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VARGA,S.A. LIGHT WATER REACTORS BRANCH 4

SUBJECT: FORWARDS RESPONSES TO ROUND ONE,SET THREE QUESTIONS FROM AUXILIARY SYS BRANCH. RESPONSES WILL BE INCORPORATED IN FSAR. (see reports)

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Washington Public Power Supply System
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May 16, 1979
G02-79-99

Docket No. 50-397

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

Attention: Mr. S. A. Varga, Chief
Branch No. 4
Division of Project Management

Subject: WPPSS NUCLEAR PROJECT NO. 2
RESPONSES TO ROUND ONE
QUESTIONS, SET THREE - ASB

Reference: Letter, S. A. Varga (NRC) to N. Strand (WPPSS), "First Round Questions on WNP-2 OL Application - ASB," dated January 13, 1979.

Dear Mr. Varga:

Attached please find sixty (60) copies of the responses to the round one, set three questions representing the Auxiliary Systems Branch. Also included are the responses to a few open items from a previous set. The responses to these questions will be incorporated formally into the FSAR in an amendment within four months.

Very truly yours,

DL Renberger
D. L. RENBERGER
Assistant Director
Technology

DLR:SAG:sg

Attachment: Responses to Round 1 Questions (60)

cc: I. Littman - WPPSS, NY - wo/att
JJ Verderber - B&R, NY - "
JJ Byrnes - B&R, NY "
RC Root - B&R, Site "
HR Canter - B&R, NY "
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FA MacLean - GE, San Jose " (1)
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WNP-2 Files - w/att (1)

B001 SE 60/60

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STATE OF WASHINGTON)
) ss
COUNTY OF BENTON)

D. L. RENBERGER, Being first duly sworn, deposes and says: That he is the Assistant Director, Technology, for the WASHINGTON PUBLIC POWER SUPPLY SYSTEM, the applicant herein; that he is authorized to submit the foregoing on behalf of said applicant; that he has read the foregoing and knows the contents thereof; and believes the same to be true to the best of his knowledge.

DATED May 15, 1979

D L Renberger
D. L. RENBERGER

On this day personally appeared before me D. L. RENBERGER to me known to be the individual who executed the foregoing instrument and acknowledged that he signed the same as his free act and deed for the uses and purposes therein mentioned.

GIVEN under my hand and seal this 15th day of May, 1979.

Daniel C. Roehl
Notary Public in and for the State of Washington
Residing at Richland

WNP-2

Responses to:

Auxiliary Systems Branch Questions
(10.10 - 10.34)

Q. 010.10
(3.4.1)

Demonstrate that all piping and electrical penetrations in safety-related structures that are below the level of the Probable Maximum Flood, are water tight.

Response:

As stated in 3.4.1.4.1 the plant site grade is higher than the design basis flood elevation resulting from the probable maximum precipitation (PMP) event. Due to the short duration of the PMP flood, the ground water level at the plant site is not affected. As stated in 3.4.1.4.2, piping and electrical penetrations are above the design basis ground-water level and are therefore not sealed against groundwater pressure.

Q. 010.11
(3.5)

We require you to provide an evaluation of the environmental effects resulting from a postulated failure of the main steam lines and the main feedwater line. Your evaluation should demonstrate conformance with our requirements that:

a. Those compartments and tunnels which house the main steam lines, the feedwater lines, including the isolation valves for these lines, are designed to withstand the environmental effects (pressure, temperature and humidity) and the potential flooding resulting from a postulated crack equivalent to the flow area of a single-ended pipe rupture in these lines.

b. The essential equipment located within these compartments, including the main steam line isolation valves and the feedwater valves and their associated valve operators, are capable of operating in the environment resulting from the crack postulated in Item (a) above.

c. If the forces resulting from this postulated crack could cause the structural failure of these compartments, the consequent failure of these compartments will not jeopardize the safe shutdown of the plant.

d. The remaining portion of the pipe in the tunnel between the outboard safety valve and the Turbine Building meet the guidelines of Branch Technical Position ASB 3-7, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment", with respect to the stress levels in this portion of the pipe and with respect to the location of the postulated break points.

We further require that you submit an analysis of the sub-compartment pressure buildup following a postulated pipe break, including the structural evaluation of the affected sub-compartments, to demonstrate that the design of the pipe tunnel conforms with our positions as stated above. If you cannot demonstrate conformance with our positions in this matter, indicate any design changes which may be required to comply with our positions. This evaluation should demonstrate that the methods used to calculate the pressure transient in the sub-compartments outside of the primary containment are the same as those for sub-compartments inside the containment for postulated pipe break. Demonstrate that the margin against a structural failure resulting from the pressure transient, are the same as those in sub-compartments inside the primary containment. If you propose to use methods of analysis for sub-compartments outside of containment which are different from those used inside containment, demonstrate that the methods of analysis for sub-compartments outside containment

assure adequate design margins. Identify the computer codes and the assumptions regarding the mass and energy release rates which you used in your analysis. Provide sufficient design data so that we may perform independent calculations.

Response:

The complete response to this question will be supplied in July 1979. The structural adequacy of the steam tunnel and the environmental conditions in the steam tunnel following a pipe break in a main steam line were previously evaluated based on a double-ended guillotine pipe break and instantaneous venting of the blowout panels. This analysis is currently being reevaluated using using the RELAP 4 code in line with the conditions in this question including single-ended breaks for both feedwater and main steam. It is expected that the results from the original analysis based on a double-ended break of the main steam line will be shown to be bounding.

Q. 010.12

Provide the results of your evaluation of the jet impingement forces and the environmental effects, including pressure, temperature, humidity, and flooding, resulting from a postulated failure of the main steam and main feedwater systems in the turbine building. This evaluation should address only those safety-related components, systems and structures, if any, in (or immediately adjacent to) the turbine building (e.g., the walls of the auxiliary building).

Response:

It has been determined that the only items with safety-related functions in the Turbine Building are some RPS sensor inputs from the Main Steam System, MSIV isolation logic inputs from the Main Steam System, and the Tower Make-up Transformers located in the basement of the Turbine Building which are required to function only for the Design Basis Tornado event. This last item is remote from the steam and feedwater lines (being located at the basement grade level of the building) and has been evaluated to have adequate protection from tornado missiles and internal flooding (see the responses to questions 10.25 and 10.34*). In addition, there is cabling for the condensate storage tank level sensors whcih provide for auto-switching of HPCS from the storage tank to the suppression pool. The routing of this cabling is currently through the turbine building, but is under design review to insure its adequate protection from accidents. Appropriate design changes will be made as a consequence of this evaluation. Accordingly, the only items of concern are the RPS and MSIV isolation logic sensor inputs. Due to their nature they cannot be made immune from pipe-break effects. However, no analysis has been performed of the specific effects of a steam line or feedwater break in the Turbine Building on this equipment since it has been determined that the complete loss of all this equipment could occur for these events without the loss of capability to bring the plant to a cold shutdown or mitigate the radiological consequences of such an incident even assuming a single failure in the safety systems that remain unaffected.

The electrical cable connected with this safety related equipment in the corridors separating the Turbine Building, Reactor Building, and Radwaste Building would be exposed to temperatures and pressure effects of a postulated failure of the main steam or feedwater lines in the Turbine Building, but the exposure conditions would be for less than the design environmental requirements contained in the purchase specifications for the cable.

*10.34 is a circulating water break which is conservative for a flooding event.

No other safety-related equipment is located in an area which would be vulnerable to the environmental effects of a pipe break in the Turbine Building. The only safety related structures adjacent to the Turbine Building are the Reactor Building and Radwate-Control Building. A pipe break in a main steam or feedwater line in the Turbine Building would result in transitory pressurization of the corridors between the Turbine Building, Reactor Building, Radwaste-Control Building, and Diesel-Generator Building. Air and steam would be forced into these corridors through openings in the south wall of the Turbine-Generator Building, and through the seismic gap between the Turbine Building, Reactor Building, and Radwaste-Control Building. No compartmental pressurization analysis is required to determine peak pressures and temperatures in the corridors due to the large volume of the Turbine Building, and the fact that the metal siding and exterior doors into the Turbine Building are not leak-tight and are not designed to withstand more than a minimal pressure differential, the peak pressures seen by the reinforced concrete walls of the Reactor Building and Radwaste-Control Building would not exceed the structural capacity of the walls. The doors to the control room are low-range blast doors, designed to withstand a pressure differential of 3 pounds per square inch, which is considered adequate to maintain control room habitability as discussed in 3.6.1.12.

It should be noted that the response to this question is directed towards the Turbine Building as a whole and does not cover the steam tunnel. The response to question 10.11 will address this area.

Q. 010.13
(3.6)

For postulated pipe breaks, you have not provided the information required to determine:

1) The mechanism which terminates the resulting blowdown; or,
2) The period of time over which blowdown occurs.

Accordingly, for each postulated pipe break or leakage crack indicate the time over which blowdown occurs and identify the mechanism which either terminates the blowdown or limits the amount of blowdown flow. These mass and energy flow rates will be used to evaluate the peak pressures and temperatures in compartments and structures following a postulated break of the high energy pipes inside these structures.

Response:

Except for the main steam isolation valves which terminate blowdown flow from the reactor building side of pipe breaks in the main steam line, and check valves in the reactor feedwater lines, which terminate blowdown flow from the reactor building side of pipe breaks in the reactor feedwater lines, no mechanism terminates flow except exhausting of the inventory of fluid in the line following the pipe break.

Where blowdown flow is not automatically terminated by isolation valves or check valves as described above, the duration of the blowdown event as the inventory of fluid in a line is exhausted is not considered in the analysis of peak compartmental pressure and temperature. To evaluate the peak pressures and temperatures in compartments and structures following a postulated break of the high energy pipes inside these structures, the blowdown analysis is extended far beyond the initial transient until the blowdown flow becomes steady or decreases continuously. The duration of the analysis is therefore sufficient to correctly predict the peak pressures and temperatures in these compartments and structures.

For a postulated pipe break or leakage crack in the main steam lines outside primary containment, the flow from the reactor side of the break is terminated by the closing of the main steam isolation valves in each of the four main steam lines. The main steam isolation valves start to close at 0.5 seconds after the break and are fully closed at or prior to 5.5 seconds after the break, as given in Table 15.6-6.

For a postulated break or leakage crack in the reactor feedwater lines outside primary containment, the flow from the reactor side of the break is terminated by the closing of the check valves in each of the two reactor feedwater lines. The check valves start to close when the direction of the flow reverses, and the flow from the reactor side of the break is therefore terminated within a fraction of a second.

Q. 10.14
(3.6)

You state in Section 3.6.1.1.1 of the FSAR, that fluid piping systems which the staff would classify as high-energy lines are considered by you to be moderate-energy systems if: (1) their fluid temperatures are below 200°F and; (2) the fluid pressure is generated by centrifugal pump instead of a fluid reservoir. (The staff classification system states that the fluid temperature must be less than 200°F and the fluid pressure must be less than 275 psi for a system to be designated as moderate-energy.) Accordingly, demonstrate that these sytems do not contain enough energy to cause pipe whip. Additionally, provide justification for your analysis of flooding based on the moderate-energy crack criteria rather than basing your analysis on the full break required by the high-energy break criteria.

Response:

The energy of the blowdown fluid from a break in a pressurized fluid system is a function of the pressure at the exit plane, the mass flow rate and the area of the fluid jet. The blowdown process of the 200°F water from high pressure to atmospheric pressure can be considered as adiabatic. Since the water is subcooled, it will not flash during the compression transient of the blowdown. Therefore, the water jet remains in the liquid phase and behaves like an incompressible fluid.

At the beginning of the decompression transient, immediately following the break, a decompression wave is formed and travels through the fluid at sonic velocity (approximately 5,142 ft/sec) to the pressure source which, in this case, is the centrifugal pump. Due to the reduction in required head, the flow rate accelerates rapidly increasing accordingly to the characteristics of the piping system until a new equilibrium is established. Since the system operating pressure is derived solely from the centrifugal pump, the complete system is depressurized after the break and the energy supplied to the pump is completely transmitted to the fluid in terms of velocity head. For an incompressible fluid in an open system, the energy of the water jet is proportional to the velocity head only. Hence, the thrust of the water jet from a break in this class of piping systems may be calculated at the exit plane of the jet using the following formula:

$F = \rho AV^2/g_c$,

Where F = jet force normal to target, Lb_f

ρ = fluid density, Lb_m/ft^3

A = flow path cross-sectional area, ft^2

V = velocity of jet, ft/sec

g_c = 32.179 Lb_m - ft/Lb_f - Sec^2

Maximum jet thrust may be obtained at the exit plane of the pump. Assuming a friction coefficient of 1.5 to include only contraction and expansion losses, two representative examples representing the highest head and highest flow cases in the plant are presented below to demonstrate that this class of high pressure systems do not cause pipe whip events if the system pressure is derived from a centrifugal pump.

	CRD Pump Discharge	Condensate Pump Discharge*
Piping Designation	2" CRD (2) - 4	20" Cond (2) - 1
Pipe Schedule	160	40
Operating Pressure	1,439 psig	142 psig
Operating Temperature	100°F	109.4°F
Jet Flow	284 gpm	19,000 gpm
Jet Force	50 Lb_f	1,800 Lb_f
Jet Pressure at Break Plane	22.3 psi	6.5 psi

*By pressure and temperature criteria, this is classified as a moderate energy break. However, in question 110.18 the NRC requested consideration of condensate piping as a high energy system.

For the maximum 10' span which exists between the pipe supports the above pressures result in pipe stresses which are below the minimum for formation of a plastic hinge and thus pipe whip will not occur.

Therefore, for piping systems with system pressure derived from a centrifugal pump, the system is treated as a moderate energy system and flooding analysis is performed based on postulated flow from a controlled leakage crack.

Q. 010.15
(RSP)

We require that you modify the main steam line isolation valve leakage control system (MSIV-LCS) to satisfy the staff positions contained in Regulatory Guide 1.96, Rev. 1, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants", June 1976. Specifically, we require that:

a) The design of the MSIV-LCS permits its actuation within 20 minutes after a postulated loss-of-coolant accident.

b) The leakage control system for the valve stems on the main steam line be designed to the same standard as the MSIV-LCS, and

c) Operation of the MSIV-LCS during normal plant operation be prevented by inter-locks capable of functioning after a postulated single failure in the inter-locking system.

Response:

a) See revised page 6.7-2.*

b) See revised page 6.7-11.*

Also: (direct response)

Stem packing leakage from the outboard main steam isolation valves is directed to equipment drain funnels located in the steam tunnel. The leak-off piping is classified Nuclear Class 2 up to and including the first manual block valve. As stated in 6.7.3m, leakage from the packing seals large enough to pressurize the steam tunnel and blow out the water seal traps in the equipment drain system would vent into areas of the reactor building for subsequent processing by the standby gas treatment system. Refer to Figure 3.2-2, Zone G2, which depicts the stem packing leakage piping.

c) Refer to Question 031.076 response.

*draft page attached

b. The MSIV-LCS and necessary subsystems are capable of performing their safety function, when necessary, considering the design basis LOCA effects including: (1) internally generated missiles, (2) the dynamic effects associated with pipe whip and jet forces from the event and (3) normal operating and accident-caused local environmental conditions consistent with the event.

c. The MSIV-LCS is capable of performing its intended function following any single active component failure (including failure of any one of the main steam line isolation valves to close).

d. The MSIV-LCS is capable of performing its intended function following a loss of all off-site power coincident with the postulated design basis LOCA.

e. The MSIV-LCS is designed with sufficient capacity and capability to control the leakage from the main steam lines consistent with containment integrity under the conditions associated with the postulated design basis LOCA.

10.15a. f. The MSIV-LCS is manually initiated and is designed to permit actuation ~~in a time period~~ at any time ~~no sooner than 10 minutes~~ following the postulated design basis LOCA. ~~The required actuation time period is consistent with loading requirements on the critical electrical buses and allows reasonable time for operator information, decision, and action~~.

g. Instrumentation and controls necessary for the functioning of the MSIV-LCS are designed in accordance with standards applicable to nuclear plant safety-related instrumentation and control systems.

h. The MSIV-LCS controls are provided with interlocks actuated from appropriately designed safety systems or circuits to prevent inadvertent MSIV-LCS operation.

l. Steam leaks into the steam tunnel escape the steam tunnel through the equipment drain system and are directed to the reactor building where the radioactive gases are subsequently processed by the standby gas treatment system.

m. The MSIV-LCS does not process MSIV stem packing leakage. Stem packing leakage from the outboard main steam isolation valves is directed to equipment drain funnels located in the steam tunnel. These equipment drains are routed to the reactor building equipment drain sump. Low leakage from the stem packing would condense in the piping to the equipment drain. Leakage large enough to pressurize the steam tunnel and blow out the water seal traps in the equipment drain system would vent into areas of the reactor building for subsequent processing by the standby gas treatment system.

The stem packing leakage potentially of concern is from the outboard MSIV located in the steam tunnel.

n. All interconnections between MSIV-LCS and other plant systems do not affect the intended function of the MSIV-LCS. These interconnections and their safety related actions are as follows:

(1) Inlet 1½"MSLC(2)-4 lines for each inboard main steam isolation valve share common 1½"MS(9)-4 drain lines. Motor operated drain valves MS-V-67 A through D close automatically by the containment isolation system on a scram signal. Thus these lines would be isolated prior to placing the MSIV-LCS in operation after a LOCA.

(2) Inlet 1½"MSLC(3)-4 line shares the outboard main steam line isolation valve drain header 3"MS(20)-4. Motor operated valve MS-V-20 isolates this header from 1½"MSLC (3)-4. This valve is only used during reactor startup to warm up the main steam lines to the turbine. During normal plant operation it is closed. Isolation of this valve is, therefore, ensured during a loss of coolant accident and subsequent utilization of the MSIV-LCS system.

Q. 010.16
(9.0)

Identify all safety-related equipment that could be exposed to, or affected by, dust storms. Describe how you propose to assure the proper functioning of this equipment during dust storms. Provide a description of the methods which will be used to prevent the blockage of vital air supplies to safety-related equipment (e.g., clogging of the air filter of the Diesel Generators). In your response to this question, provide a cross-reference to your response to 372.8.

Response:

1. Essentially all safety related equipment that could be affected by severe dust storms are contained within plant areas served by the HVAC systems for the reactor building, control room/cable spreading room/critical switchgear areas, standby service water pumphouses and diesel generator building. The only safety related system exposed directly to severe dust conditions are the service water spray ponds.

   a) The normal air intakes for the reactor building and control room/cable spreading room/critical switchgear are located 130 feet and 85 feet above ground level, respectively. At these intake locations the dust loadings will be 10 to 15 percent of ground level dust loads (See Figure 2.3-5 and the response to Question 372.8 for representative dust loads). All intake air is processed through either automatic roll type filters or replaceable filter elements in the air handling units before entering the air distribution systems for the reactor building, control room, cable spreading room, and critical switchgear areas. An air washer is also included in the reactor building air handling unit. Pressure differential across the filter units is annunciated when filter replacement is required.

      With the intake locations and filtration, discussed above, the amount of dust entering the reactor building, control room, cable spreading room and critical switchgear areas will not degrade the operating capability of safety related equipment in these areas.

   b) The standby service water pumphouses have unfiltered outside air intakes and some dust may be expected to enter the pumphouses during severe dust conditions. The amount of dust, however, should be limited since the pumphouse HVAC systems will be shut down during normal plant operations with the intakes and exhaust openings restricted by dampers.

The dust loading of the air drawn into the pumphouses, when the HVAC systems are operating, should be less than maximum ground level dust loads since the air intakes are located above the service water spray ponds and feed into a plenum before entering the intake fan. Any dust which enters will settle out within the pumphouses without blocking vital air passages.

Entry of dust into the service water pumphouse will not affect the operation of safety related equipment. Any equipment that could be affected by dust is either provided as sealed units, located in dust proof cabinets or protected by dust proof coatings.

c) The diesel generator building outside air intake is located at grade level with air filters located within the building at 15 feet above grade.

The common air filter bank processes all ventilation air into the diesel generator building. During a worst case dust storm, as defined in 2.3.1.2.1.5.2, the maximum estimated dust load will be 8.9 mg/m^3 for an 18 hour duration. After particle impaction and re-entrainment (due to intake louvers) is accounted for, the calculated dust load to the filters is 6.44 mg/m^3. Without taking any credit for particle settling in reduced velocity area before filter bank the filter will be subjected to a maximum of 0.231 $\#/F^2$ of dust.

The filter bank consists of two (2) filters (prefilter and final filter) in series with a common pressure switch which will alarm when filters need changing. The prefilter normal maximum resistance is 0.50" W.G. (equal to 0.047 $\#/F^2$). The final filter normal maximum resistance is 1.00" W.G. (equal to 0.142 $\#/F^2$). During severe dust storm conditions the prefilter can be loaded to 0.129 $\#/F^2$ or 1.00" W.G.

During the postulated dust storm an initial filter alarm would require a complete filter change followed by a maximum of two (2) prefilter changes as filter alarms are annunciated.

The 18 hours loading on the diesel air filters is calculated to be 2102.4 grams. The capacity of the air filters is 5000 grams or 2.378 times the severe dust storm loading.

This response is an elaboration of the response given in part (d) to Question 40.26.

d) The ultimate heat sink transient analysis was performed assuming 6" of sedimentation at the bottom of the spray ponds (see 9.2.5). In addition, no credit is taken in the analysis for the volume of water within the sand traps which prevent sedimentation from being swept into the pump pits.

Q. 010.17
(9.1.2)

The design of your spent fuel rack includes a neutron absorbing material encapsulated in stainless steel. However, recent experience at some spent fuel pools has shown that the stainless steel cladding may bow out due to the internal pressure of gases generated by the irradiation of the neutron absorbing material in the spent fuel pool. This bowing of the steel cladding has caused the spent fuel assemblies to become lodged in the spent fuel racks. Accordingly, describe the method (e.g., venting the stainless steel plates to release any evolved gases) you propose to prevent this from occurring in the WNP-2 spent fuel pool.

Response:

Bowing of the steel cladding is not expected to occur since the neutron absorber plates utilized in the WNP-2 racks have been shown through testing not to offgas when irradiated by a gamma source. These plates manufactured by Electroschmelzwerk - Kempten (ESK) differ substantially in composition and manufacturing process from the type of plates which underwent decomposition at Connecticut Yankee. Because of the nonoffgassing characteristic of these plates, venting of the racks is not planned. For additional information on the offgassing tests, refer to the response to question 010.18 (9.1.2).

Q 010.18
(9.1.2)

In Section 9.1.2 of the FSAR, you list the test results involving radiation, thermal, seismic and borated water testing of the boron carbide plates. Describe the procedures used for these tests. Alternatively, provide a cross-reference to any of these test procedures which have previously been accepted by the NRC staff on another application.

Response:

When the FSAR was originally written, the manufacturer of the boron carbide plates had not been identified. Accordingly, data from previously licensed plates was used based on a program description and results of the qualification tests conducted on boron carbide neutron absorber plates submitted to the NRC under the Connecticut Yankee docket 50-213, letter D.C. Switzer to R.A. Purple dated April 15, 1976. Subsequently ESK was selected as the manufacturer of the plates. With the exception of the full scale seismic test, essentially all described tests have been performed by ESK for the plates of their manufacture. Because of the similarity in physical characteristics with the plates previously tested and because Modules of Rupture tests show plates will withstand two times calculated seismic stresses, repetition of shaker table testing was not deemed necessary. Test results for the ESK plates were submitted to the NRC under the Kewaunee docket 50-305 in a letter E.W. James to V. Stello dated September 5, 1978.

As a result of our decision to use ESK plates, section 9.1.2 is being revised.*

*See attached draft

Q 10.18

d. Shielding for the spent fuel storage arrangement is sufficient to protect plant personnel from exposure to radiation in excess of 10 CFR Part 20 limits. Since provisions for portable shielding are not provided in the drywell, administrative control is used during refueling operations to avoid overexposure of personnel as the result of a postulated fuel drop accident such as a drop occurring on the reactor seal plate.

9.1.2.1.2 Power Generation Design Bases

a. Spent fuel storage space in the fuel storage pool is for 2658 fuel assemblies.

b. Spent fuel storage racks are designed and arranged so that fuel assemblies can be handled efficiently during refueling operations.

9.1.2.2 Facilities Description

9.1.2.2.1 Spent Fuel Storage Racks

Spent fuel storage racks provide a place in the fuel pool for storing the spent fuel discharged from the reactor vessel. They are top entry racks, designed to maintain the spent fuel in a space geometry that precludes the possibility of criticality under both normal and abnormal conditions. This is accomplished with the aid of neutron absorbing plates. The location of the spent fuel pool within the plant is shown in Figure 1.2-6.

The spent fuel storage rack design, shown in Figure 9.1-2, consists of fuel storage cells which are square stainless steel tubes with neutron absorbing B_4C plates between them. A stainless steel plate grid at the top and the bottom of the tubes, to which the tubes are welded, form the tubes into racks and maintain center-to-center spacing between the tubes at 6.5 inches. The racks are welded together into modules which are held firmly in place by seismic restraints attached between the rack modules and the pool wall. The storage racks are made of stainless steel. The square tube storage cells are 1/8 inch thick.

The neutron ~~absorber~~ plates are 0.21 inches ~~thick and are com-~~ posed of B_4C powder bonded ~~together to form~~ a plate with uniform properties. ~~The plate consists of 50% by volume B_4C~~

Delete

//

G10.18

Delete

~~with the remainder being binder and voids. The plate has been shown by tests to be chemically inert in water and thermally stable over the range of pool water temperatures that can occur. The plates are seal welded in the cavity between tubes to prevent water intrusion, with rack and plate dimensions specified to preclude the plates slipping past each other. There are no load bearing requirements for the plates. Plate integrity and mechanical properties have been verified by a comprehensive test program which included seismic testing at frequencies from 7 to 33 Hz, thermal cycling from room temperature through 350°F, soaking for 16 days in 200°F solutions of boric acid and distilled water, and gamma irradiation of approximately 2 x 10^{11} rads. The tests showed no swelling or weight loss, no cracking or dimensional changes and verified the mechanical properties assumed in the design.~~

Add Insert "A" (attached)

Different rack sizes are used (12 x 16, 12 x 13, 8 x 13, 7 x 18 and 11 x 16 arrays) to take full advantage of the fuel storage space in the pool (see Figure 9.1-3). The upper rack structures are welded to an elevated base plate which, in turn, is supported by a system of welded beams and stiffeners. The base serves to support the weight of the fuel assemblies and to distribute the load on the pool floor. The base plate contains an opening at each fuel assembly storage location which accommodates the fuel assembly lower nozzle. Natural circulation of pool water flows upward through the lower nozzle and the fuel assembly to remove decay heat. The storage cells are designed to provide lateral support for the stored assemblies.

The seismic restraints are stainless steel turnbuckles located between the pool walls and the racks around the periphery of the pool (Figure 9.1-3). They are located at both the top and bottom of the rack and, once adjusted will transmit the seismic forces of the OBE and the SSE between the racks and the walls and remain functional. The turnbuckles are connected at the wall to stainless steel bands which are embedded in the concrete wall and seal welded to the pool liner.

9.1.2.2.2 Spent Fuel Storage Pool

The spent fuel storage pool is designed to withstand earthquake loadings as a Seismic Category I structure. It is a reinforced concrete structure completely lined with stainless steel, which provides a leakproof membrane that is resistant to abrasion and damage during normal and refueling operations. The stainless steel liner plates are seamwelded

Insert "A" *Q10.18*

The neutron absorber plates have nominal dimensions of 19 inches long, 5.88 inches wide, and 0.2 inches thick. They are composed of B_4C granular material bonded ~~and sintered~~ together to form a plate of uniform properties. They have a nominal B^{10} loading ~~is~~ *of* 0.0959 grams per square centimeter of plate and a plate density of 0.05 lbs/in^3. The plate has been shown by tests to ~~be chemically inert~~ *have negligible corrosion* in water and thermally stable over the range of pool water temperatures that can occur. The plates are seal welded in a stainless steel cavity to prevent water intrusion.

There are no load bearing requirements for the plates. Based on the results of the Modulus of Rupture tests, the plates will withstand approximately two times the calculated stresses caused by a postulated seismic event. Plate integrity and mechanical properties have been verified by comprehensive tests. These tests included Modulus of Rupture and Modulus of Elasticity tests. The Modulus of Rupture testing was performed using a ~~four~~ *three* point support method and was done on specimens at temperatures varying from ambient to 300°F, specimens soaked in water, and irradiated specimens. The Modulus of Elasticity was performed using a resonance procedure and was done at varying temperatures and after the plate had been immersed in water. The tests showed no swelling, cracking or dimensional changes and provided verification of the plate mechanical properties required for the rack design.

In addition to the mechanical tests, extensive irradiation induced offgassing tests have been performed using gamma sources. These test results clearly indicate that the amount of offgassing is negligible and *will* not cause . rack distortion.

Q 010.19
RSP
(9.1.2)

In Section 9.1.2.3.3 of the FSAR, you state that the interlocks which prevent the 125 ton crane in the reactor building from traversing the spent fuel pool, are occasionally by-passed. This by-passing is unacceptable. Accordingly, we require you to modify your procedures so that the interlocks on the reactor building crane prevent the crane from traversing over the spent fuel pool whenever there is spent fuel in the pool.

Response:

Regulatory Guide 1.13, c.3 allows for movement of loads necessary for fuel handling over the spent duel. Occasionally it is necessary to operate the reactor building crane over the spent fuel pool in conjunction with maintenance of fuel storage and fuel handling facilities, or other activities associated with fuel handling and storage. Therefore it is necessary to retain the ability to bypass the interlocks and use administrative control procedures under those conditions. Movement of objects in excess of the rack design drop load (one fuel assembly at four feet above the top of the fuel rack) will be prohibited. The electrical interlocks are bypassed only by actuation of a cab-mounted key-lock switch.

See revised section 9.1.2.3.3 Appendix c.3 page 11, and revised FSAR Figure 9.1-17 which shows the interlock-controlled restricted area for crane travel over the spent fuel pool.*

* See attached draft pages.

9.1.2.3.3 Spent Fuel and Cask Handling

~~The path of loads handled in the area of the spent fuel pool will be administratively controlled.~~ The 125 ton reactor building crane traverses the full length of the refueling of floor level ~~and~~ the reactor building. The design of the refueling floor provides aisles on both sides of the fuel pool for moving components past (and not over) the fuel storage pool. ~~Interlocks on the reactor building crane limit crane load movement to along these aisles. For the rare occasions when the interlocks are bypassed, operational procedures limit crane movement.~~ Interlocks on the reactor building crane prevent travel over the spent fuel pool. The interlock-controlled restricted area for crane travel is shown in Figure 9.1-17. The interlocks are bypassed only when it is necessary to operate the crane in the fuel pool area in conjunction with activities associated with fuel handling and storage. During these rare occasions when the interlocks are bypassed, administrative controls are used to prevent the crane from carrying loads that are not necessary for fuel handling or storage, and which are in excess of the rack design drop load (one fuel assembly at four feet above the top of the fuel rack). See 9.1.2.3.2.e.

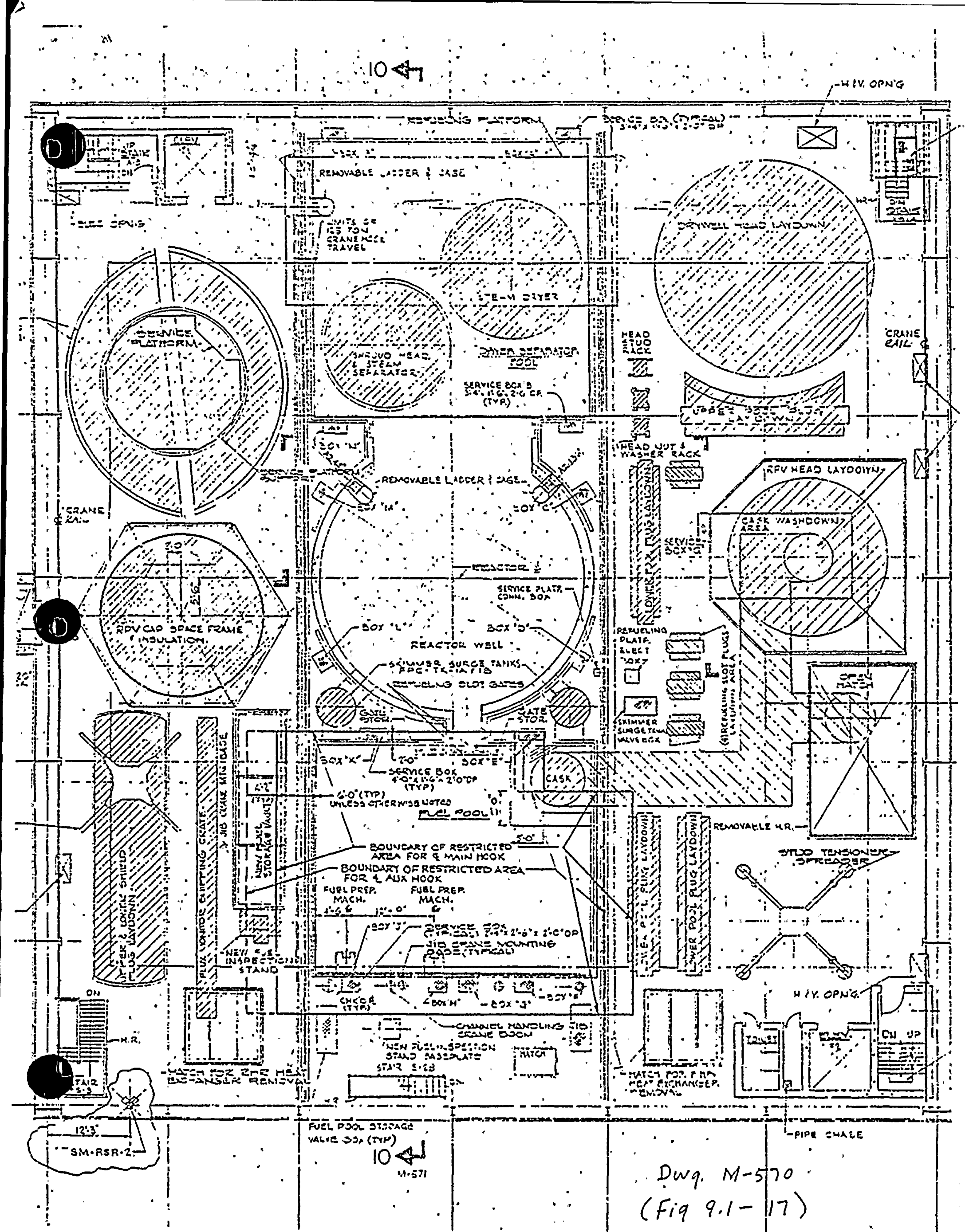
Transfer of fuel assemblies between the reactor well and the spent fuel pool is performed with the refueling platform (see 9.1.4.2.10.2). The fuel grapple or the auxiliary fuel hoist may be used, depending on the transfer operation.

The grapple and hoist are provided with load sensing and limiting devices designed to the following limits:

	Fuel Grapple (lbs)	Auxiliary Fuel Hoist (lbs)
Load limiting switch	1200	1000
Load sensing switch	485	485
Stall torque or hoist system	3000	3000

The load limiting features of the refueling platform grapple and auxiliary fuel hoist will prevent damage to the fuel racks if a fuel assembly accidently engages a rack while being lifted. These load limits provide a redundant safety feature since the fuel handling grapple is not lowered below the upper fuel rack and is designed to interface only with the fuel bail. Thus, the possibility of inadvertent direct lifting of the racks with the grapple is precluded.

Guard rails around the spent fuel pool prevent the falling of fuel handling area machinery into the pool. Other objects that could conceivably fall into the pool will not transfer energy amounts exceeding the specified limits of the fuel racks.



Dwg. M-570
(Fig 9.1-17)

Regulatory Guide 1.13, Rev. 1, December 1975

Spent Fuel Storage Facility Design Basis

Compliance or Alternate Approach Statement:

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

General Compliance or Alternate Approach Assessment:

A controlled leakage building is provided enclosing the fuel pool. The building is not designed to withstand extremely high winds, but leakage is suitably controlled during refueling operations. The building is equipped with a ventilation and filtration system which is designed to limit the potential consequences of the release of radioactivity specified in Regulatory Guide 1.25 to those guidelines set forth in 10CFR100.

The movement paths of heavy objects such as the reactor pressure vessel head, containment vessel head and the spent fuel cask are designed not to pass over the spent fuel pool. Furthermore, the reactor building crane and its auxiliary hoist are prevented by means of interlocks from passing over any of the spent fuel pool except the spent fuel cask area. Bypassing of the interlocks is permitted only during fuel handling and storage operations and is administratively controlled.

Although all of the spent fuel cooling and cleanup system equipment is not Seismic Category I, the source of emergency makeup water is from the Seismic Category I standby service water system.

The fuel pool is designed so that no pipe break will drain water from the fuel pool.

Specific Evaluation Reference:

Refer to 9.1.

Q 010.20
RSP
(9.1.2)

In Section 9.1.2 of the FSAR, you state that a portion of the fuel handling building above the refueling floor is constructed of sheet metal. Accordingly, we require you to demonstrate that the spent fuel pool is housed in a Seismic Category I structure which can withstand the impact of tornado missiles.

Response:

Table 3.2-1 states that the Reactor Building is designed to Seismic Category I requirements. Section 3.5.1.4 states that Seismic Category I structures are designed to include the effects of missiles generated by the design basis tornado. Section 3.8.4 provides details of the design features of the Reactor Building and spent fuel storage pool. Tables 3.8-15 and 3.8-16 provide the load combinations and load factors used in design of Seismic Category I structures. Section 3.5.1.4.1.a discusses the tornado missile-resistant design features of the Reactor Building. Section 3.3.2 discusses the design features of the Reactor Building for tornado wind loading. As stated in 3.5.1.4.1.a and 9.1.2.3.5, which reference GE Topical Report APED-5696, the design basis tornado missile for the refueling floor has been evaluated and found to not have sufficient energy to damage the spent fuel or the equipment and structures in the pool.

Q. 010.21
(9.1.3)

Provide a cooling system and a source of makeup water for the spent fuel pool which are both designed to seismic Category I criteria in accordance with the staff positions contained in Regulatory Guide 1.13, Revision 1, "Spent Fuel Storage Facility Design Basis," December 1975.

Response:

WNP-2 has a seismic Category I source of makeup water for the spent fuel pool from the seismic Category I standby service water system. This is shown on Figure 9.1-4 and stated in section 9.1.3.3. Cooling under emergency conditions for the fuel pool is supplied by evaporation of pool water. Reg. Guide 1.13, Rev. 1, makes no specific statements about requiring a seismic category I spent fuel pool cooling system. As a result, WNP-2 meets the applicable criteria of the Reg. Guide and the intent of the question. However, further evaluation of the design in this area is ongoing due to the interaction of fuel pool cooling and post-LOCA secondary containment pressure-temperature response. (See the response to question 312.18)

Q 010.22
(9.2.1)

Identify which valves are used to isolate that portion of the plant service water system which is not designed to Seismic Category I criteria from that portion which is designed to these criteria. Provide a failure modes and effects analysis for the plant service water system, assuming a seismic event has occurred.

Response:

The plant service water system (TSW) is not required for safe shutdown and accordingly is not designed to Seismic Category I requirements. A failure modes and effects analysis is not considered necessary. The portions of the TSW system piping in the Reactor Building have been designed to Seismic Category I requirements so that they will not fall and damage safety related equipment.

The standby service water system (SW) is used for safe shutdown and is designed to Seismic Category I criteria. The SW is discussed in 9.2.5. The plant service water system (TSW) and the standby service water system (SW) are independent systems and are not connected, therefore there are no valves which are used to isolate these systems from each other.

Q. 10.23
RSP
(9.2.5)

Provide the results of your analysis of the capability of the ultimate heat sink to absorb heat over a thirty-day period following a postulated design basis accident. Indicate the total heat absorbed in the ultimate heat sink, including the sensible heat, the station auxiliary system heat, and the decay heat released by the reactor core. In particular, provide the following information in both tabular and graphical formats:

a. The total integrated decay heat.

b. The heat rejection rate and the integrated heat rejected by the station auxiliary systems, including all operating pumps, ventilation equipment, diesels and other heat sources.

c. The heat rejection rate and integrated heat rejected due to sensible heat removed from the containment and the primary system.

d. The total integrated heat rejected; i.e., the sum of the Items (a), (b) and (c).

Additionally, provide the following information:

e. The maximum allowable temperature of the inlet water taking into account the rate at which heat must be removed, the cooling water flow rate, and the capabilities of the respective heat exchangers.

f. The required and available net positive suction head (NPSH) at the suction lines of the service water pumps at the minimum water level of the ultimate heat sink.

This analysis should demonstrate the capability of the ultimate heat sink to provide: (1) an adequate water inventory; and (2) sufficient heat dissipation which will limit the essential cooling water operating temperatures within the design ranges of system components. In this regard, we require you to use the methods contained in Branch Technical Position ASB 9-2 "Residual Decay Energy for Light Water Reactors for Long Term Cooling," when evaluating the residual decay energy release rate from the reactor core due to fission product decay and heavy element decay. Assume an initial cooling water temperature based on the most adverse conditions possible during normal operations. The meteorological conditions should be established following the

guidance contained in Position C.1 of Regulatory Guide 1.27, Revision 1, "Ultimate Heat Sink for Nuclear Power Plants," March 1974.

Response:

See revised Section 9.2.5* which provides the revised results of the analysis including the information requested above. This revision also addresses the concerns of Question 371.6.

*See attached draft pages

The worst storm of these was storm No. 3. While it was also shown in this study that once a given dust storm terminated, there existed a 5% probability that another one would occur within 10 hours and a 50% probability that another one would occur within 30 days, none of the above six worst case dust storms had occurred within 30 days of each other. Most had occurred in different years during the 1953-1970 study period.

The dust loading for storm No. 3 is conservative in terms of its being considered as the worst case storm for use in plant design evaluations. As a result of the shorter storm durations of the measured August 11, 1955, January 11, 1972, and April 1972 dust storms, their time integrated dust loadings at 5-6 feet above the ground are not worse than that computed for storm No. 3 [33].

2.3.1.2.2 Design Snow Load

The American National Standards Institute (ANSI) in "Building Code Requirements for Minimum Design Loads in Buildings and other Structures" [19] provides weights of 100-year return period ground level snow packs for the site region. The ANSI value of 20 pounds per square foot was used as the design snow load for all WNP-2 structures.* Assuming a snow density (specific gravity) of 0.1 or 6.24 lbs/ft^3, this design value corresponds to a snow depth of 3.2 feet. The above snow load is conservative for the site as snow depth seldom exceeds six inches, and the greatest depth of 21 inches was recorded in February 1916.[4] The weight of the 48-hour probable maximum winter precipitation can be determined from the data presented in Table 2.3-3. Since the greatest snowfall in 24 hours was 7.1 inches (January 1954) and a record depth of approximately 12 inches lasted four days (December 1964) these depths would correspond to snow loads of 3.7 and 6.24 lbs/ft^2 respectively.

2.3.1.2.3 Meteorological Data Used for Evaluation of Ultimate Heat Sink~~**~~

The meteorological data presented in Figures 2.3-7 to 2.3-9a and Tables 2.3-1, 2.3-5, and 2.3-7a-7h was used to evaluate the performance of the WNP-2 spray ponds in 9.2.5 with

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* Ice loading is included in this WNP-2 estimate.

~~**The meteorological data used for evaluation of the UHS presented here is currently undergoing review for compliance with R.G. 1.27, Rev. 2.~~

charge header of the pumps stops the jockey pump and starts one of the two main pumps on an increase in demand of system flow. Upon a further increase in flow demand, above 140 gpm, the flow meter automatically starts the second pump. When the flow demand decreases below 140 gpm, the second main pump stops. If flow demand continues to decrease to below 50 gpm, the jockey pump starts and the main pump stops.

During the starting sequence if one pump fails to start, the sequence automatically continues to the next pump and a local alarm and light indicate pump failure. Upon indication of low potable water storage tank level, all pumps stop.

The reactor building potable water booster pumps are automatically cycled on and off by a pressure switch in the pressurizing tanks on the pump discharge in order to maintain header pressure between 20 and 50 psig.

All electric water heaters are thermostatically controlled to maintain the tank at the desired setpoint. The hot water circulating pumps in the service building and radwaste building are cycled by a thermostat with sensor in the hot water recirculation line set to maintain the loop at a minimum setpoint.

9.2.5 ULTIMATE HEAT SINK

9.2.5.1 Design Bases

a. The ultimate heat sink, a spray pond system, supplies cooling water to remove heat from all nuclear plant equipment which is essential for a safe and orderly shutdown of the reactor and to maintain it in a safe condition.

b. The ultimate heat sink is capable of accomplishing its safety function for a normal cooldown or an emergency cooldown following a loss of coolant accident without the availability of off-site power. The sink provides this cooling capability for a period of 30 days without outside makeup. Provisions are made for replenishment of the sink to allow continued cooling capability beyond the initial 30-day period. The sink will accomplish its safety function despite the occurrance of the most severe site related natural events including earthquake, tornado, flood or drought.

The following worst month meteorological data were used in 9.2.5 to establish the second through thirtieth day worst pond thermal performance and worst 30 day drift loss and evaporation(21):

1) July 9 - August 8, 1961 at HMS, presented in Table 2.3-7g (minimum heat transfer)

2) July 2 - August 1, 1960 at HMS, presented in Table 2.3-7h (maximum evaporation and drift loss)

Diurnal variations in dry bulb and wet bulb temperatures for both 30 day periods assumed that the hourly temperature variation approximated a sine wave of one cycle in 24 hours(21). The average wind speeds during both 30-day periods was approximately 5.5 mph. The highest daily average wind speed for the 30-day mass loss period is 10.3 mph. ~~Both of these wind speeds were considered in the mass loss calculation presented in 9.2.5.~~

¶ For conservatism in the thermal analysis, the worst day data for thermal performance was ~~repeated~~ assumed to repeat in the analysis until pond temperature peaked (three days repetition). For conservatism in the mass loss analysis, five times the calculated drift loss for the highest daily average wind speed (10.3 mph) was assumed to occur for the entire 30 day period. See 9.2.5 for details.

2.3.2 LOCAL METEOROLOGY

2.3.2.1 Data Comparisons

The local meteorology at the WNP-2 site can be described from FSAR meteorological data procured during the period April 1, 1974 to March 31, 1976 from the permanent onsite 7 foot and 245 foot meteorological towers. Data collected from the 245 foot WNP-2 tower have been used for the short term (accident) and long term (routine) diffusion estimates. Onsite meteorological data were also obtained from a temporary 23 foot tower which commenced operation in April 1972 for the purpose of determining optimum cooling tower geometric orientation for performance during high wet bulb periods. The 23 foot meteorological tower data were also used with other regional data to establish the potential impact of proposed mechanical draft cooling tower atmospheric releases in the vicinity of WNP-2(22). The permanent tower data have been compared where appropriate and possible, with simultaneously recorded and historical data obtained from the Hanford Meteorological Station (HMS) for the purpose of documenting the representativeness of the two years of onsite meteorological measurements. For the months of April through August 1974, comparisons have also

c. The ultimate heat sink is designed to satisfy the regulatory requirements of Regulatory Guide 1.27 (Rev. 1). See Appendix C and Section 2.3.1.2.3.

9.2.5.2 System Description

During all normal operating conditions, including startups and normal shutdown, waste heat from the reactor auxiliaries is transferred to the circulating water system. Heat from this system is in turn rejected to the atmosphere by the normal plant cooling tower system.

Following any event that would prevent the use of the plant cooling towers, the heat rejection duties are transferred to the spray ponds. The ultimate heat sink consists of two concrete ponds with redundant pumping and spray facilities. The pond and pumphouse arrangements are shown on Figure 9.2-11. The ponds and pumphouses are designed to Seismic Category I requirements. Standby service water (SSW) loop A draws water from pond A, cools the Division I equipment required for safe shutdown, and discharges ~~into~~ through the spray ring in pond B for heat dissipation. Similarly, SSW loop B draws water from pond B, cools Division II equipment, and discharges ~~into~~ through the spray ring in pond A. The HPCS SSW system draws water from pond A, cools division III and discharges without spray into pond A. A syphon between the ponds allows for water flow from one pond to the other.

The spray system illustrated in Figure 9.2-11 consists of two annuli of spray trees -- one for each of the concrete ponds. Each annulus is 140.0 feet in diameter and contains 32 spray trees equally spaced (13.75 feet between vertical centerlines) on the circumference. The vertical trees are serviced by the annulus water pipe, 20 inches in diameter, mounted above the water level. The annulus pipe is fed by the main header from each respective pumphouse. Each spray tree consists of a vertical riser pipe or trunk 8 inches in diameter and 7 horizontal limbs of 1-1/2 inch pipe. The limbs are attached to the riser at 2'8" intervals of heights and are rotated at 90° subsequent angles from each other so that the arms resemble a counter-clockwise helix with increasing height. The arms radial to the annulus are 4'6-7/16" long. The lowermost arm is a tangent arm. The arms tangent to the annulus pipe are 3'6" long. Spray nozzles are located at the end of each arm and are connected by fittings so that the orientation of every nozzle is radially inward with an angle of 55° upward from horizontal. The nozzles are 1-1/2-CX-27-55 Whirljet nozzles supplied by Spraying Systems Company. Since each

tree nozzle is located at a different elevation, each nozzle pressure is different. The uppermost nozzle water pressure is 17.0 psig, and the total water flow from a tree is approximately 300 gpm.

The HPCS SSW flow, 1192 gpm, is treated as a straight heat dump in the thermal analyses.

The combined water volume of the spray ponds is adequate to provide cooling water for 30 days without makeup. Although the pond is not used for cooling during normal operation, some small losses are to be expected due to normal evaporation from the surface and occasional blowdown needed to maintain water chemistry. A gravity makeup line is provided from the circulating water pumphouse to the spray ponds to automatically maintain the pond water at the required level. The ponds can also be supplied directly from the plant makeup water pumps (see 10.4.5). Design parameters for the spray pond are given in Tables 9.2-1 and 9.2-2.

A standby service water pump is located in each spray pond pumphouse along with its associated equipment so that an accident, such as a fire or pipe break associated with one pump would not affect the operation of the redundant pump.

~~The bottom of the pump sump is depressed below the pond bottom. This ensures that there is still sufficient submergence for the pumps at the lowest possible water level in the pond.~~ A sand trap, stop log, and screen, precedes the pump sump to prevent heavy debris from entering the pump sump area. A skimmer wall and fixed screen prevent floating debris from entering the pumps.

A spray ring bypass is provided so that the water temperature may be controlled during cold weather operation. When the pond temperature drops below approximately 60°F, the spray ring may be bypassed by opening the dump valve returning water directly to the pond.

To prevent adverse operation during freezing weather, all SSW piping and components are either below the frost line, within the heated pump houses, heat traced, or, in the case of the spray rings, kept drained by the return header dump when not in operation.

The pump suction for ~~the~~ standby service water loops A and B is at elevation 409'3". The pump suction for the HPCS standby service water is at 417'0". This ensures that the required net positive suction head (NPSH) is ~~adequate~~ maintained on the standby service water pumps (see Figures 9.2-12 and 9.2-13) even if the water level is at the bottom of the spray pond, 420'. The required NPSH for loops A and B service water pumps is 3 feet, and for the ~~9.2-12~~ HPCS service water pump it is 2 feet.

9.2.5.3 Safety Evaluation

An oriented spray cooling system (OSCS) is utilized for cooling the water inventory of the ultimate heat sink. OSCS has been developed as a result of intensive analytical studies and experimental verification over a period of more than six years. Details of the OSCS experimental and analytical developmental efforts are described in Topical Report, Oriented Spray Cooling System(OSCS) for Ultimate Heat Sink Application (UHS), I-R 100 which has been submitted for Nuclear Regulatory Commission staff review. The meteorological data for the UHS is discussed in 2.3.1.2.3.

The thermal performance model is based on the correlation of the Canadys test data described in Section 3.1 of Topical Report, I-R 100. The resulting KAV/L for this application is 2.66. This includes a 10% derate of the KAV/L to cover conservatively the data scatter experienced at Canadys. Since the KAV/L represents the performance of the specified geometry and nozzle pressure, the KAV/L combined with the meteorological data are sufficient to determine the system cooling performance.

The system model for both the thermal performance ~~and~~ analysis and the mass loss analysis was based on the following assumptions:

a. The pond contains total inventory upon onset of LOCA less 0.5 feet for sedimentation of the pond basin.

b. Water losses result only from drift, evaporation of the sprayed droplets, and evaporation due to heat rejection on the pond surface.

It was found that repeating the worst day conditions for three days resulted in the maximum peak.

c. All the heat transfer is accomplished by evaporation, none of the heat transfer is accomplished by sensible heat transfer.

d. The first three days of the thermal performance analysis ~~is~~ assume the worst single day of record conditions (Table 9.2-4 Page 1 of 3). The ~~second~~ fourth through thirtieth days are the average meteorological conditions of the worst 30 day period of record (Table 9.2-4 Page 2 of 3). For conservatism, the average wind speed during each period is used rather than the diurnal variation.

e. The first through thirtieth day of the mass loss analysis are the average meteorological conditions of the worst 30 day period of record (See 2.3.1.2.3 and Table 9.2-4 Page 3 of 3). ~~except that~~ The analysis assumes a mass loss due to drift of ~~.73%~~ .57% of the spray flow. The spray flow is based on continuous operation of one spray ring.

INSERT (A) HERE.

f. Off-site power is lost and Division ~~loop~~ 2 diesel fails to start, resulting in a loss of ~~one of~~ the ~~two~~ POND A spray header~~s~~. (Division I heat loads are slightly higher)

g. ~~The heat rejection load to the ultimate heat~~ sink is comprised of decay heat from fission products, decay heat from heavy elements, sensible heat from the reactor coolant system, and heat removed from the emergency equipment in the two operating divisions. These heat loads are tabulated in Table 9.2-8.

The average wind speed during the thirty day mass loss period is 5.5 mph. The highest daily average is 10.3 mph. These wind speeds result in drift losses of approximately .11% and .23% of the spray flow, respectively. In assuming a drift loss of .73% of the spray flow, the mass loss analysis demonstrates that the spray ponds contain sufficient water inventory to meet drift losses from wind speeds significantly higher than expected.

Diurnal psychrometric data, averaged over a 14 year period (1957-1970) for each month and presented in Battelle Northwest Laboratories "Climatography of the Hanford Area", ~~BNWL-1605, June 1972, was used~~ in determining the highest initial pond temperature. The data for the month of July yielded the highest initial pond temperature, 77.4°F.

Using the assumptions stated earlier, two analyses were run. The first assumed a clean RHR heat exchanger and the second assumed a fouled heat exchanger. The pond temperature transient for the first analysis is shown in Figure 9.2-7. The pond temperature peaks at 87.0°F on the thirtieth hour. The mass inventory is shown in Figure 9.2-8, and the mass losses are tabulated in Table 9.2-3. The diurnal oscillations in pond temperature are due to the changes in solar loading. The second analysis yielded a peak pond temperature of 1/4°F lower and at the same hour after the accident. The only significant difference between the two analyses is ~~the temperature transient predicted for the suppression pool.~~

~~A second~~ An analysis was conducted ~~to verify~~ which verified that ~~the second~~ ~~single~~ failure ~~as a loss~~ of Division 1 or Division 2 power ~~.~~ results in the most severe service water transient.

~~This analysis took into account the heat load from the emergency equipment, including [illegible] of all three divisions, but took credit for cooling from both spray rings. The peak pond temperature was less than 85°F.~~

no ¶

If the failure was postulated in Division 3 (HPCS) instead of Division 1 or 2, the peak pond temperature is ~~below~~ lower. ~~[illegible]~~ The HPCS ~~S~~SW flow is a straight heat dump; therefore, inasmuch as the spray pond is concerned, it raises rather than lowers the temperature transient.

"INSERT A"

g. The major heat loads considered are reactor core decay heat, sensible heat from both the coolant and the reactor, fuel pool decay heat, pump work, and the heat removed from the station auxiliaries. These heat loads are detailed in Table 9.2-8 and Figures 9.2-7b, -7c and -7d. No credit was taken for heat sinks in the primary containment other than the suppression pool volume.

The actual average wind speed during the selected thirty day period for the mass loss analysis was 5.5 mph. However, for conservatism, the drift loss assumed in the analysis was based on five times the calculated drift value at the highest daily average wind speed of 10.2 mph. The mass loss analysis thus demonstrates that the spray ponds contain sufficient water inventory to meet drift losses significantly higher than expected.

The analyses assume an initial temperature of 77°F. This is approximately the highest monthly average temperature expected if the sprays are not operated. To maintain the pond temperature below this limit, the spray headers will be operated and/or river water make-up to the cooling towers will be diverted through the spray ponds. Analyses have been performed which demonstrate that the above operations can maintain the spray pond below 77°F.

Replace with "Insert B"

The resulting peak SSW temperature, 87°F, predicted by the "worst case" analysis is considerably below the 95°F SSW temperature assumed in the analyses performed in 6.2.1 for containment heat removal. The peak suppression pool temperatures stated in 6.2.1 and 6.2.2 are therefore conservative. The SSW peak temperature, however, exceeds the design bases SSW temperature used for HVAC equipment by 2°F for a period of 12 hours. This increase has been evaluated. It would result in a peak temperature for those areas served by emergency HVAC equipment of, at most, 2°F higher than was originally calculated. However, since the temperature rise is small and exists for only a short period of time, it was assessed as not being deleterious to equipment operation.

Drift losses following loss of makeup to the ponds are controlled during two spray ring operation by bypassing the spray header on one pond whenever spray pond temperatures drop below approximately 80°F. Continuous, simultaneous operation of both spray rings is not required after a LOCA. Since the two SSW loops are redundant to each other, ~~it is expected that~~ the operators will be able to secure any redundant safe shutdown equipment when they determine that the peak temperatures have been past. In addition, the difference between assumed and calculated drift losses for continuous operation of one spray ring, is more than adequate to account for drift losses from the operation of the second spray ring for several days after the accident.

Table 9.2-7 lists the available sources of makeup water to provide continued cooling beyond the initial 30-day period. This table assumes that off-site power is restored within the 30 days. No credit is taken for the water stored in the cooling tower basins. However, it is expected that this water will not be instantaneously lost and will flow to the pond for the same period of time. Table 9.2-7 also summarizes the effects of natural phenomena and of a LOCA on the water supplies to the spray pond.

The possibility of a tornado passing over the spray pond and removing a significant amount of water is considered a credible event. For this reason, the makeup water pumphouse is designed to be tornado proof, with all piping and electrical power supply between the plant and the pumphouse

"INSERT B"

The resulting peak spray pond temperature, 88.6°F, predicted by the "worst case" analysis is considerably below the 95°F service water temperature assumed in the analysis performed in 6.2.1 for containment heat removal, adding further conservatism to the containment temperature and pressure transients therein presented. The service water temperature, however, exceeds the design basis temperature, 85°F, at the emergency reactor building and control room HVAC equipment for a short period of time as shown in Figure 9.2-7a. This results in a peak temperature for some of the electrical equipment rooms served by emergency HVAC equipment of, at most, 3°F higher than the nominal lifetime rating for the equipment. This has been assessed as not being deleterious to the equipment operation.

A sensitivity study was performed to determine the effect of the RHR heat exchanger effectiveness on the suppression pool and spray pond temperature transients. The RHR heat exchanger effectiveness varies with the amount of fouling and with the flow rates. RHR heat exchanger flows different from the rated values in Table 6.2-2 are anticipated only if the operator delays or fails to close the RHR heat exchanger shellside bypass valve as discussed in 6.2.2.3. Anticipated variations in flow and fouling were determined to have essentially no affect on the spray pond temperature transient following a design basis LOCA, but were determined to have an impact on the suppression pool temperature transient. The most severe postulated suppression pool temperature transient results from assuming a fully fouled RHR heat exchanger and no operator action to close the shellside bypass valve. This suppression pool transient presented in Figure 9.2-7a, is slightly less severe than the suppression pool transient presented in 6.2.1 which assumed a steady 95°F service water temperature and that the operator closed the RHR heat exchanger bypass valves.

The results of the mass loss analysis assuming an unfouled heat exchanger is shown in Figure 9.2-8 and is tabulated in Table 9.2-3. The mass loss assuming a fouled heat exchanger is less severe, but only by approximately 2,000 gallons.

underground. Since it is not credible to assume an earthquake coincident with a tornado, this system need not be Seismic Category I. Two 12,500 gpm plant makeup water pumps are provided, one powered from each emergency diesel generator. Should pond water be lost due to a tornado, one of these pumps will be started to provide makeup. Valves are provided in the makeup water line to isolate the flow ~~from~~ to the cooling tower and to ensure that it goes to the spray pond.

9.2.5.4 Testing and Inspection Requirements

After completion of the spray pond, an inspection and test program has been established to ensure that the spray system will accomplish its safety function as discussed in 14.2.

All valves and piping in the system have been hydrostatically tested in the shop per ASME Section III, Class 3. After installation the system is hydrostatically tested and visually inspected. During plant operation the system is periodically tested.

Preservice and inservice inspections for the spray system will be in accordance with 6.6.

9.2.5.5 Instrumentation Requirements

The spray pond is equipped with redundant level and temperature sensors which are alarmed and indicated in the main control room as well as locally.

In the event that the spray pond level falls below the minimum level required for 30 days of cooling, an alarm is sounded and makeup automatically is provided directly from the plant makeup water line to the spray pond.

High and low temperature alarms are provided. In the event that the pond water temperature approaches the design limit, the spray system is initiated to lower the temperature. Upon low water temperature signal, return water is dumped directly into the ponds to prevent spray trees and spray headers from icing.

TABLE 9.2-3

TOTAL SPRAY POND WATER LOSSES AND CONTENT 30 DAYS AFTER LOCA EVENT

Drift losses	2,488,867 ~~3,081,653.14~~ gal
Spray evaporation	6,152,171 ~~5,705,675.53~~ gal
Surface evaporation	478,683 ~~623,628.65~~ gal
Total	9,119,721 ~~9,410,957.32~~ gal
Remaining inventory	3,380,279 ~~3,089,042.68~~ gal

TABLE 9.2-4

DIURNAL VARIATION IN METEOROLOGICAL DATA (FOR WORST SINGLE DAY OF RECORD USED TO ANALYZE THE POND THERMAL RESPONSE ~~ON FIRST DAY~~ FOLLOWING LOCA)

during first three days

Hour	Dry Bulb (°F)	Dew Point (°F)	Wet Bulb (°F)	* Wind Speed (mph)	Solar Radiation ($\frac{BTU}{hr}$)
Noon	100.91	59.41	72.98	~~3.89~~ 6.60	290.81
1:00 p.m.	103.09	59.69	73.58	~~5.22~~ 6.60	282.71
2:00	105.20	58.91	73.96	~~5.30~~	261.30
3:00	105.71	56.00	72.80	~~5.90~~	226.27
4:00	104.93	54.11	71.78	~~9.37~~	180.98
5:00	102.48	55.88	71.81	~~12.21~~	127.56
6:00	101.15	56.05	71.50	~~8.35~~	70.89
7:00	98.27	56.13	70.68	~~5.37~~	16.86
8:00	96.21	56.59	70.27	~~4.26~~	0.00
9:00	90.72	60.53	70.57	~~3.14~~	0.00
10:00	91.33	57.68	69.31	~~7.30~~	0.00
11:00	91.49	60.48	70.77	~~11.76~~	0.00
Midnight	90.91	58.03	69.35	~~8.36~~	0.00
1:00 a.m.	85.92	59.17	68.39	~~12.16~~	0.00
2:00	84.24	57.28	66.88	~~9.19~~	0.00
3:00	80.61	56.21	65.14	~~5.08~~	0.00
4:00	80.24	58.48	66.21	~~1.56~~	0.00
5:00	78.27	59.55	66.15	~~6.53~~	16.86
6:00	83.25	62.99	69.65	~~7.10~~	70.89
7:00	86.77	62.91	70.67	~~4.15~~	127.56
8:00	90.64	61.09	70.83	~~3.89~~	180.98
9:00	92.64	62.00	71.90	~~5.12~~	226.27
10:00	95.23	63.36	73.38	~~3.77~~	261.30
11:00	98.32	62.40	73.73	~~5.74~~ 6.60	282.71

Data based upon 10 July 1975.

* wind speed is average wind speed for period

TABLE 9.2-4 (Continued)

DIURNAL VARIATION IN METEOROLOGICAL DATA (FOR DAY ~~1~~ 4 THRU 30
USED TO ANALYZE POND THERMAL RESPONSE FOLLOWING LOCA)

Hour	Dry Bulb (°F)	Dew Point (°F)	Wet Bulb (°F)	Wind Speed (mph) *	Solar Radiation ($\frac{BTU}{hr}$)
Noon	95.40	45.9	65.5	5.50	290.81
1:00 p.m.	96.80	46.1	66.0	5.50	282.71
2:00	97.30	46.1	66.2	5.50	261.30
3:00	96.80	46.2	66.0	5.50	226.27
4:00	95.40	46.2	65.5	5.50	180.98
5:00	93.10	46.0	64.7	5.50	127.56
6:00	90.10	45.6	63.6	5.50	70.89
7:00	86.60	45.6	62.3	5.50	16.86
8:00	82.80	45.6	61.0	5.50	0.00
9:00	79.00	45.2	59.6	5.50	0.00
10:00	75.60	45.6	58.4	5.50	0.00
11:00	72.50	46.0	57.3	5.50	0.00
Midnight	70.20	46.2	56.5	5.50	0.00
1:00 a.m.	68.80	46.0	56.0	5.50	0.00
2:00	68.30	46.3	55.8	5.50	0.00
3:00	68.80	46.1	56.0	5.50	0.00
4:00	70.20	46.2	56.5	5.50	0.00
5:00	72.50	45.8	57.3	5.50	16.86
6:00	75.60	46.0	58.4	5.50	70.89
7:00	79.00	46.6	59.6	5.50	127.56
8:00	82.80	45.8	61.0	5.50	180.98
9:00	86.60	45.6	62.3	5.50	226.27
10:00	90.10	45.8	63.6	5.50	261.30
11:00	93.10	45.8	64.7	5.50	282.71

Data based upon average values for the period 9 July - 8 August 1961.

* wind speed is average wind speed for the period

TABLE 9.2-4 (Continued)

DIURNAL VARIATION IN METEOROLOGICAL DATA (FOR DAY 1 THRU 30 USED TO ANALYZE MASS LOSS FOLLOWING LOCA)

Hour	Dry Bulb (°F)	Dew Point (°F)	Wet Bulb (°F)	Wind Speed (mph)*	Solar Radiation ($\frac{BTU}{hr}$)
Noon	96.40	42.50	64.70	10.30 ~~5.50~~	290.81
1:00 p.m.	98.00	43.50	65.40	10.30 ~~5.50~~	282.71
2:00	98.50	43.50	65.60	5.50	261.30
3:00	98.00	43.50	65.40	5.50	226.27
4:00	96.40	42.50	64.70	5.50	180.98
5:00	93.90	42.00	63.70	5.50	127.56
6:00	90.70	42.00	62.30	5.50	70.89
7:00	86.90	40.50	60.70	5.50	16.86
8:00	82.90	40.00	59.00	5.50	0.00
9:00	78.90	40.00	57.30	5.50	0.00
10:00	75.10	39.00	55.70	5.50	0.00
11:00	71.90	39.00	54.30	5.50	0.00
Midnight	69.40	39.00	53.30	5.50	0.00
1:00 p.m.	67.80	39.00	52.60	5.50	0.00
2:00	67.30	39.00	52.40	5.50	0.00
3:00	67.80	39.00	52.60	5.50	0.00
4:00	69.40	39.00	53.30	5.50	0.00
5:00	71.90	39.50	54.30	5.50	16.86
6:00	75.10	39.00	55.70	5.50	70.89
7:00	78.90	40.00	57.30	5.50	127.56
8:00	82.90	40.00	59.00	5.50	180.98
9:00	86.90	40.70	60.70	5.50	226.27
10:00	90.70	42.00	62.30	5.50	261.30
11:00	93.90	42.20	63.70	10.30 ~~5.50~~	282.71

Data based upon average values for the period 2 July - 1 August 1960.

* wind speed is the highest daily average wind speed for the period

Table 9.2-8

Heat Loads Rates Used in UHS Analysis

I. Core Decay Heat Load[(1)]

See Table 6.2-11

II. Reactor Coolant Sensible Heat Load[(1)]

The energy (414 x 10^6 BTU referenced to 32°F) of the reactor coolant is accounted for by starting the suppression pool at 150°F.

III. Reactor Vessel, Piping, and Core Sensible Heat Load[(1)]

Time (hours)	Rate(10^6 BTU/hr)
$t \leq 24$	8.14
$t > 24$	negligible

IV. Metal-Water Reaction Heat Load [(1)]

Time (hours)	Rate (10^6 BTU/hr)
$t \leq 1$	.47
$t \geq 1$	negligible

V. ECCS Pump Work Load [(1)(2)(3)]

Time (hours)	Rate (10^6 BTU/hr)
$t \leq 8$	12.35
$t > 8$	5.49

VI. HPCS (Div. 3) Service Water System Heat Load[(3)(4)]

Time (hours)	Rate (10^6 BTU/hr)
$t \leq 8$	8.73
$t > 8$	0

VII. Constant Div. 1 Service Water System Heat Load(5)(6)

Time (hours)	Rate (10^6 BTU/hr)
$t \geq 0$	18.18

VIII. Fuel Pool Heat Load(7)

Time (hours)	Rate ($x10^6$ BTU/hr)
$0 \leq t \leq 10$	0
$10 < t \leq 20$	.5
20	.54
22	.76
24	1.09
26	1.41
28	1.74
30	1.96
32	2.39
34	2.61
36	2.82
38	3.04
40	3.26
42	3.48
44	3.69
46	3.86
48	4.02
50	4.07
$t \geq 52$	4.13

Table 9.2-8 (continued)

Notes:

(1) Rejected initially to the suppression pool and subsequently transferred by the RHR heat exchangers to the UHS.

(2)

RHR pump	1.93×10^6 BTU/hr
LPCS pump	3.56×10^6 BTU/hr
HPCS pump	6.86×10^6 BTU/hr

(3) HPCS system and HPCS SW system shut down after 8 hours. LPCS system and RHR loop A maintain long-term cooling.

(4)

HPCS service water pump work	$.13 \times 10^6$ BTU/hr
HPCS diesel coolers	7.40×10^6 BTU/hr
HPCS coolers (Table 9.2-5)	1.20×10^6 BTU/hr

(5)

Div. I Service water pump work	3.82×10^6 BTU/hr
Div. I Diesel Generator	11.69×10^6 BTU/hr
Coolers and misc. equip. (Table 9.2-5)	2.67×10^6 BTU/hr

(6) Excludes fuel pool and RHR heat exchanger heat loads

(7) Added to the RHR service water system

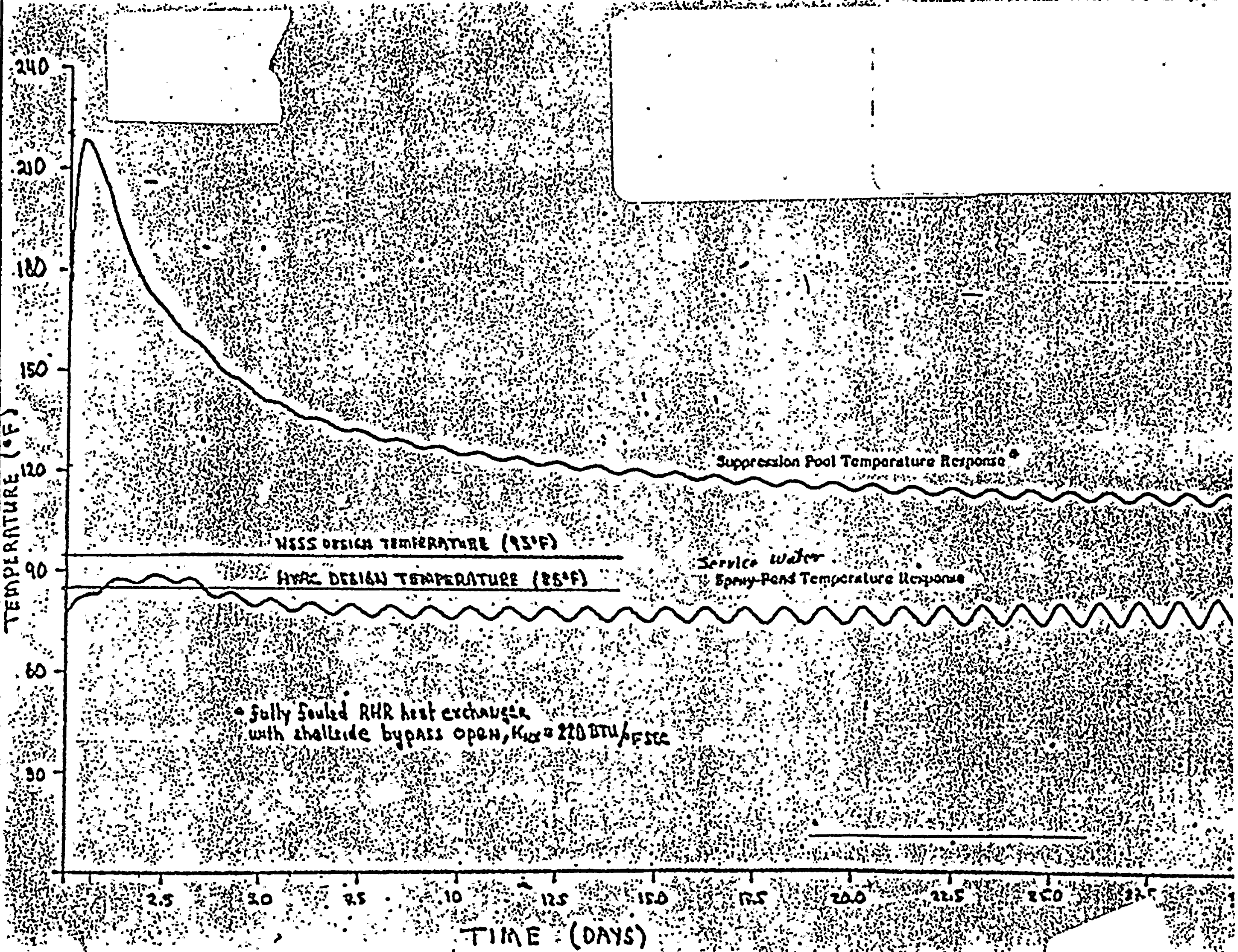
Table 9.2-9

Integrated Heat Data - WNP-2 UHS Re-analysis

Time After LOCA Min.	Q Decay(1)	Q Sens(2)	Q Aux 1(3)	Q Aux 2(4)	Q Aux 3(5)	Q Total(6)	Q SW(7)
	10^7 BTU						
0	0	0	0	0	0	0	0
1	3.51	.014	.020	.030	.015	3.59	.174
2	4.28	.027	.041	.061	.029	4.44	.355
4	5.57	.054	.083	.121	.058	5.89	.719
10	8.72	.136	.205	.303	.146	9.51	1.83
20	13.02	.271	.413	.606	.291	14.62	3.75
40	20.26	.543	.823	1.21	.582	23.45	7.80
90	35.16	1.22	1.85	2.73	1.31	42.32	18.69
120(2H)	43.03	1.63	2.48	3.64	1.75	52.57	25.57
240(4H)	70.65	3.26	4.94	7.27	3.49	89.66	54.51
360(6H)	94.84	4.88	7.41	10.91	5.24	123.3	84.37
480(8H)	117.0	6.51	9.88	14.54	6.98	155.0	114.3
720(12H)	157.6	9.77	12.08	21.92	6.98	208.4	172.4
960(16H)	194.9	13.02	14.27	29.39	6.98	258.6	227.3
1200(20H)	229.9	16.28	16.47	36.86	6.98	306.5	279.3
1440(1D)	263.1	19.54	18.66	44.45	6.98	352.8	328.6
2160(1½D)	354.5	19.54	25.25	68.67	6.98	475.0	461.1
2880(2D)	435.3	19.54	31.84	94.64	6.98	588.3	581.1
4320(3D)	577.2	19.54	45.02	148.2	6.98	796.9	796.6
5760(4D)	702.3	19.54	58.19	201.7	6.98	988.8	995.4
7200(5D)	816.2	19.54	71.37	255.3	6.98	1169	1182
8640(6D)	922.0	19.54	84.54	308.8	6.98	1342	1358
11520(8D)	1116	19.54	110.9	415.9	6.98	1669	1689
14400(10D)	1292	19.54	137.2	523.0	6.98	1979	2001
17280(12D)	1456	19.54	163.6	630.1	6.98	2276	2300
23040(16D)	1756	19.54	216.3	844.2	6.98	2843	2870
28800(20D)	2029	19.54	269.0	1058	6.98	3383	3412
34560(24D	2282	19.54	321.7	1273	6.98	3903	3935
43200(30D)	2635	19.54	400.8	1594	6.98	4656	4689

Table 9.2-9 (continued)

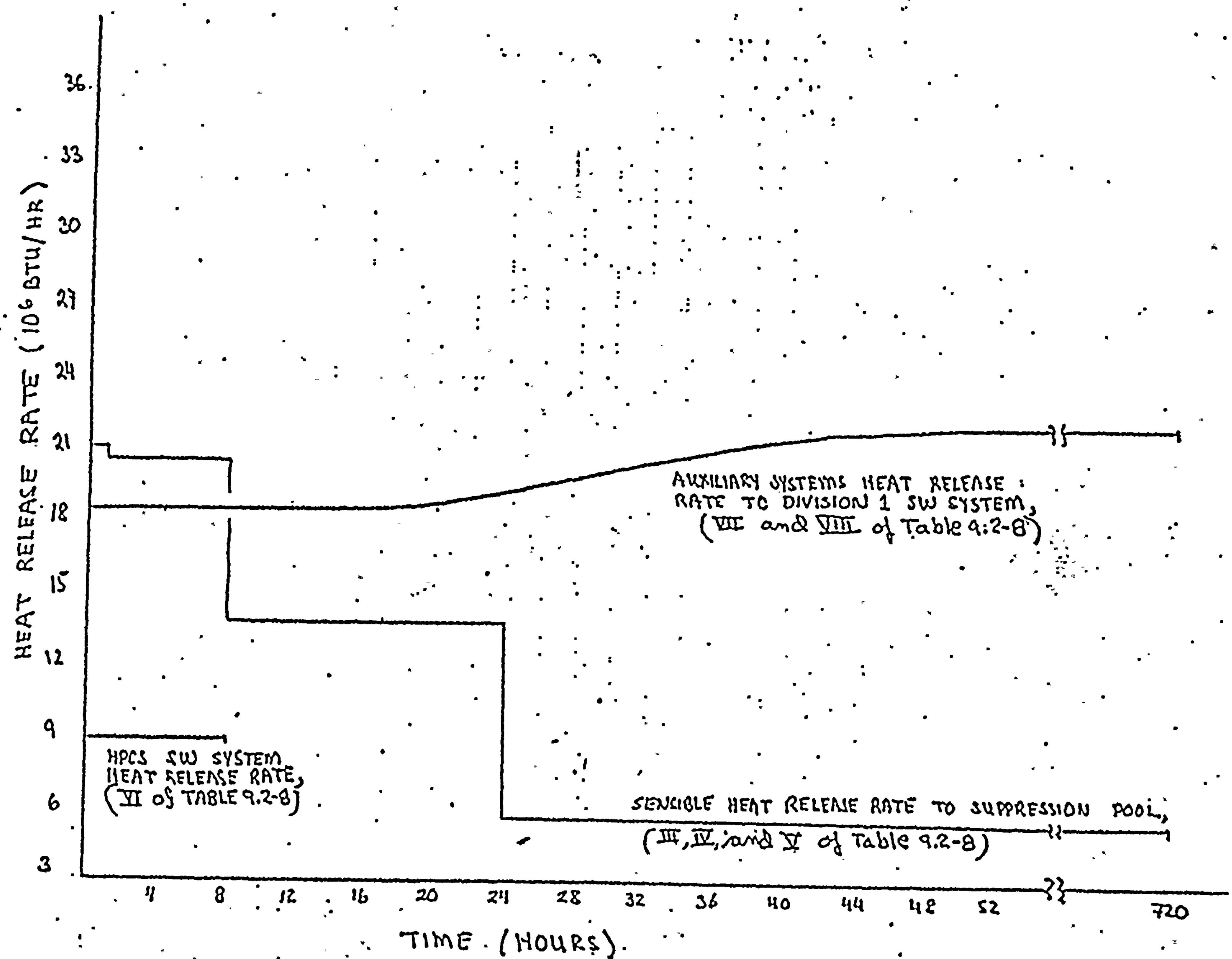
(1)	Q Decay	Integrated core decay heat rejected to suppression pool.
(2)	Q Sensible	Integrated sensible heat rejected by the reactor vessel, piping, and core to the suppression pool.
(3)	Q Auxiliary 1	Integrated heat from ECCS pump work rejected to the suppression pool.
(4)	Q Auxiliary 2	Integrated heat from auxiliary systems rejected to division 1 service water system. This heat includes all sources of heat into division 1 SW system except for the RHR heat exchanger. The RHR heat exchanger transfers heat from the suppression pool to division 1 SW system.
(5)	Q Auxiliary 3	Integrated heat from HPCS service water system. This heat is a straight heat dump into spray pond A.
(6)	Q Total	Sum of Q Decay, Q Sensible, Q Auxiliary 1, Q Auxiliary 2, and Q Auxiliary 3.
(7)	Q Service Water	Sum of Q Auxiliary 2 and the heat rejected by the RHR heat exchanger into Division 1 service water system, i.e., the sum of the heat rejected through the spray nozzles.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

TEMPERATURE RESPONSE
FOLLOWING DESIGN BASIS LOCA

FIG. 9.2-7a

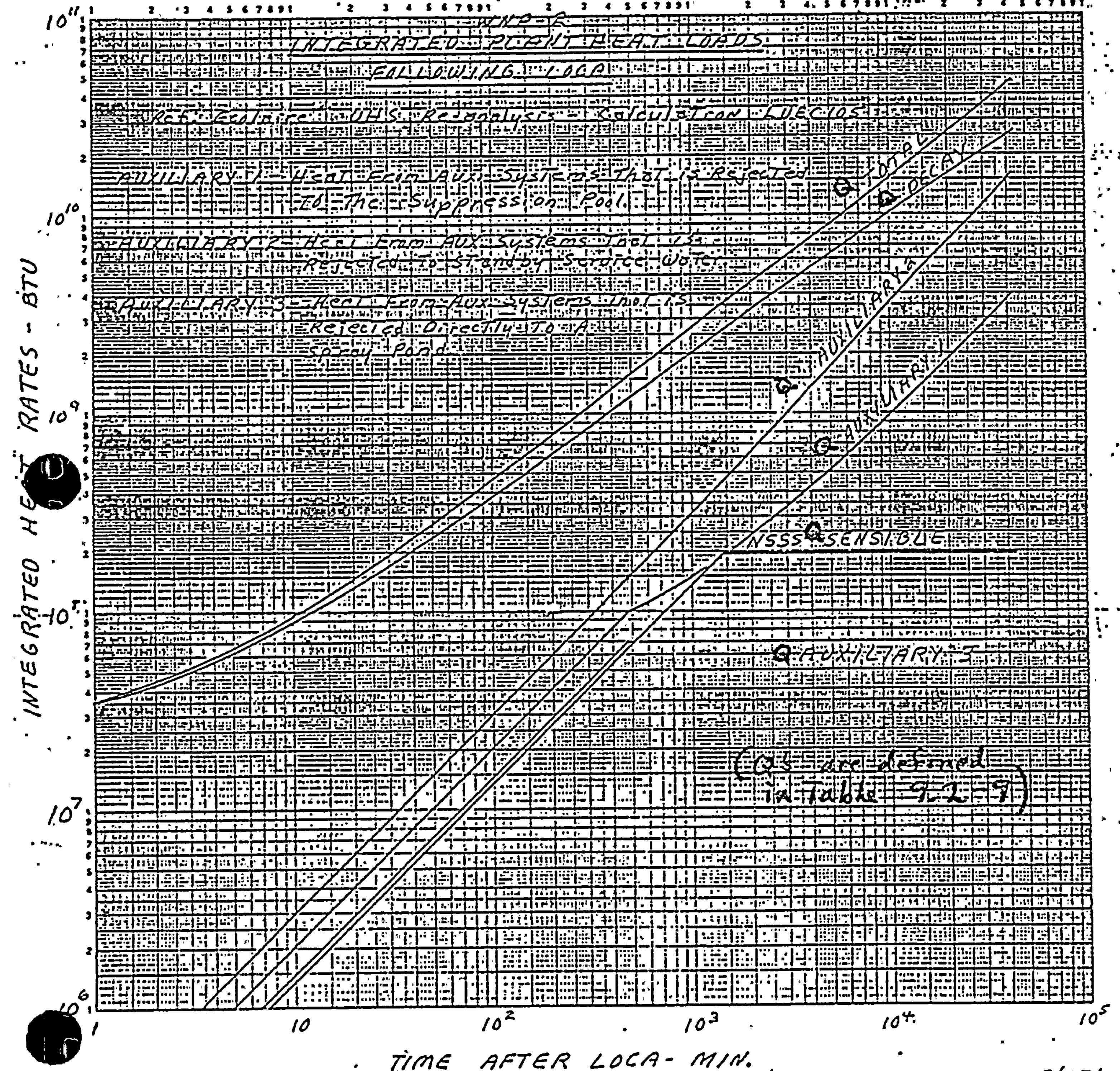


WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

HEAT RELEASE RATE VS TIME ASSUMED IN UHS ANALYSIS

FIGURE 9.2-7b

FIGURE 1



3/27/79
KRC

Fig. 9.2-7c

K+E LOGARITHMIC 3 X 5 CYCLES 46 7520
KEUFFEL & ESSER CO.

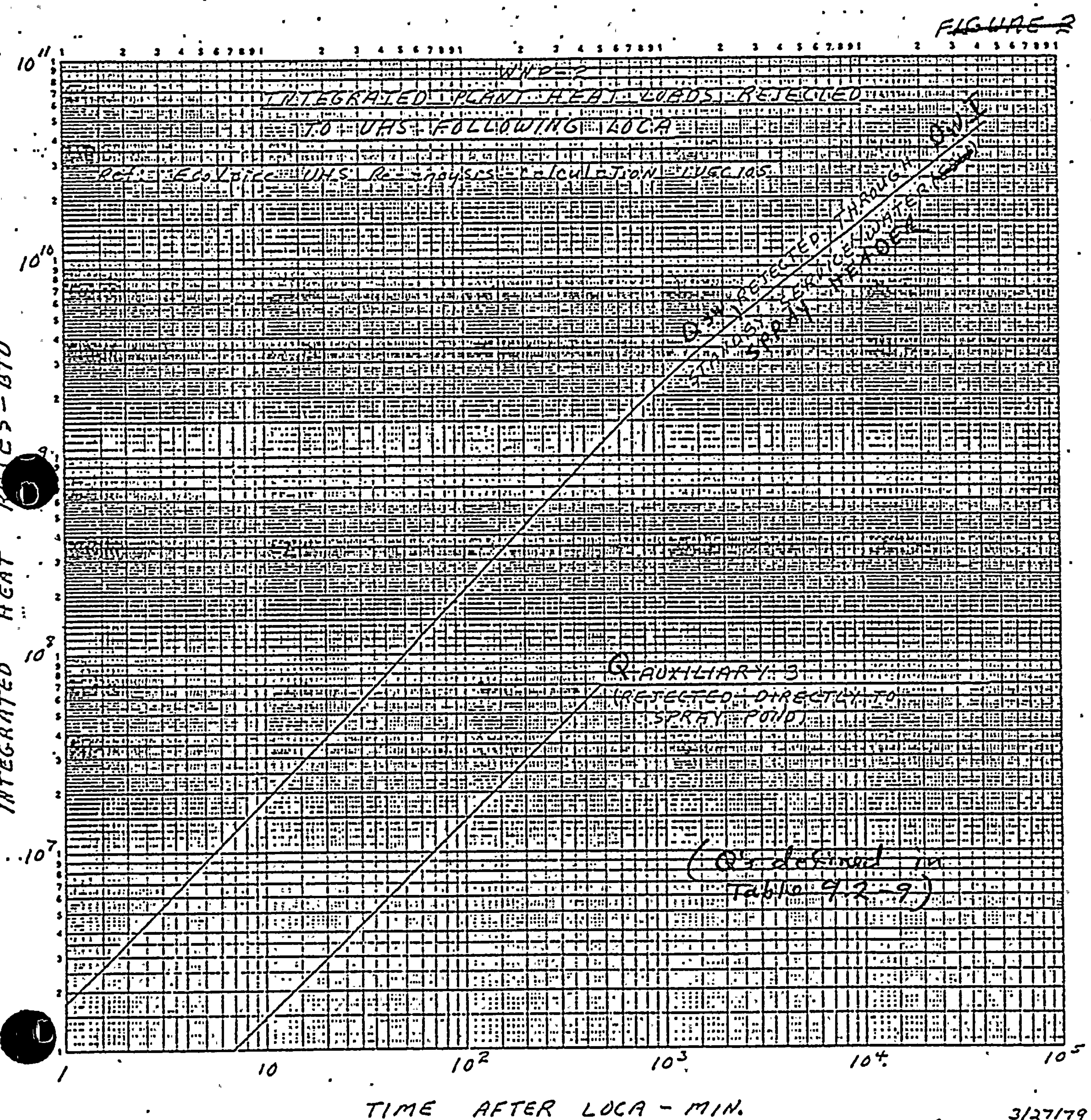


Fig 9.2-7d

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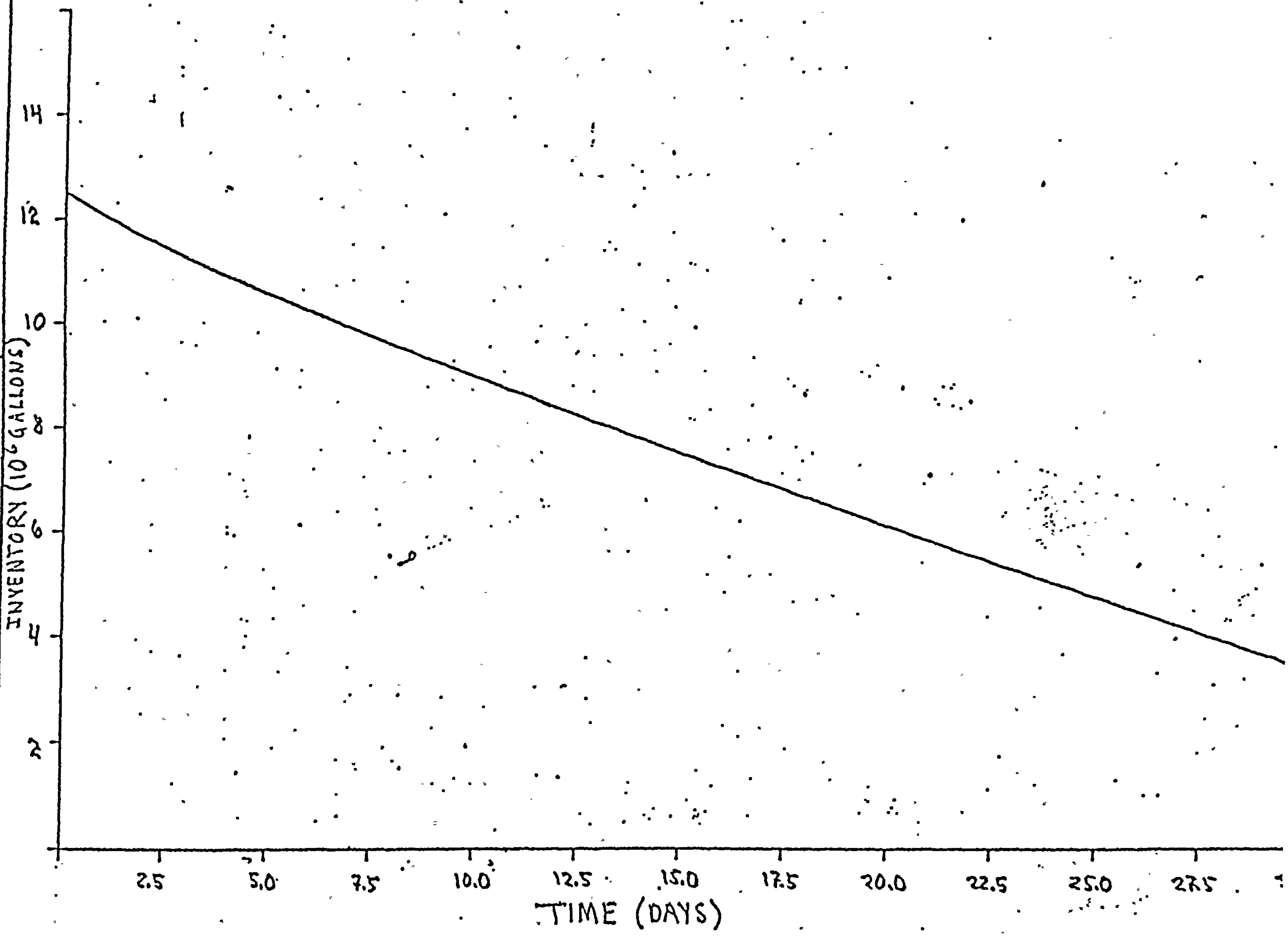
KRC



WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2.

WATER INVENTORY IN UHS
FOLLOWING DESIGN BASIS LOCA

FIGURE
9.2-8

Regulatory Guide 1.27, Rev. 2, January 1976

Ultimate Heat Sink for Nuclear Power Plants

Compliance or Alternate Approach Statement:

WNP-2 does not comply with the guidance set forth in Revision 2 of this regulatory guide.

WNP-2 complies with the intent of the guidance set forth in Revision 1 of this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

The basic design and much of the construction of the spray ponds was completed prior to the issuance of Revision 2 of this regulatory guide. ~~Therefore, thermal analysis was based on Revision 1, and mass loss analysis was based on~~ a 30 day period with the worst dew point depression and average winds during that period. [INSERT 2]

replace with INSERT 1

Two Seismic Category I spray ponds are used, each with a capacity of 6.5 million gallons each. The makeup for these ponds is supplied from the pumphouse at the Columbia River. The makeup water piping is buried under a minimum of 5 feet of Quality Class I fill. The makeup water supply system is utilized only in the event of a design basis tornado, and therefore, it is not designed and constructed to withstand the effects of the OBE and water flow based on severe historical events in the region.

Specific Assessment Reference:

Refer to 9.2.5.

INSERT 1 [Therefore, thermal and mass loss analyses were based on Rev 1 using

Insert 2 [For conservatism, however, in the thermal analysis the worst day of the 30 day period was assumed to repeat until the spray pond temperature peaked (3 day repetition). For conservatism in the mass loss analysis, five times the drift loss for the highest daily average wind speed during the 30 day period was assumed to occur for the entire 30 day period.

Q 010.24
RSP
(9.2.5)

We require that you protect the sprays in the ultimate heat sink from the effects of tornados and tornado missiles.

Response:

As discussed in Section 3.3.2.3, the WNP-2 UHS design provides for continuous water make-up to the spray ponds in the event that both the spray systems are rendered inoperable due to tornado missiles. Therefore the sprays are not required to be protected from the effects of tornado missile since an alternate UHS operating mode (continuous Make-up) is available which is protected from the effects of tornadoes and tornado missiles

Q. 010.25
(9.2.5)

In the event that a tornado siphons water from the ultimate heat sink (UHS), the make-up water pumps will replenish the UHS. Demonstrate that the transformers located in the turbine building and the electric cabling which are both required to operate the make-up pumps, are protected from tornados and tornado missiles.

Response:

As described in section 3.3.2.3, the TMU transformers (TR-75-72 and TR-85-82) are located at gound level in the southeast corner of the turbine building where they are protected by the exterior walls of the turbine building, the reactor building to the south, the service building to the east, and other reinforced concrete interior walls to the north and to the west, and are therefore not considered vulnerable to tornado missile impact. As described in 3.5.2, electrical cabling to the TMU pumphouse is buried at sufficient depth in compacted backfill to provide protection against tornado missiles. Electrical cabling from each transformer is routed separately to two switchgear units at ground level in the southwest corner of the turbine building. For missile trajectories which would jeopardize the TMU transformers, associated cabling, and switchgear, the exterior walls of the turbine building provide adequate protection against design basis missile penetration and spalling.

*Appropriate draft FSAR changes are attached.

The availability of essential electric power to the makeup water pumphouse systems is assured. The electrical lines are underground with sufficient earth cover to resist tornado-generated missiles.

The electrical lines are installed in such manner as to provide two redundant electrical systems from the power source to the makeup water pumphouse. The two electrical systems are physically separated to provide adequate missile protection of one system from the other. At the one end of each system, redundant power source transformers, ~~are provided in separate, isolated missile-proof concrete vaults in the turbine generator building~~. The terminal ends and transformers at the makeup water pumphouse are enclosed within the tornado-resistant pumphouse. Manholes within each system are also designed to withstand tornado generated missiles.

The spray pond piping and supports are designed to withstand the effects of the design basis tornado. The piping system cannot be protected from the impact of tornado generated missiles. In the event of missile damage to one of the pond spray headers, the alternate spray system which is 100% redundant is placed in operation. In the event that both spray systems are rendered inoperative, the cooling tower makeup water system is placed into operation to provide continuous makeup to the spray ponds with Columbia River water, the temperature of which never exceeds 70°F. The cooling tower makeup water system is provided with sufficient protection to prevent its loss of function in the event of a design basis tornado passing over the project site. Since the makeup water flow rate exceeds that of the standby service water systems, and since the makeup water temperature is substantially lower than the standby service water system design temperature of 85°F, the continuous availability of cooling water at a maximum temperature of 70°F is assured. The method of detection of spray pond header failure and procedures for alternate spray pond usage is described in 9.0.

Failure of non-tornado resistant cooling towers due to tornado loads does not endanger Seismic Category I structures since the plant arrangement provides sufficient distance between the cooling towers and Seismic Category I structures.

associated switchgear, and cabling is provided at the ground floor of the turbine building where, for the trajectories required to cause damage to this equipment, they are protected against missile impact and spalling by the exterior walls of the turbine building and the floor slabs overhead.

c. Radwaste and Control Building

The exposed exterior concrete walls and roofs, housing safety related systems, equipment and components, are designed to withstand the effects of the design basis tornado generated missiles. Figures 1.2-3, 1.2-4, 1.2-5, 1.2-9 and 1.2-11 illustrate the radwaste and control building and their relative location in the plant complex.

d. Standby Service Water Pumphouses and Spray Ponds

The exterior walls of both pumphouses are constructed of reinforced concrete and are 2'-4" thick, minimum. This thickness is adequate to withstand design basis tornado generated missiles. In addition, the two pumphouses are redundant to each other. In the event that one pumphouse is inoperable, the other is capable of providing sufficient service water for safe shutdown.

The ability of the spray ponds to tolerate the design basis tornado generated missiles is discussed in 3.3.2.3.

Figure 1.2-14 illustrates the pumphouses and spray ponds.

e. Makeup Water Pumphouse

The exterior walls and roof of the makeup water pumphouse are of reinforced concrete and are sufficiently thick to withstand the effects of the design basis tornado generated missiles as discussed in 3.3.2.3. The exterior walls are 2'-4" thick and the roof slab is 1'-4" thick. Figures 1.2-1 and 1.2-13 furnish its location and arrangement.

f. Turbine Building

Safety-related components in the turbine building are located in areas where tornado missile protection is provided by reinforced concrete exterior walls of a minimum of 18 inches in thickness, and two reinforced concrete slabs overhead at elevation 471' and 501'. Figure 3.5-19 1.2-8 illustrates the turbine building.

f. All openings for heating, ventilation and air conditioning system fresh air intakes (FAI) and exhausts (EXH), in buildings housing safety-related equipment, are protected against externally generated missiles by means of shield walls as indicated in Table 3.5-6. Examples are the louvred openings above the floor elevation 572'-0" in the north and south walls of the reactor building. These openings are protected by a labyrinth of missile shield walls immediately inside the opening.

at least

3.5.3 BARRIER DESIGN PROCEDURES

The design objectives emphasize missile containment and structural integrity without secondary missile generation. Concrete missile barriers are designed in accordance with the modified Petry equation (Reference 3.5-2). In all cases, except for barriers exposed to turbine missiles, a concrete thickness of twice the penetration thickness determined for an infinitely thick slab is provided to prevent perforation, spalling OR scabbing. For discussion of turbine generated missiles see 3.5.1.3. See Table 3.5-5.
or

The formulae used to determine penetration depths into steel barriers are given in 3.5.1.1.2.

The overall response of barriers subject to impact are investigated by the use of general energy equations given in "Introduction to Structural Dynamics", J. M. Biggs (Reference 3.5-9). Upon determination of penetration depth and duration of impact, an effective dynamic force is computed. The additional calculation of the natural period of the target structure and the selection of a ductility ratio facilitates the determination of the required structural resistance. In this manner, missile impact is translated to an equivalent static load in an effort to quantify bending moments and shear. The detailed method used for predicting the overall response of missile barriers, including the forcing function method of determining ductility in structural elements and the basis for the ductility ratios used in the calculations, is provided in Appendix C of the report "Protection Against Pipe Breaks Outside Containment" (Reference 3.5-13) that was presented to and approved by the NRC.

TABLE 3.5-5

DEPTHS OF MISSILE PENETRATIONS INTO CONCRETE

MISSILE	TARGET	PENETRATION DEPTH (1) (in.)
35' UTILITY POLE	Quality Class I structures up to 30' above grade	~~20.6~~ 14.0
STEEL ROD 3" diameter x 3 ft. long.	Quality Class I structures at any elevation	~~8~~ 16.6

Note (1): Equal to twice the penetration depth calculated for an infinitely thick slab.

Q 010.26
(9.2.5)

In Section 9.2.5 of the FSAR, you state that the two ponds which comprise the ultimate heat sink are connected by a siphon that allows water to flow from one pond to the other. Demonstrate that a failure in this siphon line, or in one of the ponds, will not result in draining of both ponds.

Response:

The siphon between the two ponds is a Seismic Category I, Quality Group C,30 inch pipe, whose centerline is 4 ft. 6 in. below the normal water level of the spray ponds.

Therefore a siphon line failure would be considered a passive failure. Applying single failure criteria indicates that if the siphon failed then both SW loops would be operating, thus keeping them at the same level. If one of the SW loops fails, then an additional failure of the passive siphon is not considered credible.

The spray ponds are Seismic Category I structures located below grade with continuous waterstops in all joints and bounded with Quality Class I high density backfill. Both ponds together form the Ultimate Heat Sink, a concept which has been accepted on other plants that only have a single pond which contains the redundant spray networks. Failure of either Pond A or Pond B will result in drainage of the other pond, which results in the same consequence if the WNP-2 UHS were a single pond design. However, as described above and in section 3.8.4.1.5 the spray ponds have been conservatively designed to preclude pond failure.

Q 010.27
RSP
(9.2.7)

We require that you protect the standby service water system from tornado missiles.

Response:

The standby service water system (except for the spray pond spray piping) is protected from tornado missiles. The structures which house the standby service water systems (Reactor Building, DG Building, Control Building, and SW Pumphouse) have been designed to withstand design basis tornado generated missiles as described in section 3.5.1.4.1.

Buried portions of the standby service water system are protected from tornado missiles as described in Section 3.5.2.

See the response to question 10.24 as to why it is not necessary to protect the spray pond spray headers from tornado missiles.

Q 010.28
(9.3.4)

Describe how flooding of safety-related equipment due to backflooding through the equipment and floor drainage system, is prevented. Demonstrate that those portions of the drainage system necessary to prevent backflooding (e.g., check valves) are designed to Seismic Category I criteria and that their system function will be maintained, assuming a single active failure.

Response:

It is assumed that the question is directed to FSAR section 9.3.3.2.2.1, Reactor Building Floor Drains, and not 9.3.4, Chemical and Volume Control System.

As shown on Figure 9.3-8, the floor drain piping in the reactor building drains to one of four sumps listed below.

Floor Drain Sump	Room Location	Rooms Served
FDR-R-1	RHR A Pump Room	RCIC RHR A
FDR-R-2	RHR B Pump Room	RHR B
FDR-R-3	HPCS Pump Room	HPCS CRD
FDR-R-4	RHR C Pump Room	LPCS RHR C

Each of the four downcomers is equipped with instrumentation which alarms in the control room to tell the operator at which elevation an excess of water is collecting in the downcomer. Each sump is equipped with level intrumentation which: 1) controls the sump pumps, 2) alarms in the control room (on high sump level), and 3) initiates closure of the isolation valves in the downcomers and in the piping between interconnected rooms. Not currently shown on Figure 9.3-8 are Class 1E level instrumentation to be installed just above floor level in each ECCS pump room. This instrumentation will alarm in the control room.

The floor drain system is analyzed against the potential sources of flooding within the reactor building, i.e. pipe break outside containment and passive failures in the ECCS during post-LOCA long term cooling. Using the acceptance criteria for either event (Standard Review Plan 3.6.1 and Reactor Systems Branch Technical Position, Leak Detection Requirements for ECCS Passive Failures), the floor drain system design is acceptable in mitigating the consequences of flooding ECCS pump rooms.

The effects of pipe breaks outside containment are addressed in Section 3.6.1.11.4, i.e. ruptures in fluid systems have no effect on the ability to bring the reactor to a cold shutdown condition. Single random active failures are assumed in the analysis and credit is taken for systems not affected by the flooding. As stated in Section 3.6.1, these assumptions and the approach taken are consistent with the guidance of Branch Technical Position APCSB 3-1. This is in conformance with the criteria of Standard Review Plan 3.6.1, March 1975.

The effects of passive failures in the ECCS during post-LOCA long term cooling is addressed in the response to Question 212.003. The largest passive failure has been identified as the total failure of an RHR pump seal and it is equivalent to a 23 gpm leak. Class 1E instrumentation in each ECCS pump room will detect the leak and give the operator at least 44 hours to identify and isolate the passive failure before it has any additional adverse effects on ECCS operation.

[illegible]

QUESTION 010.29

Demonstrate that the heating, ventilating and air conditioning systems (HVAC) for the engineered safety features are protected from tornado missiles.

RESPONSE

Except for standby service water piping, control room remote air intake piping and control room remote air intakes, the HVAC systems serving engineered safety features are all located within reinforced concrete structures designed to withstand the effects of design basis tornado missiles. These structures include the reactor building, radwaste and control building, diesel generator building and standby service water pumphouses. Design of the buildings, including protection for HVAC system air intakes and exhausts, is discussed in 3.5.

The standby service water piping runs between the standby service water pumphouses and the reactor building and supplies water to the cooling coils of critical HVAC equipment during a design basis accident. The control room remote air intake lines run between the remote air intakes and the radwaste and reactor buildings, respectively, to supply control room pressurization and makeup air during accidents involving radioactive releases. Since these piping runs are all covered to a depth of over five feet with Class I compacted earth fill, they are adequately protected from tornado missiles as discussed in 3.5.2.

The two control room remote air intakes are over 200 feet from any major plant structure and are located in a northwest and southeast direction, respectively, from the turbine-generator, radwaste and reactor builidng complex. The intakes are of reinforced concrete construction designed to withstand design basis tornado missiles.

The roof slab of each intake is 12 feet square and 2 feet thick with a grated 3 foot square opening for the intake air. Eight 4" pipes surround the intake opening and serve both as barriers and as alternate air intakes in the event the grated opening is blocked or damaged. The top of the roof slab is 15 inches above the surrounding grade level. The walls and floor of the intake structure are 18 inches thick and are buried to a depth of approximatey 9 feet in Class I compacted earth fill. An internal barrier is provided as additional protection for the intake piping and to assure an unobstructed path for the flow of control room air. This barrier, which is 4 feet wide and 18 inches thick, is supported from two sidewalls and two one by two foot columns. Details of the intake structure are shown on Figure 3.5-52.*

*To be supplied in FSAR Amendment No. 4

Q 010.30
(9.4.0)

You state in the FSAR that the outdoor design temperature range for the HVAC is 0°F to 105°F. However, you also indicate on Page 9.4-2 of the FSAR, that the extreme outside temperature range is minus 27°F to 115°F. Provide the results of your analysis which demonstrate that the functional capability of safety-related equipment will not be impaired by the outdoor temperatures which would occur during these extreme meteorological conditions. The effect of extreme low temperatures on safety-related equipment located outdoors should also be discussed.

Response:

This question has been addressed in 9.4 (Page 9.4-2). Even though the normal outside temperature range for the design of the HVAC systems was 0°F to 105°F, as stated in the FSAR, operation at the extreme conditions of -27°F and 115°F were also evaluated. Where necessary, equipment was selected to assure operation of safety related systems during the extreme conditions.

As discussed in 9.4.7, 9.4.8 and 9.4.10, the heating equipment for the diesel generator building, diesel generator cable corridor and standby service water pumphouses are capable of maintaining temperatures at 35°F or above during the extreme -27°F conditions. In sizing heating equipment for primary operating areas of the plant, such as, the control room, reactor building or turbine-generator building, no credit was taken for heating available from plant lighting or operating equipment. Including these additional heating sources, even in a shut-down mode, the existing heating equipment is adequate to maintain the areas served above minimum set temperatures during the extreme cold condition.

The extent and duration of any room temperature increases which may result during operation at the extreme summer temperature of 115°F with the existing cooling systems, will not be sufficient to degrade the operating capability of any safety related equipment.

Q 010.31
(9.4.4)

The Radwaste Building chilled water system which is not designed to Seismic Category I criteria, is connected to the HVAC system at the control room, and to the standby service water system. Provide your analysis which demonstrates that the potential failure of the Radwaste Building chilled water system during an earthquake will not cause an unacceptable degradation of the control room HVAC system and the standby service water system.

Response:

The Radwaste Building chilled water system is completely isolated from the standby service water system. Both control room air handling units are Seismic Category I and are provided with two "N" stamped cooling coils. One coil is connected to the Radwaste building chilled water system and the other coil is connected to the standby service water system. Failure of the chilled water system will not adversely effect control room or the standby service water system. Please see 9.4.1.2.1 for additional information.

Q 010.32
(9.4.7)

Demonstrate that the ventilation system of the Diesel Generator fuel oil pump room is designed to Seismic Category I criteria, and receives power from the Class 1E buses.

Response:

Please see 9.4.7.3, Figure 9.4-7 and Figure 8.3-1d for requested information.

Q 010.33
(9.4.7)

Provide your analysis which demonstrates that the potential failure of the heaters in the Diesel Generator HVAC System which are not designed to Seismic Category I criteria, will not have an adverse affect on the functional capability of either the Diesel Generator or the Diesel Generator HVAC System.

Response:

There are two types of heaters in the diesel generator spaces, electric unit heaters in the diesel oil pump rooms and electric heating coils in the duct systems in the diesel engine rooms themselves. The electric unit heaters in the pump rooms are Seismic Category II. These heaters are supported as Seismic Category I, however, and can fail in place without affecting any safety related equipment. Oil pump room unit heaters are used only for maintenance during cold weather for personnel comfort. The heating coils in the generator room themselves are Seismic Category I and are designated Class 1E.

Q 010.34
(10.4.5)

Your response to Item 010.09 is unacceptable. Specifically, your analysis of flooding due to failure of the circulating water system is based on a crack whose area is equal to one-quarter of the pipe diameter times the pipe thickness (.5t X .5d). Provide an analysis of flooding due to a postulated failure of the expansion joint in the circulating water system assuming a double-ended guillotine break at this location.

Response:

The original response to Item 010.09 has been rewritten for clarity (see 10.4.5).*

The double-ended guillotine break referred to above was not considered. The circulating water system is a moderate energy system by definition. Therefore, in accordance with NRC Standard Review Plan Section 3.6.1, 3.6.2, and 10.4.5, and the associated Branch Technical Position MEB 3-1, the criteria for a postulated failure shall be a through - wall leakage crack of the type addressed in the written response (10.4.5). In any case, as stated at the end of 10.4.5, circulating water piping is located remote from any safety-related equipment. The piping is located in a large room containing little other equipment and no safety-related equipment. Accordingly, safety-related equipment is not vulnerable to environmental effects of a circulating water pipe rupture. The pipe exits the room below grade in its routing to and from the cooling towers. It should also be noted that the condenser is located on grade level. Therefore, water above the floor elevation will drain outside and not collect other than in collection basins.

*See attached draft pages

10.4.5.3 Safety Evaluation

The circulating water system is a non-safety related system. Consequently, the circulating water system is not designed to Seismic Category I requirements. Refer to 9.2.5 for a description of the ultimate heat sink which is designed to perform safety-related functions.

The condenser design assures that the pressure on the tube side is always maintained higher than the pressure on the shell side, thus eliminating leakage into the circulating water system should tube failure occur. Consequently, the design of the circulating water system precludes radioactive leakage into the system.

Periodic injection of chlorine is performed for biocide treatment, and sulfuric acid is added for scale-corrosion control within the circulating water system. An analysis of the transportation, handling, storage, and utilization of chlorine is presented in 6.4.

A detailed evaluation was performed to determine the effects of a postulated failure in the circulating water system inside the turbine building. For this analysis a moderate energy crack was postulated to occur in the circulating water system barrier, (e.g., the rubber expansion joints) at the inlet to the main condenser. The inlet side was selected because it yields the severest results.

The entire condenser area is drained by means of sumps (see Figure 9.3-9), each equipped with duplex pumps. Sumps T-2 and T-3, servicing the inlet and outlet of the condenser, each have 50 gpm pumps. Each of these sumps is equipped with a level alarm and is therefore capable of detecting a circulating water system barrier failure. The level alarm will annunciate in the main control room upon reaching high level, providing a means of detecting the postulated failure within 5 minutes.

The crack area for this postulated failure was assumed to be equal to 1/2 the pipe diameter times 1/2 the pipe wall thickness.

$$A = \frac{d}{2} \times \frac{t}{2} \quad \text{(see 3.6.2.1.4.2.b)}$$

The flow exiting from such a crack would be an orifice flow. The head at expansion joint for normal ~~through~~ three pump operation at 186,000 gpm each was determined (from system energy gradients) to be 90 feet. The flow for these conditions was calculated to be:

Q = 1,737 gpm

The system has different operating pressures for the various modes of pump operation. The piping was designed for an internal pressure of 60 psig, which is well above the design energy gradient.

The motor operated inlet and outlet valves at the condenser are designed and manufactured to close in 60 seconds to avoid excessive pressures caused by fast valve closure. Therefore, rapid valve closure is not a consideration. After closure of the inlet and outlet valves, however, the system will be operating with 2/3 of the condenser capacity. With 3 circulating water pump operation and 2 sections of the condenser in operation, the system flow as determined from the pump operating point diagram will be approximately 450,000 gpm. Comparing the system energy gradients for this mode of operation to that when all three condenser units are in operation, the resultant difference in pressures will be:

At the inlet side, an increase of approximately 4.3 ft. of head (2 psi) occurs

At the outlet side, a decrease of approximately 5.2 ft of head (2 psi) occurs

Detection of the postulated failure will occur within 5 minutes, as described above, by the annunciation in the control room of the sump high level alarm. It is assumed that there will be a 15 minute time allowance for an operator in the control room to check the circulating water system barriers and close both the inlet and outlet valves of one unit of the condenser as may be required. This closure is accomplished by the activation of a remote manual switch in the control room, and therefore no control circuitry time delays nor coastdown times are involved. Flow will continue, however, after valve closure for about 106 minutes at a decreasing rate, until the remaining water from the condenser is completely discharged.

In the first 5 minutes after a crack, 8,435 gallons of water will spill into the inlet basin. The capacity of each basin and its capability to store excess flow were calculated to be as follows:

a. Inlet basin: 22,500 gallons from El. 436 to El. 441

b. Outlet basin: 27,500 gallons from El. 436 to El. 441

c. Net volume under condenser: 180,500 gallons from El. 433 to El. 441.

The time required to fill the inlet basin, after a postulated crack occurs, is computed to be 13.3 minutes. This includes the 50 gpm outflow from the sump pump. The circulating water leakage flow will continue for 6.7 minutes after filling the inlet basin, until reaching the total estimated shutoff time of 20 minutes. It can be assumed that 10% of this water will flow out over the floor at El. 441, and the remainder, about 10,170 gallons, will flow into the condenser basin area. During this same time period, 4 sump pumps in the condenser basin area will have alternately pumped out 670 gallons, leaving 9500 gallons or 0.42 feet of water in the condenser basin. The rate of rise of water, therefore, is 0.021 ft/min during the first 20 minutes after the postulated crack occurs. Note that on high sump level, both pumps run simultaneously rather than alternately, thus doubling the calculated outflow capacity.

After the valves are closed, the water contained in the condenser unit water box will continue to discharge to the area. The quantity of water remaining is estimated to be 87,000 gallons. The flow will vary with a diminishing head, the head going from about 25 feet to zero feet. Using a 20 ft head and the same orifice flow criteria, the rate of flow will be approximately 819 gpm, discharging the remaining water in about 106 minutes. There will be an outflow from all the sump pumps of 150 gpm, with 10% of the flow from the crack again assumed to flow out over the floor. The water will accumulate in the condenser basin at about 590 gpm. After 106 minutes, the water level in this basin will rise an additional 2.77 feet, or 0.0261 ft/min. The total height of water when the discharge has stopped is therefore 3.19 feet to El. 436.19.

There are no safety-ralated system components that could be affected by the flood elevation established above. Additionally, there are no safety-related electrical systems or system components that could be potentially submerged. ~~All safety-related electrical systems routed through the turbine generator building enter the building at sealed wall penetrations inside conduits at El. 490 and terminate in instrument racks on El. 501 as well as terminal boxes mounted above El. 501. The flood waters cannot reach these elevations in the event of a failure as postulated.~~

Discharge operation of water accumulated under the condenser shall be performed in accordance with radioactivity checking requirements for sump discharges.

10.4.5.4 Tests and Inspections

All system components, except the condenser, are accessible during operation and may be inspected visually. The circulating water pumps are tested in accordance with the Hydraulic Institute Standards.

In addition, the circ. water piping is located in a large rooms ~~at each end of the condenser~~ ~~containing~~ containing little other equipment and no safety related equipment. Accordingly, spray effects are of ~~no~~ no consequence. The pipes and fire exit the rooms below grade in their routing to the cooling Towers. Also, the floor onto which water would spill in event of a break is grade level. As a result, excess water would accumulate either in drainage basins or leak outside the building.

Responses to previous questions:

Hydrology-Meteorology (371.6)

Geosciences Branch (360.4, 360.5)

Q. 371.6

Provide the results of a transient analysis to determine the adequacy of the ultimate heat sink spray ponds under emergency conditions, including consideration of the requirements for both the temperature and volume of the water. (Refer to Regulatory Guide 1.27, Rev. 2, for guidance on this matter.) Provide the basis for any assumptions used in your analysis and a discussion of your analytical techniques.

Response:

The mass loss and thermal transient analyses for the UHS following a design basis LOCA are presented in revised section 9.2.5. See the response to Q 010.23 for additional information.*

*draft changes to 9.2.5 are included with Q 10.23.

Q. 360.4

In the Weston Geophysical Research, Inc. report, "Qualitative Aeromagnetic Evaluation of Structures in the Columbia Plateau and adjacent Cascade mountain Area," March 28, 1978, Figure 13 shows several north to northwest trending aeromagnetic linears in the vicinity of Badger Mountain and Jump Off Joe Anticline. However, the Weston report does not discuss the origin or interpretations of these particular linears. The north trending linear crossing the Columbia River at the junction with the Snake River has an apparent offset of the magnetic low defining the Rattlesnake Hills anomaly. Since these aeromagnetic linears trend toward the WNP-2 site, provide: (1) an interpretation of these features, including but not limited to the potential for their continuation to the north to near site area: and (2) a discussion of the fault parameters, if such an interpretation is proposed.

Response:

The concerns raised in this question relate to recent information which post-dates the information now before the staff. The reference letter proposes a meeting to update the staff with respect to this information. As stated in the letter, a generic report is scheduled for early fall 1979 which will place this information in perspective and respond to the concerns of this question.

Reference: Letter, D.L. Renberger (WPPSS) to O.D. Parr (NRC), "Update of Geological Studies", dated April 27, 1979.

Q. 360.5

Some of the data and discussions in the FSAR of those Columbia Plateau structures relevant to the WNP-2 site are slightly different from the information provided in Amendment 23 to the WNP 1 & 4 PSAR (Docket Nos. 50-460 and 50-513). For example, with regard to the Wallula Gap Fault, your FSAR states that the "...probable fault movement occurred after the deposition of the Touchet beds, and thus less than 12,000 years ago." However, in Appendix 2RH.4 of the WNP 1 & 4 PSAR (Amendment 23), you indicate that the fault is older than the Quaternary Kennewick fanglomerate based on trenching. Additionally, in this same amendment to the WNP 1 & 4 PSAR, you indicate that the faulting along the Horse Heaven Hill Anticline occurred about 3.5 million years before the present (mybp). The WNP-2 FSAR does not discuss this particular point but, rather, questions the existences of faulting along the Horse Heaven Hill Anticline and indicates that it could be the sole result of folding. Clarify these apparent discrepancies and provide cross-references in the WNP-2 FSAR to the appropriate sections of the WNP 1 & 4 PSAR.

Response:

The concerns raised in this question relate to recent information which post-dates the information now before the staff. The reference letter proposes a meeting to update the staff with respect to this information. As stated in the letter, a generic report is scheduled for early fall 1979 which will place this information in perspective and respond to the concerns of this question.

Reference: Letter, D.L. Renberger (WPPSS) to O.D. Parr (NRC), "Update of Geological Studies", dated April 27, 1979.

[illegible]