safety function	GE	2	R	N/A	I	I	
[.9] .8 Cable, with safety function	P	2	R,W	N/A	I	I	

TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
12. High Pressure Core Spray (Figure 3.2-7)							
.1 Piping, within outermost isolation valve	P	1	C,R	A	I	I	(12)
.2 Piping, return test line to condensate storage tank beyond second isolation valve	P	G	R,O	D	II	II	
.3 Piping, beyond outermost isolation valve, other	P	2	R	B	I	I	(12)
.4 Pump	GE	2	R	B	I	I	
.5 Water Leg Pump	P	2	R	B	I	I	
.6 ~~.5~~ Pump motor	GE	2	R	N/A	I	I	
.7 ~~.6~~ Valves, beyond diesel shutoff valves	P	3	P	C	I	I	
.8 ~~.7~~ Valves, isolation, Reactor Coolant Pressure Boundary	P	1	C	A	I	I	
.9 ~~.8~~ Valves, beyond isolation valves, motor operated	GE	2	R	B	I	I	(12)
.10 ~~.9~~ Valves, other	P	2	R,P	B	I	I	
.11 ~~.10~~ Electrical modules, with safety function	GE	2	R	N/A	I	I	
.12 ~~.11~~ Electrical auxiliary equipment	GE	3	DG	N/A	I	I	
.13 ~~.12~~ Cable with safety function	P	2	W,R	N/A	I	I	

(HPCS Emergency Power Supply - see 38a)

TABLE 3.2-1 (Continued)

Principal Component (1)	Scope of Supply (2)	Safety Class (3)	Location (4)	Quality Group Classification (5)	Quality Class (6)	Seismic Category (7)	Comments
13. RCIC System (Figure 3.2-8)							
.1 Piping, within outermost isolation valves Reactor Coolant Pressure Boundary	P	1	C,R	A	I	I	(12)
.2 Piping, beyond outermost isolation valves	P	2	R	B	I	I	(12 & 23)
.3 Piping, return test line to condensate storage tank beyond second stop valve, drip pot discharge valve to condenser	P	G	R	D	II	II	(12)
.4 Pumps	GE	2	R	B	I	I	
.5 Water Leg Pumps	P	2	R	B	I	I	
.6 .5 Valves, isolation and Coolant Pressure Boundary	P	1	C	A	I	I	(12)
.7 .6 Valves, other	P	2	R	B	I	I	(13)
.8 .7 Turbine	GE	2	R	N/A	I	I	
.9 .8 Electrical modules, with safety function	GE	2	R	N/A	I	I	
.10 .9 Cable, with safety function	P	2	R,W	N/A	I	I	
14. Fuel Service Equipment							
.1 Fuel preparation machine	GE	3	R	N/A	I	I	
.2 General purpose grapple	GE	3	R	N/A	I	I	
15. Reactor Vessel Service Equipment							
.1 Steam line plugs	GE	3	R	N/A	I	I	
.2 Dryer and separator sling and head strongback	GE	3	R	N/A	I	I	
16. In-Vessel Service Equipment							
.1 Control rod grapple	GE	3	C	N/A	I	I	

Q. 022.081

In Section 6.2.6.4 of the FSAR, you discuss containment penetrations which employ a continuous leakage monitoring system and indicate they will be Type B tested at every other refueling outage but in no case at intervals greater than three years. Identify these specific containment penetrations.

Response:

Please see revised Table 6.2-14 for the information requested.*

*A draft revised page change attached.

TABLE 6.2-14

CONTAINMENT PENETRATIONS SUBJECT TO TYPE B TESTS

I ELECTRICAL PENETRATIONS

PENETRATION NUMBER	TYPE SERVICE	COMMENTS
X-100 A,B,C and D*	Neutron Monitoring	Electrical Penetrations are provided with double seals and are separately testable at 45 psig. The test taps and seals are so located that tests can be conducted without entry into nor pressurization of the primary containment.
X-101 A,B,C and D*	Control Rod Position Indicator	
X-102 A and B*	Thermocouple and RTD	
X-103 A,B, ~~and C~~ C AND D	Medium Voltage Power	
X-104 A,B,C and D*	Low Voltage Power	
X-105 A,B,C and D*	Control and Indication	
X-106 A,B,C and D	Spares	
X-107 A and B*	Low Voltage Power Control and Indication.	* PROVIDED WITH A CONTINUOUS LEAKAGE MONITORING SYSTEM

II PERSONNEL AND EQUIPMENT ACCESS PENETRATIONS

PENETRATION NUMBER	TYPE SERVICE	COMMENTS
X-15	Equipment Hatch	Separately testable at 45 psig without pressurization of the primary containment. Testing of the personnel access lock is described in detail in 3.8.2.7.5.
X-16	Personnel Access Lock	
X-28	CRD Removal Hatch	
X-51	Suppression Chamber Access Hatch	

Q. 022.082

In note 29 of Table 6.2-16 of the FSAR, you indicate that the TIP system will not be Type C tested as required by Appendix J to 10 CFR Part 50. It is our position the TIP system isolation valve should be Type C tested. Accordingly, we require you to provide a commitment consistent with our position on this matter.

Response:

Table 6.2-16 will be revised to show the TIP System isolation valve tested in accordance with Appendix J to 10 CFR 50. Please see response to Question 022.073.

Q. 022.083

The pool dynamic loads resulting from a postulated LOCA which are currently acceptable to the staff are discussed in NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria." Specifically, Table IV-1 of NUREG-0487 summarizes these acceptable Mark II pool dynamic loads. To expedite our review of the WNP-2 facility, indicate by referring to Table IV-1, which of our generic criteria will be adopted for the WNP-2 facility. Indicate the alternative criteria that you will use for each item for which an exemption is requested. Provide references which discuss these alternative criteria.

Response:

Please refer to Section 1.1 and Table 1.1-1 of the "Plant Design Assessment Report for SRV and LOCA Loads," Revision 2, for the information requested.

Q. 022.084

Provide the input data for your pool swell model, including all initial and boundary conditions. Demonstrate that the model input represents conservative values of the initial and boundary conditions (i.e., those values which will yield maximum pool swell loads). In the case of input which is calculated (i.e., the drywell pressurization and the vent clearing time), describe and justify your calculational methods.

Response:

The pool swell model, including all initial and boundary conditions, conservative assumptions and calculational methods are discussed in length in Section 3.2 of the "Plant Design Assessment Report for SRV and LOCA Loads," Revision 2.

Q. 022.085

Provide in graphic form, the following information:

a. the pool surface velocity versus position; and

b. the maximum pressures of the suppression pool air slug and the wetwell air space.

Response:

a. Plots of the wetwell pool surface motion during a LOCA are presented in Figures 3.2-22 (Velocity vs. Time), 3.2-23 (Acceleration vs. Time), 3.2-24 (Elevation vs. Time), and 3.2-25 (Velocity vs. Elevation) of the "Plant Design Assessment Report for SRV and LOCA Loads, " Revision 2.

b. The maximum pressures of the suppression pool air (bubble) slug and the wetwell air space are presented in Figures 3.2-25 and 3.2-26, respectively, of the above referenced document.

Q. 022.086

If your pool swell model is significantly different from the pool swell model previously found acceptable by the staff, compare your calculated drywell pressure response and the enthalpy flux in the downcomer vent with the data obtained from the series of tests conducted in the 4T facility using the 2 1/2-inch and 3-inch venturis.

Response:

The WNP-2 pool swell model conforms to that previously found acceptable by the NRC. Please refer to Table 3.2-2, "Short Term LOCA Hydrodynamic Load Summary Table" in Section 3.2 of the "Plant Design Assessment Report for SRV and LOCA Loads," Revision 2.

Q. 022.087

Provide the information requested in Items 022.084, 022.085, and 022.086, where applicable, for pool swell in the pedestal region.

Response:

Since no structures or downcomers are located in the pedestal region, pool swell effects there are not considered.

Q. 022.088

Your performance tests on a scaled down model of the catalyst bed for the WNP-2 hydrogen recombiner, were conducted in a laboratory test facility in which the gas flow rates and size of the catalyst bed are significantly different than those which will be used in the production model. Indicate the scaling factors used in determining the size of the catalytic bed and the gas flow rates when you established the design of the full-scale recombiner. Provide justification for these scaling factors, including a discussion of the catalyst bed volume, the bed depth, the bed area and any experimental verification of the adequacy of these scaling factors. Additionally, we require you to conduct full-scale tests on a production recombiner unit including the catalyst, to demonstrate that the hydrogen recombiner will perform its intended function in the containment environment which would occur following a postulated LOCA.

Response:

Scaling factors used in determining the size of the catalytic bed and gas flow rates for the full-scale recombiner are discussed in detail in Section 3.1 of APCI-78-6, "Air Products Post-LOCA Recombiner Test Summary", dated June 1978. (APCI-78-6 was submitted to the NRC on the WNP-2 docket by Washington Public Power Supply System letter G02-78-176, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated July 10, 1978.)

As stated in Section 6.2.5.4 of the FSAR, full-scale performance tests of the WNP-2 recombiners were performed. Testing parameters duplicated the feed gas pressure, temperature, flow, steam content, hydrogen and other gas contents expected during a postulated LOCA. At no time was the efficiency of hydrogen recombination less than 99%. Detailed information relating to the recombiner tests may be found in Section 2.0 of APCI-78-6.

Q. 022.089

You conducted catalyst performance tests using a feed gas composition which did not contain steam. Investigate the effect on the various components of the recombiner system and the overall effect on system performance, of having superheated steam in the feed gas.

Response:

As stated in Section 6.2.5.4, each hydrogen recombiner system has been shop tested. The full scale performance tests were performed using a feed gas composed of air, hydrogen, and steam at simulated pressure and temperature conditions following a postulated LOCA. At no time during the test was the efficiency of hydrogen recombination less than 99%.

Detailed information relating to the hydrogen recombiner full scale performance tests is provided in the follwing Air Products and Chemicals, Inc. reports:

1.) APCI-78-6, Air Products Post-LOCA Recombiner Test Summary, dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-78-176, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated July 10, 1978.)

2.) APCI-78-6P, Air Products Post-LOCA Recombiner Test Summary (Proprietary Supplement to APCI-78-6), dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-78-200, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated August 11, 1978.)

Q. 022.091

Since the hydrogen recombiner may be required to operate for months following a postulated LOCA, justify the length of time you conducted the performance tests to qualify the production model's capability to perform for extended periods of time.

Response:

All components of the recombiner skid, control panel and mounted instrumentation were designed for continuous operation when subjected to accident environmental conditions of temperature, pressure, humidity, radiation, and seismic event, following 40 years of periodic cyclic testing under normal environmental conditions.

Purchased components had to be tested or analyzed at or above the given service conditions and, in the event of insufficient support data, additional testing was performed. Individual components were then installed in the same configuration as they had been previously qualified, which allowed correlation of test results to the production units.

IEEE reliability qualification is discussed in Section 5.0 of the Air Products and Chemicals, Inc. report APCI-78-6 titled Air Products Post-LOCA Recombiner Test Summary, dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter Number G02-78-176, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated July 10, 1978.)

Detailed description of IEEE reliability qualification including actual documentation and test reports is provided in the Air Products and Chemicals, Inc. report titled IEEE Reliability Qualification Report, Revision A, dated January 7, 1980. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-80-201, D. L. Renberger to B. J. Youngblood, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated September 16, 1980.)

In addition to the reliability qualification work already performed, documented, and submitted to the NRC, the hydrogen recombiner system is currently being evaluated per NUREG 0588 criteria.

Q. 022.092

The staff guidance in Regulatory Guide 1.7 indicates that equipment for measuring and sampling containment atmosphere should be disigned to appropriate engineered safety feature criteria; i.e., seismic Category I and Quality Group B. Indicate whether the measuring and sampling equipment in the WNP-2 facility conforms to this guidance. Indicate whether the hydrogen analyser is part of the recombiner package and state whether the analyzer catalyst is the same as the recombiner catalyst.

Response:

The WNP-2 containment hydrogen concentration and oxygen concentration are continuously monitored during normal operation and following a postulated LOCA by analyzers that meet the staff guidelines in Regulatory Guide 1.7. Additionally, the hydrogen/oxygen analyzers and associated indications comply with requirements of NUREG 9737. See FSAR Appendix B, item II.F.1

The hydrogen/oxygen analyzers are not part of the recombiner package and operate independently from the recombiners. The analyzers are of the thermal conductivity type and do not utilize a catalyst.

Q. 022.093

Provide the results of the testing program for the blower, the preheater, the after-cooler, the water jet eductor and the separator, which demonstrate that these components will perform their intended functions in the containment environment which would occur following a postulated LOCA. In your response, include a table of the test parameters and the range over which these parameters were varied. Our position is that your test program should consider the effects of the following variables: a. irradiation of all components including electrical equipment; b. seismi conditions; c. thermal cycling of the equipment and the catalyst bed; d. the temperature of the components and the effluent gas; e. the air flow rate; f. the inlet hydrogen concentration; g. the fission products and their potential for leakage; and h. the steam content.

Response:

The WNP-2 hydrogen recombiner system does not utilize a water jet eductor. However, a comprehensive testing program was performed for the system components as well as a full scale test of the integrated recombiner system under simulated environmental conditions that would occur following a postulated LOCA.

Results of the testing program are provided in the referenced Air Products and Chemicals, Incorporated report:

a. Radiation

IEEE Reliabiltiy Qualification Report, Revision A, dated January 7, 1980. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-80-201, D. L. Renberger to B. J. Youngblood, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated September 16, 1980.)

b. Seismic Conditions

IEEE Reliability Qualification Report, Revision A.

Dynamic Testing Report for Hydrogen Recombiner System dated January 3, 1980. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-80-201, D. L. Renberger to B. J. Youngblood, "Post LOCA hydrogen Recombiner Supplemental Information", dated September 16, 1980.)

APCI-78-6, Air Products Post-LOCA Recombiner Test Summary, dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-78-176, D. L. Renberger to B. J. Youngblood, "Post LOCA hydrogen Recombiner Supplemental Information', dated September 16, 1980.)

Addenda No. 1 to Air Products Post-LOCA Recombiner Test Summary Report No. APCI-78-6P, dated September 19, 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-78-223, D. D. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated September 18, 1978.)

c. Thermal Cycling

Thermal Cycle Test Performance for the Hydrogen Recombiner System, dated December 14, 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-80-201, D. L. Renberger to B. J. Youngblood, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated September 16, 1980.)

d. Temperature

IEEE Reliability Qualification Report, Revision A.

e. Air Flow Rate

APCI-78-6, Air Products Post-LOCA Recombiner Test Summary.

g. Fission Products and Potential Leakage.

APCI-78-6P, Air Products Post-LOCA Recombiner Test Summary (Proprietary Supplement to APCI-78-6), dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-78-200, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated August 11, 1978.)

All pressure containing equipment including piping between components is considered an extension of primary containment and is subject to appropriate leakage rate testing.

h. Steam Content.

APCI-78-6, Air Products Post-LOCA Recombiner Test Summary.

Full scale performance tests of the actual WNP-2 recombiner systems, each consisting of a control

panel and skid-mounted process component package, demonstrated a hydrogen recombination efficiency of greater than 99.0% in all simulated post-LOCA conditions. A table of the test parameters and the range over which these parameters were varied is provided in APCI-78-6, Table 2-1. A summary of performance test objectives can be found in Table 2-3 of the same report.

Q. 022.094.

Clearly identify the interfaces between the hydrogen recombiner and the plant design, including a discussion of:

a. whether the recombiner can be operated from the reactor control room;

b. the instrumentation which will be available to the control room operator to permit the operator to monitor the recombiner performance; and

c. any special equipment or power supply needed for the operation of the recombiner.

Response:

Interfaces between the hydrogen recombiner and the plant design are identified on Figure 3.2-17 and discussed in detail in Section 6.2.5. Specifically:

a. As stated in Section 6.2.5.1.j:

"The system is designed to operate remotely from the main control room which includes monitoring of combustible gas concentration. The presence of personnel in the vicinity of the operating hydrogen recombiner units is not required."

b. All of the function's necessary to monitor the recombiner performance are located in the main control room and include:

- Process gas flow rate;
- Scrubber water flow rate;
- System pressure at the blower;
- Temperatures at the blower exit, preheater, and recombiner catalytic bed;
- Recycle flow rate.

Primary containment hydrogen and oxygen concentrations are monitored by each of two redundant analyzers that draw samples of the containment atmosphere from both the drywell and wetwell. The analyzers are discussed in detail in Section 7.5.1.5.

c. The recombiners are supplied with redundant Class 1E power. The safety grade cooling water supply, Standby Service Water, is placed into operation by the same signals which startup the ECCS.

Q. 022.095

You state in your proposal for the hydrogen recombiner that the recombiners will be remotely operated. However, you do not discuss the need for gaining access to the combustible gas control equipment area following apostulated LOCA. Discuss the necessity and/or requirements for such access and your proposed criteria for potential radiation exposure to operating personnel. Indicate how you considered these requirements and criteria when you selected suitable locations for the combustible gas control equipment.

Response:

As stated in FSAR section 6.2.5.1.j, the hydrogen recombiner system is designed to operate remotely from the main control room which includes monitoring of combustible gas concentration. The presence of personnel in the vicinity of the operating hydrogen recombiner units is not required.

Q. 022.097

You state on page 68 of your report, APCI-78-6P, that a hydrogen concentration level of 2.5 percent is representative of the hydrogen containment atmosphere. We find that a much higher hydrogen concentration could exist in the containment prior to initiating operation of the recombiner. Accordingly, discuss the applicability of the test you performed using a hydrogen concentration of 2.5 percent in light of the maximum anticipated hydrogen concentration inside containment.

Response:

Section 3.3.4.d of APCI-78-6P describes a unique test that demonstrates the effects of feed gas preheat temperature and hydrogen concentration on catalyst activity in the presence of extremely severe halogen concentration. The hydrogen removal efficiency was recorded over a preheat temperature range of 450°F - 560°F with a feed gas hydrogen concentration was then increased to 2.5% and the hydrogen removal efficiency was tested over the same preheat temperature range.

The results of this test are shown in Figure 3-15 of APCI-78-6P and confirm that a 550°F design preheat temperature is conservative even with massive influent iodine concentrations.

APCI-78-6P does not state that a hydrogen concentration level of 2.5% is representative of the hydrogen containment atmosphere. Rather, in the context of comparing 1.1% versus 2.5% hydrogen concentration levels over the same preheat temperature range, it is stated that a hydrogen concentration of 2.5% is more representative of a containment atmosphere with hydrogen accumulation.

Full scale performance tests with up to 4% hydrogen in the feed gas successfully demonstrated a hydrogen removal efficiency of greater than 99.0%. Details of the full scale performance test may be found in Air Products and Chemicals, Incorporated report APCI-78-6, Air Products Post - LOCA Recombiner Test Summary, dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-78-176, D. L. Renberger to S. A. Varga, "Past-LOCA Hydrogen Recombiner Supplemental Information," dated July 10, 1978.)

Q. 022.101

In reviewing your report, APCI-73-10, we find that only iodine and methyl iodide were used in the tests conducted to determine the effect potential poisonous materials on the catalyst. Other materials were not tested because of similar work done by Southern Nuclear Engineering (SNE) and reported in SNE-100. With regard to these tests:

a. Justify why the noble gases and their decay products were not tested, since they will come in contact with the catalyst;

b. Justify why solvents, such as potassium hydroxide and sodium peroxide which dissolve platinum compounds, were not tested;

c. The tests conducted by SNE used a flow velocity which is significantly different than the design flow velocity in the WNP-2 hydrogen recombiner (i.e., the Air Products model). Justify the applicability of the SNE tests to the Air Products hydrogen recombiner;

d. In the SNE tests which were of short duration, various materials which poison the catalyst were found to reduce the efficiency of the catalyst by zero to 17 percent. Discuss the possibility that this performance degradation will increase with time;

e. It is reported that various poisonous materials have only a slight effect on the efficiency of the catalyst. Discuss the cumulative effect on the efficiency of the catalyst, of all the poisons tested;

f. The argument is made in concluding that various poisons will have only a slight effect on the efficiency of the catalyst, that tests were conducted using poison concentrations well in excess of those predicted in containment after a LOCA. However, tests conducted with methyl iodide do not support this argument. In the methyl iodide tests, the poisoning effect did not change as the concentration was reduced. Discuss the possibility of this same effect occurring with other poisons, including the poison which caused a 17 percent reduction in catalyst efficiency;

g. The miscellaneous halide test conducted by SNE showed that the efficiency of the catalyst could be reduced from 50 to 95 percent.

Response:

The response to NRC Question 022.101 is provided in the Air Products and Chemicals, Incorporated report titled "Attachable Amendment No. 1 - Response to Request for Additional Information - Topical Report - Air Products Catalytic Recombiner System - APCI-74-4" dated October 24, 1974 (reference item No. 3, page 4). This report was submitted to the NRC by an Air Products and Chemicals, Incorporated letter to O. D. Parr, Chief, Light Water Reactors Project Branch 1-3, Division of Reactor Licensing, dated October 25, 1974.

Q. 022.102

Indicate whether all potential catalyst poisons have been tested and justify your response. Demonstrate that the potential catalyst poisons will not have a detrimental effect on the catalyst.

Response:

The response to NRC Question 022.102 is provided in the Air Products and Chemicals, Incorporated report titled "Attachable Amendment No. 1 - Response to Request for Additional Information - Topical Report - Air Products Catalytic Recombiner System - APCI-74-4" dated October 24, 1974 (reference item No. 4, page 10). This report was submitted to the NRC by an Air Products and Chemicals, Incorporated letter to O. D. Parr, Chief, Light Water Reactors Project Branch 1-3, Division of Reactor Licensing, dated October 25, 1974.

Q. 022.103

The concentrations of iodine and methyl iodide used in the tests reported in APCI-73-10 appear to be quite low when compared to the concentrations expected in containment following a postulated LOCA. State the assumptions which you used to determine the concentrations you used and provide appropriate references. In your response, include a discussion of both radioactive and stable isotopes.

Response:

The response to NRC Question 022.103 is provided in the Air Products and Chemicals, Incorporated report titled "Attachable Amendment No. 1 - Response to Request for Additional Information - Topical Report - Air Products Catalytic Recombiner System - APCI-74-4" dated October 24, 1974 (reference item No. 5, page 12). This report was submitted to the NRC by an Air Products and Chemicals, Incorporated letter to O. D. Parr, Chief, Light Water Reactors Project Branch 1-3, Division of Reactor Licensing, dated October 25, 1974.

Q. 022.104

In your discussion in APCI-73-10 of the effect of particulates on recombiner performance, the only solids you considered were the daughter products of radioactive decay of xenon, krypton, iodine and bromine. Additionally, you indicate that 25 percent of the iodine and bromine was assumed to be released to the containment, in accordance with Regulatory Guides 1.3 and 1.4. However, these regulatory guides are to be used in evaluating the radiological consequences following a postulated LOCA; they are not conservative when evaluating the amount of hydrogen released. Accordingly, analyze the effect of particulates on the performance of the hydrogen recombiner if 100 percent of the noble gases, 50 percent of the halogens and one percent of the solids present in the reactor core are released to the containment, as stated in Regulatory Guide 1.7.

Response:

The response to NRC Question 022.104 is provided in the Air Products and Chemicals, Incorporated report titled "Attachable Amendment No. 1 - Response to Request for Additional Information - Topical Report - Air Products Catalytic Recombiner System - APCI-74-4" dated October 24, 1974 (reference item No. 7, page 14). This report was submitted to the NRC by an Air Products and Chemicals, Incorporated letter to O. D. Parr, Chief, Light Water Reactors Project Branch 1-3, Division of Reactor Licensing, dated October 25, 1974.

Q. 022.105

Provide assurance that the catalyst bed will not deteriorate over the 40-year lifetime of the plant. Consider both chemical deterioration of the catalyst and "packing" of the bed.

Response:

The response to NRC Question 022.105 is provided in the Air Products and Chemicals, Incorporated report titled "Attachable Amendment No. 1 - Response to Request for Additional information - Topical Report - Air Products Catalytic Recombiner System - APCI-74-4" dated October 24, 1974 (reference item No. 9, page 16). This report was submitted to the NRC by an Air Products and Chemicals, Incorporated letter to O. D. Parr, Chief, Light Water Reactors Project Branch 1-3, Division of Reactor Licensing, dated October 25, 1974.

Q. 022.106

Provide a description of the dynamic testing proceures used in establishing the design of the WNP-2 hydrogen recombiner system to withstand vibratory loads arising form seismic events, postulated accidents and other causes (e.g., pool dynamic loads). Describe the methods and procedures you employed to calculate the frequency spectra and amplitudes at the equipment supports of this system. If your analyses and/or testing procedures do not include evaluation of the equipment in the operating mode, indicate how you will assure that this equipment will function when subjected to the combination of seismic loads, accident loads and other vibratory loads. Criteria acceptable to the sfaff for a seismic qualification program is contained in Sections 3.9.2 and 3.10 of the Standard Review Plan.

Response:

The WNP-2 recombiner skid and control panel have been dynamically tested on a shaker table. The testing was performed in accordance with the recommended practices of IEEE-344, 1975. The results of the testing verified the recombiner system's ability to withstand vibratory loads arising from seismic events, postulated accidents, and pool dynamic loads. The recombiner system operated satisfactorily before, during, and after the dynamic tests.

A detailed discussion of the recombiner system dynamic test program, including the test procedure and response spectra, is provided in the Air Products and Chemicals, Inc. report titled "Dynamic Testing Report for Hydrogen Recombiner System", dated January 3, 1980. (This report was submitted to the NRC by Washington Public Power Supply System letter number GP2-80-201, D. L. Renberger to B. J. Youngblood, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated September 16, 1980.)