Washington Public Power Supply System
A JOINT OPERATING AGENCY

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April 23, 1976

Docket Nos. 50-460
50-513

Director of Nuclear Reactor Regulation
ATTN: Mr. John F. Stolz, Chief
Light Water Reactors Project
Branch No. 1
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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APR 26 1976
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Subject: WPPSS NUCLEAR PROJECTS NOS. 1 & 4
PROPOSED PLANT DESIGN CHANGES

Dear Mr. Stolz:

This letter is to inform the Regulatory Staff of proposed design changes for WNP-1/4. These changes, which are discussed in detail in the attachments, are:

1. Seismic Design and Safety Classification of the Component Cooling Water System (Attachment A)

   APCSB CSB

   The component cooling water system (CCWS), as described in the PSAR is entirely seismic Category I and ASME III, Class 3. It is proposed to separate the CCWS into a safety related Category I and ASME III, Class 3 subsystem and a non-safety related, non-Category I and non-ASME III subsystem. Isolation valves would separate the two subsystems.

2. Pipe Break Criteria (Attachment B)

   APCSB STB MEB MTEB

   In Section 3.6 of the PSAR we committed to design for pipe breaks inside and outside Containment so as to comply with Regulatory Guide 1.46 and NRC documents titled, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment" and "Pipe Whip Analysis". We also indicated compliance to the extent practicable to the NRC letter of July 12, 1973, to the industry concerning pipe breaks outside Containment. We now propose to design for pipe breaks outside Containment using NRC Branch Technical Positions APCSB 3-1 and MEB 3-1 in lieu of the previously mentioned documents. Inside Containment Regulatory Guide 1.46 will be used, however, MEB 3-1 will be used to determine break types, orientations and locations. The definitions of high energy, moderate energy and normal operations will be taken from APCSB 3-1 for breaks inside and outside Containment.

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April 23, 1976

3. Seismic Design of Cranes (Attachment C)

ARCSB

In PSAR Table 3.2-2 the polar crane, cask handling crane and new fuel handling crane are specified to be non-Category I, but it is required that they be designed to prevent an uncontrolled lowering of the load in the event of an SSE. We have recently determined that the drop of any crane supported load will not impair the safe shutdown capability of the plant or cause a radioactive release in excess of a small fraction of 10CFR100 limits. We, therefore, propose to not require that these cranes support their loads in the event of an SSE.

4. Onsite Power System (Attachment D)

EICSB

The PSAR currently describes a four channel wiring and cable system. With such a system it is necessary for two of the channels (C and D) to include non-safety related circuits. It is proposed to add a fifth channel which would include all non-safety related circuits. This arrangement would eliminate a large amount of the associated circuits which were such only because they were forced to share a four channel raceway system with the safety related circuits.

The attachments describe in greater detail for each proposed change the current PSAR commitment, the proposed new design, the applicable principal criteria, and the safety significance of the change. In all cases we have concluded that the change does not represent a nonconservative departure from the principal engineering methods and criteria currently discussed and committed to in the PSAR.

The procurement of equipment and design of the facility is proceeding based upon design changes. Therefore, we request early notice of any concerns the Staff may have with these design changes.

Very truly yours,

D L Renberger

D. L. RENBERGER
Assistant Director
Generation & Technology

DLR:AGH:km

Attachments

cc: CR Bryant - BPA, w/att.
JB Knotts - Conner & Knotts, w/att.
JR Schmieder - UE&C, w/att.

Attachment A

Seismic Design of the
Component Cooling Water System

The Component Cooling Water (CCW) system as depicted in the PSAR provides ASME III, Class 3 and Seismic Category I piping to all components served by CCW regardless of the ANS safety class of the individual component. The CCW system will be re-designed to limit the ASME III, Class 3 piping only to those components important to plant safety as defined and required by Regulatory Guide 1.26 (Quality Group classifications for water-steam and radioactive waste containing components of nuclear power plants). The CCW system pumps, heat exchangers and the piping to and from components not important to plant safety are being re-designed in accordance with the requirements of Regulatory Guide 1.26, Quality Group D. Provisions are made in the system design to automatically isolate the safety related portions of the system from the non-safety related portions at the time a safety function is required. The automatic isolation provisions have been designed to be single-failure proof such that no single failure will prevent a safe shutdown of the plant. Cross-connects for the Nuclear Service Water (NSW) system are provided to assure cooling water to those components important to plant safety under all transients and conditions, such as SSE, LOOP, LOCA or any combination thereof.

Regulatory Guide 1.26 requires the primary and secondary cooling water systems associated with decay heat removal for the spent fuel storage facilities, as well as "cooling and seal water systems or portions of these systems and components important to safety such as reactor coolant pumps, diesels and control room" be designed to NRC Quality Group C (ASME III, Class 3, Seismic Category I) requirements. In addition, Regulatory Guide 1.29 requires these systems or portions of these systems be designed to withstand the effects of a SSE. Our interpretation of the above requirements is that, in addition to the components listed in Regulatory Guide 1.26 (spent fuel pool heat exchangers and the cooling water side of the RCP seal heat exchangers) the letdown coolers and the seal return coolers must also be supplied with ASME III, Class 3 and Seismic Category I cooling. However only the spent fuel pool heat exchangers require cooling during a LOCA and/or SSE. Under these conditions the CCW system will receive cooling water from one of the Shutdown Cooling Water (SCW) trains of the Nuclear Service Water System through ASME III, Class 3 and Seismic Category I cross-connects. The cross-connects are interlocked such that the cross-connect cannot be established unless the cross-connects to the other SCW train are closed and all NNS portions of the CCW system have been isolated. The attached figure is a functional flow diagram of the re-designed CCW system showing the cross-connects to the SCW system.

The following is a summary of the salient features of the CCW system during normal conditions and other transient and accident conditions such as LOOP, LOCA, SSE and any combinations thereof.

Normal

The CCW system during normal operation is the same as described in the PSAR. No single failure in the CCW system will prevent safe shutdown and cooldown.

LOOP

CCW pumps will trip on LOOP. Pressure switches located in the LOOP pump discharge piping will automatically start the LOOP cooling pumps. All loads

D. L. Renberger to John F. Stolz, entitled "Proposed Plant Design Changes"

STATE OF WASHINGTON)
) ss
COUNTY OF BENTON)

D. L. RENBERGER, Being first duly sworn, deposes and says: That he is the Assistant Director, Generation and 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 April 23, 1976

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 23rd day of April, 1976.

Reba B. Helgeson
Notary Public in and for the State of Washington
Residing at Richland

not required will be isolated from the CCW system. Since each nonessential load has both an inlet and outlet automatic isolation valve, the failure of any one single valve to close will not prevent supplying required loads. The valves which automatically close are indicated on the attached drawing with "L" written next to the valve operator. The spent fuel pool heat exchanger does not require cooling immediately after a LOOP. When cooling water demand and heat loads from the other components have decreased, cooling water can be diverted to the spent fuel pool heat exchanger. This will be accomplished prior to the spent fuel pool temperature reaching 200°F (approximately seventeen hours).

SSE

Following a postulated SSE, cooling is required for the spent fuel pool heat exchangers only. If the CCW system remains functional, normal operation can continue, however, if the CCW pumps fail or if there is a rupture in the safety class NNS portion of the CCW system, the CCW/SCW cross-connect must be established. Since the spent fuel pool can be operated safely without cooling for approximately seventeen hours, the SCW/CCW cross-connect will not be automatically established. If the SSE causes a rupture in a non-nuclear portion of the CCW system, two Class 1E pressure monitoring systems will shut all the Class 3/Class N isolation valves. The operator must remote-manually open the SCW/CCW isolation valves. An interlock ensures that all non-nuclear safety portions of the CCW system have been isolated before the SCW/CCW isolation valves may be opened. Initially, cooling is supplied only to the spent fuel pool coolers; however, at the operators discretion the seal return coolers, RC pump seal heat exchangers and letdown coolers may be cooled by remote-manually opening CV-3632.

The SCW/CCW interlock is designed so that there are redundant methods of isolating each portion of non-nuclear piping from the SCW supplies to the spent fuel pool heat exchangers. Therefore, no single failure of a Class 3/Class N isolation valve to close will prevent supplying cooling water to the spent fuel pool heat exchangers.

Rupture in non-nuclear portions of CCW has been evaluated for possible flooding problems. Calculations have shown that the entire CCW loop may be emptied into either the General Services Building (GSB) or Containment without endangering any equipment required for plant safety.

LOCA

Following a LOCA, an ESFAS signal (mode 1) will automatically close the SCW/CCW cross-connect isolation valves and the Containment isolation valves, CV-3632, CV-[illegible]8 and CV-3586. These valves are indicated on the attached drawing with an "[illegible]" next to the valve. The only components which require cooling are the spent fuel pool heat exchangers. The SCW/CCW cross-connect isolation valves must be remote-manually opened as described in SSE above.

The following is a listing of components supplied by the CCW system:

A. Components supplied from ASME III, Class 3 and Seismic Category I piping

1. Spent Fuel Pool Heat Exchanger
2. Letdown Coolers
3. R.C. Pump Seal Heat Exchangers
4. Seal Return Coolers

B. Components supplied from non-qualified piping*

1. Demineralized Water Heat Exchangers
2. Degasifier Vacuum Pumps
3. Steam Generator Sample Coolers
4. Control Rod Drives
5. R.C. Bleed Packages
6. Containment and Contaminated and non-Contaminated Service Area Compressor Condensing Units Chiller Units.
7. R.C. Pump Motor and Bearing Coolers
8. Waste Gas Compressors and Recombiners

*Provided with ASME III, Class 3 piping in PSAR.

Conclusion Relative to Principal Criteria

The PSAR commits to the design of a component cooling water system whose quality group classification and seismic design conforms to Regulatory Guides 1.26 and 1.29 (PSAR Section 3.2 and 3.12). In addition, the general commitment is made that structures, components and systems necessary to ensure:

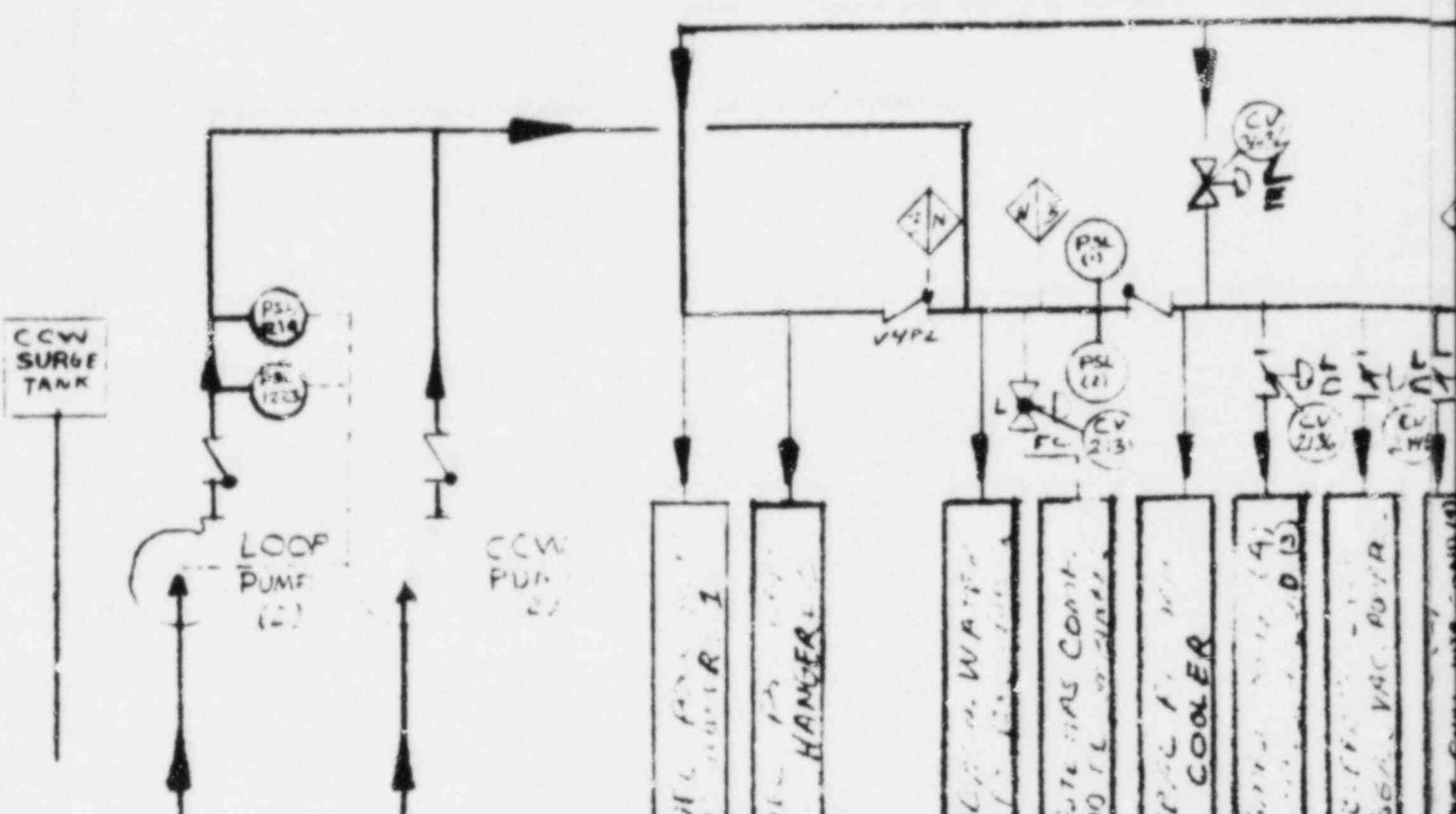
a. the integrity of the reactor coolant pressure boundary;
b. the capability to shut down the reactor and maintain it in a safe shutdown condition, or
c. the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposure of 10CFR100.,

will be maintained during and after a safe shutdown earthquake. (PSAR Subsection 3.2.1).

Finally, the commitments are made to design structures, systems and components important to safety to comply with General Design Criterion 2 and to design cooling water systems to comply with General Design Criterion 44 (PSAR Subsections 3.1.2 and 3.1.40).

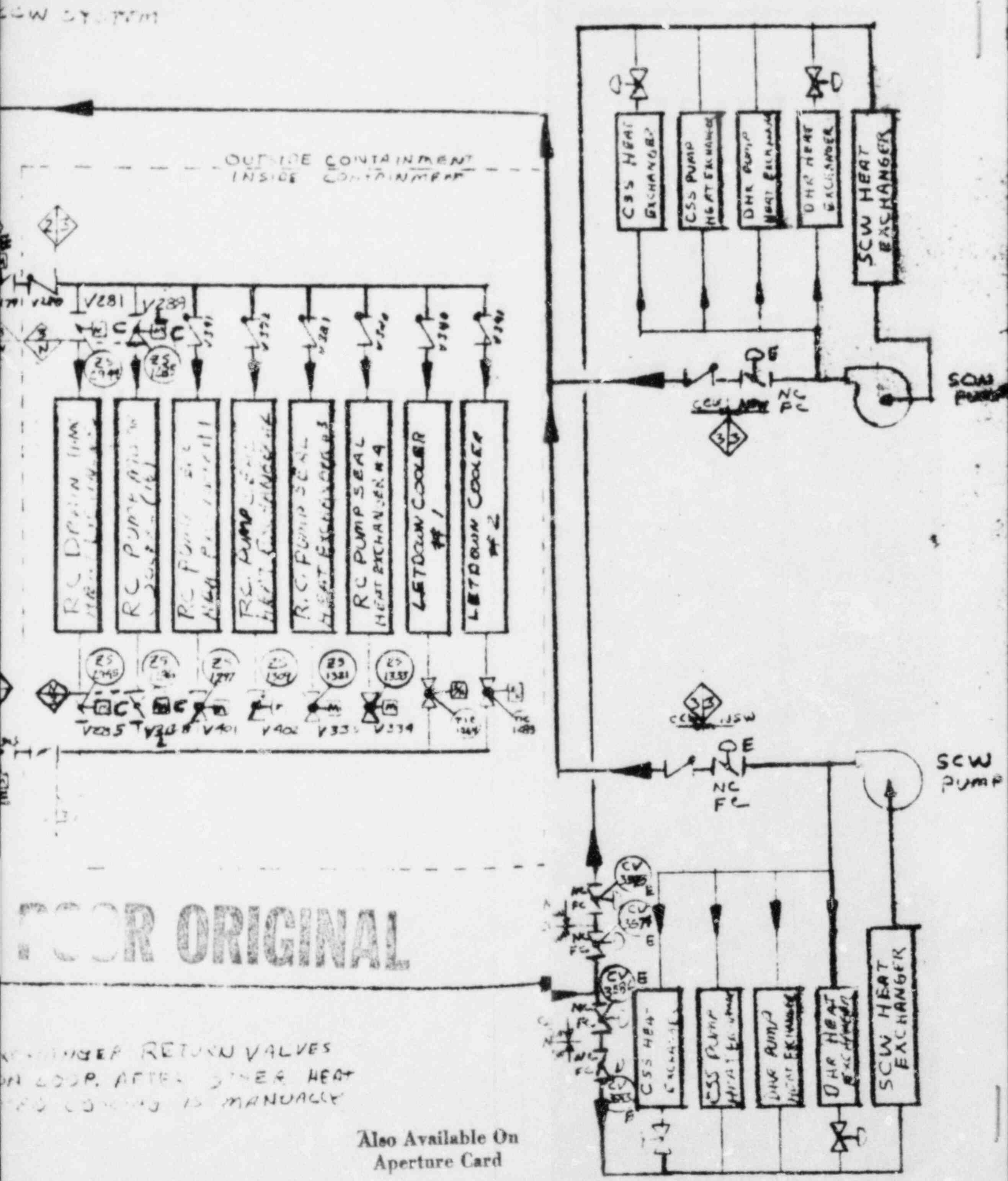
The redesigned CCWS will comply with all of these criteria. The Applicant's position is, therefore, that the redesigned CCWS does not represent a non-conservative departure from the principal architectural and engineering criteria described in the applications for construction permits and operating licenses for WNP-1/4.

FUNCTIONAL



L- AUTOMATICALLY CLOSED ON LOOP
E- AUTOMATICALLY CLOSED ON ESFAS
C- AUTOMATICALLY CLOSED ON
INDICATION OF CCW ... CONNECT ...

NOTE 1
FUEL POOL ...



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ON LOOP. AFTER OTHER HEAT
COOLING IS MANUALLY

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Attachment B

Pipe Break Design Criteria

The present Section 3.6 to the PSAR was developed to meet the requirements set forth by the NRC in PSAR question 3.39. This question stated that the Supply System must fully comply with the following references:

a. Regulatory Guide 1.46, entitled "Protection Against Pipe Whip Inside Containment."

b. Letter dated December 12, 1972, from the Deputy Director for Reactor Projects (A. Giambusso) to industry representatives on the subject, "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment."

c. An attachment to the question entitled "Pipe Whip Analysis."

We further indicated compliance to the extent practicable with the following, as required by the NRC:

d. Letter dated July 12, 1973 from the Director of Licensing (Mr. J. O'Leary) to industry representatives concerning postulated piping ruptures outside containment.

The NRC has subsequently (March 1975) issued Branch Technical Position (BTP) APCSB 3-1 entitled "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" and Branch Technical Position MEB 3-1 entitled "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment." Compliance with these BTP's in lieu of references b. through d. is authorized by the NRC in the BTP's and the choice is left to the individual applicants.

Based on a detailed review of the above references, we have decided to implement the provisions of NRC BTP MEB 3-1 and NRC BTP APCSB 3-1 dated March 1975 into WNP-1 and WNP-4 design as follows:

I. Section 3.6 of the PSAR will be modified to incorporate the provisions of recent NRC guidance on pipe break and protection outside the containment (BTP APCSB 3-1 and MEB 3-1).

II. Inside the containment, Regulatory Guide 1.46 will be followed and the following additional guidelines will be used with respect to selection of breaks:

1) Breaks shall be postulated at all intermediate locations where maximum stress ranges (for normal and upset conditions and an OBE event) exceed 0.8 $(1.2S_h + S_A)$.

2) PSAR Subsection 3.6.2.4 currently states that "where the maximum stress range in the axial direction is at least twice that in the circumferential direction, considering upset plant conditions, then only a circumferential break need be postulated." The ratio of stresses will be changed from 2.0 to 1.5 to be in conformance with the philosophy of MEB BTP 3-1.

3) Further, when postulating a minimum of two breaks regardless of stress levels between terminal ends, longitudinal breaks will not be postulated.

4) Longitudinal breaks will not be postulated at terminal ends provided the piping at the terminal ends contains no longitudinal pipe welds. (If longitudinal welds are used at the terminal ends, the requirement of (2) above will apply).

Above items have been allowed by the NRC in BTP MEB 3-1 for use outside the containment.

III. Definitions of high energy, moderate energy and normal operations contained in the PSAR Section 3.6 will be revised in accordance with the BTP APCSB 3-1 for both inside and outside the containment.

IV. Pipe break propagation criteria of the Section 3.6 will be revised to comply with the latest B&W criteria (changes to PP 3.6-2, 3 discussed below).

V. Pipe break criteria for Class N piping outside the containment will be added in accordance with BTP MEB 3-1.

The paragraph by paragraph proposed changes to the PSAR Section 3.6 are indicated below:

Page	Subsection	Changes
3.6-1	3.6	State that the design philosophy inside containment meets the requirements of Regulatory Guide 1.46 dated May 1973 and those provisions of MEB 3-1 dated March 1975 indicated in paragraph II above. State that the design philosophy outside containment meets the requirements of MEB 3-1 dated March 1975 and APCSB 3-1 dated March 1975.
3.6-1	3.6.1	Define "high energy" and "normal plant conditions" in accordance with APCSB 3-1 dated March 1975 as follows: "High energy systems include those fluid systems that during normal plant conditions, are either in operation or maintained pressurized and where either or both the maximum operating temperature exceeds 200°F, or the maximum operating pressure exceeds 275 psig. "Normal plant conditions are plant operating conditions during reactor startup, operation at power, hot standby, or reactor cooldown to cold shutdown conditions."
3.6-2,3	3.6.1	Delete redundant and not applicable break propagation criteria, by deleting all material on page 3.6-2 from paragraph d. to the end of the page, and all material on page 3.6-3 up to paragraph 3.6.2.
3.6-4	3.6.2.2	Re-write the last paragraph on page 3.6-4 to state that the break type at the postulated locations shall be determined in accordance with the provisions of BTP MEB 3-1 dated March 1975 indicated in paragraph II above.

Page	Subsection	Changes
3.6-5	3.6.2.4	Re-write paragraphs b and c discussing orientation of longitudinal and circumferential breaks to conform with Regulatory Guide 1.46 and BTP MEB 3-1 as follows: "Longitudinal breaks are parallel to the longitudinal axis of the pipe and are oriented (but not concurrently) at two diametrically opposed points on the piping circumference such that the jet reaction causes out-of-plane bending of the piping configuration." "Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially.
3.6-13	3.6.6.1	Add at the end of line 8 in the first paragraph the following statement regarding maximum allowable stresses at a containment penetration caused by a piping failure outside the containment penetration: "The maximum stress as calculated by Eq (9) in paragraph 3652, ASME Code, Section III under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping will not exceed 1.8 S_h."
3.6-14	3.6.6.1	Change locations for required breaks to comply with MEB 3-1. Add required break locations for non-nuclear class piping using the wording in MEB 3-1.
3.6-15	3.6.6.1	Add a statement after paragraph b. stating that break types postulated will be in accordance with MEB 3-1 as follows: "At each of the postulated break locations, consideration is given to the occurrence of either a longitudinal or circumferential type break. The break type at the postulated locations shall be determined in accordance with the provisions of BTP MEB 3-1.
3.6-15	3.6.6.2	State at the end of paragraph 1 that in accordance with MEB 3-1, cracks instead of breaks may be postulated in piping for fluid systems that qualify as high energy for only short operational periods, as follows: "Through wall leakage cracks instead of breaks may be postulated in the piping of those fluid systems that qualify as high-energy fluid systems for only short operational periods (about 2% of the time), but qualify as moderate-energy fluid systems for the major operational period. These systems include: Decay Heat

Page	Subsection	Changes
		Removal System, Auxiliary Feedwater System, and the Steam Generator Warm-up Circulating System."

Conclusion Relative to Principal Criteria

The PSAR commits to the design of structures, systems, and components so as to mitigate the consequences of a postulated pipe break, including blowdown jet and reaction forces such that for a postulated break:

a. the capability to shut down the reactor and maintain the reactor in a safe shutdown condition is ensured, and

b. the guideline exposures of 10CFR100 are not exceeded (PSAR Section 3.6).

The application of NRC Branch Technical Positions APCSB 3-1 and MEB 3-1 will satisfy entirely the above criteria. The Applicant's position is, therefore, that the use of these Branch Technical Positions does not represent a nonconservative departure from the principal architectural and engineering criteria described in the applications for construction permits and operating licenses for WNP-1/4.

Attachment C
Seismic Qualification of the Polar, Cask Handling and New Fuel Handling Cranes

PSAR Table 3.2-1 specifies the Containment polar crane, the cask handling crane and the new fuel handling crane to be Non-Seismic Category I cranes, but requires they be designed to prevent structural or mechanical failure, or other failure that could result in an uncontrolled lowering of the load in the event of an SSE. This requirement was considered necessary in the absence of a firm layout of safety-related components and piping to eliminate concerns of crane failures resulting in damage to safety-related components, or causing radioactive releases constituting a hazard to the public. Subsequently, a detailed review of the plant layout and equipment arrangement has been conducted to determine the consequences of such crane failure. Based on this review, the foregoing crane requirements are now considered unnecessary because the plant layout and equipment arrangement are such that the drop of any crane supported loads will not impair the safe shutdown capability of the reactor plant, or cause a radioactive release in excess of a small fraction of 10CFR100 limits. Further, analysis indicates that the possibility of any damage as a result of a seismic event with the above cranes loaded is remote due to the extremely low probability of the occurrence of a seismic event during the crane duty cycle.

Containment Polar Crane

The operational capacity (260 tons) of the polar crane is based on the reactor vessel (RV) head and service structure weighing 220 tons, and the core support assembly (CSA) weighing 180 tons; the largest normally lifted load besides these occasional lifts is about 80 tons. It is therefore concluded, that for loads other than the reactor head and core support assembly, possibility of passive failure need not be considered, since there is such a large margin between the design rated capacity and the lifted loads, even considering the seismic forces.

The evaluation of affects of load drop has therefore been limited to the dropping of loads which approach the design capacity of the crane. The core support assembly has no safety significance since at the time CSA is lifted no fuel is contained in the reactor.

RV Head Lift - During normal reactor operation the crane is parked, unloaded on the north-south axis of Containment. Any substantial use of the polar crane takes place only when the reactor cooling system is cooled down and depressurized. However, prior to this and during reactor operation, entry into the Containment may be initiated and preparation for refueling shutdown undertaken. As part of this preparation, the Containment polar crane may be checked and moved around and used to make minor lifts of up to a maximum lifted load of about one ton. The largest operational load (reactor vessel head) will be lifted twice during each refueling and carried only along the north-south axis of Containment. This load path precludes the load being carried over critical portions of the reactor coolant pressure boundary, other than the reactor vessel itself. Consequently, effects analysis requires only consideration of load drops with respect to the reactor vessel.

RV Head Drop Over the Reactor Vessel - The lift of the RV head is limited to a maximum of 5 ft. by an interlock system, which is activated when the main hoist load sensing mechanism indicates an under hook load in excess of 200 tons. The interlock system contains redundant limit switches which are calibrated to de-energize the hoist motor when the RV head is lifted to the extent necessary to clear the guide studs (height of guide studs 3' 10-1/3" - total lift 4'-5"). The hoist motor may not be energized in the upward direction until the missile shield trips the interlock by returning to its normal position over the reactor vessel. The RV head must be moved south (horizontally), clear of the reactor vessel, to permit relocation of the missile shield, thus limiting the height of the RV head over the reactor vessel to 5 feet.

The RV head must travel above the missile shield to the head laydown area. The lift height is limited, to provide a maximum of six inches clearance above the missile shield, by the redundant upper limit switches of the main hoist. The missile shield is designed to withstand the six inch drop of the RV head thereby protecting the reactor vessel.

The above limits and interlocks restrict the lifting of the RV head such that the maximum drop of the RV head onto the reactor vessel is five feet. Several analyses were performed to evaluate the RV head drop, the most pertinent of which are:

1) A 5 ft. free fall through air and drop onto the reactor vessel with no credit taken for the inertia effects of the water in the RV.

2) A 5 ft. uncontrolled lowering (considering the inertia of the hoist components) through air onto the reactor vessel with no credit taken for inertia effects of the water in the RV.

The above situations are considered conservative as the maximum drop height is actually 4'-5" and the reactor vessel will always contain water when any lifts are made. Following is a list of the assumptions/methods used in the analysis of the RV head drop.

1. It is assumed that the head sets as a dead weight dropping directly upon the upper flange of the nozzle belt assembly and the load is transmitted evenly to the nozzles and nozzle supports.

2. The potential energy of the falling weight is transferred to kinetic energy and on impact is equated to the strain energy in the system.

3. The impact load is developed using a "shock factor" defined from:

   $$P = (\text{shock factor})\ W$$

   where P = impact load, lb.
   W = falling weight, lb.

   The shock factor is determined from the elastic stiffness of the members involved, the drop height, and the falling weight.

4. Consideration is given to the inertia of the reactor vessel after impact which is similar to Case Number 2 of Roark's "Formulas for Stress and Strain", 4th Edition Page 371.

The loads resulting from these analyses which also consider elastic/plastic effects of the concrete supports are within the design limits of the reactor vessel and the RV supports.

RV Head Drop at Other Places Along the North-South Axis of the Containment - Of the other places along the north-south axis of the Containment, only a drop over the head laydown area has potential safety concerns. Underneath the 4 ft. 3 inch thick reinforced concrete floor at the head laydown area, one of the two 12" Decay Heat Removal inlet lines is routed. The line joins the core flooding nozzle and the reactor vessel through two check valves in series, one of which is located inside the shield wall. Therefore, drop of the RV head at this location will be resisted by the reinforced concrete flooring. Should, however, the flooring be postulated to fail catastrophically as a result of the RV head drop, the integrity of the DHR piping underneath the flooring would be threatened. For purposes of analysis this piping is considered to have suffered a total severence as a result of the structural failure of the floor above, and the resulting missiles. The following discussion indicates that the consequences of such a failure are acceptable from the point-of-view of continued reactor core cooling and the containment of radioactivity.

As previously mentioned, this could occur only when the reactor is shutdown, cooled down and depressurized. Therefore, the effect of this load drop is evaluated with respect to the ability to maintain the plant in a safe shutdown condition. This analysis is broken down in two phases: 1) the immediate effects and, 2) the impact of subsequent (post-accident) failures:

1) Immediate Effects:

The DHR system utilizes the Core Flooding System (CFS) nozzle to provide fl[illegible] to the reactor vessel. The CFS nozzles on the reactor vesse[illegible] located above the minimum water level required to keep the c[illegible]vered. Therefore, if a load were to fail and rupture a CFS or DHR [illegible]etween the check valves provided or if the check valves were t[illegible] the reactor vessel would neither drain to a level which would unc[illegible] core, nor preclude operation of the redundant DHR train.

2) Subsequent (Post-Accident) Failures:

a. Active Failure

The DHR system, the source of core cooling during refueling, consists of two redundant trains - each equipped with one pump and one heat exchanger; therefore, core cooling can be continued without any interruption following the loss of one loop. The two loops are cross-connected (outside Containment) to provide redundancy and operational flexibility so that a single active failure concurrent with the loss of one loop due to a load drop can be tolerated.

b. Passive Failure

Since the operating conditions of the DHR system at the time of a postulated polar crane failure (plant shutdown, cooled down and depressurized) are significantly less severe than design conditions (680 psig, 350^{0}F), it is felt that consideration of a long-term passive failure is unnecessary.

In conclusion, the consequences of any polar crane load drop are found to be acceptable with respect to continued reactor core cooling. Consequently, the polar crane is designed only to remain structurally intact and generate no missiles during a seismic event.

Spent Fuel Cask Handling and New Fuel Handling Cranes

The cask handling and the new fuel handling cranes do not carry loads directly over equipment required for safe reactor shutdown, therefore, concerns about damage resulting from a seismic event should center around radioactive releases resulting from damage to the loads dropped. Results of such calculations have been reported in the Environmental Report which consider the releases associated with the failure of fifteen fuel assemblies as a result of a cask drop. The resulting off-site doses are a small fraction of 10CFR100 limits.

The largest currently licensed spent fuel shipping cask is capable of accommodating a maximum of seven PWR spent fuel assemblies, and possible future designs a maximum of fourteen. Therefore a radioactive release accident associated with damage to a spent fuel shipping cask is acceptable in that the resulting off-site dose would be small fractions of 10CFR100 limits.

The new fuel shipping container holds only two new fuel assemblies. Failure of the new fuel assemblies as a result of a load drop from the new fuel handling cranes is not generally associated with concerns of radioactivity release. Therefore, as in the case of the polar crane, the cask handling and new fuel handling cranes are designed only to remain structurally intact and generate no missiles during a seismic event.

Conclusion Relative to Principal Criteria

The PSAR commits to the design of structures, systems, and components in accordance with Regulatory Guide 1.29 (PSAR Section 3.12). Regulatory Position 2 of this guide requires that:

"Those portions of structures, systems, or components whose continued function is not required but whose failure could reduce the functioning of any plant feature included in items 1.a. through 1.r. above to an unacceptable safety level should be designed and constructed so that the SSE would not cause such a failure."

The above analysis demonstrates that the specifying of non-Category I cranes, as described, will not reduce the functioning of the safety related structures, systems, and components described in item a) through r) of Regulatory Position 1. Therefore, it is the Applicant's position that the above described change to the seismic qualification for the cranes does not represent a non-conservative departure from the principal architectural and engineering criteria described in the applications for construction permits and operating licenses for WNP-1/4.

Attachment D

Proposed Change from a Four Channel (ABCD) Wiring and Raceway System to a Five Channel (ABCDX) System

The WNP-1/4 wiring and cable system is presently a four channel design. Channel allocations presently are as shown in Attachment D-1.

PSAR Subsection 8.3.3.1 states that in order to comply with Regulatory Guide 1.75, "All other (Balance of Plant) cables and raceways are considered as 'associated' circuits and equipment " since all four raceway systems must carry both safety related and non safety-related circuits.

The existing arrangement is illustrated in the left side of Figure 1.

We propose the implementation of a five channel system. Ch A & B would be the same as originally. Ch C & D would be the original C & D safety related items only and would be run entirely in conduit. Channel X, the fifth channel would be non-safety-related, containing no associated circuits and would be a composite of the original channel C&D non-safety related circuits including NNI and ICS, and IMS from Channel B. The arrangement is tabulated in Attachment D-2 and is illustrated on the right side of Figure 1.

It should be noted that certain housekeeping (e.g., LOOP loads) which are not strictly safety related and which are desirable during ESFAS conditions are presently, and will remain, installed as associated circuits in channels A and B. Other LOOP loads are connected to the Class 1E power sources by isolation devices and will be run in Channel X.

The proposed arrangement eliminates a large amount of "associated" circuits which were such only because they were forced to share the four channel raceway system with the safety related circuits.

The following benefits will result:

1. Considerable reduction in the amount of isolation relays and other electrical circuit isolation arrangements required to meet Regulatory Guide 1.75, especially in the areas of redundant non-safety loads and instrumentation and control of separately channeled BOP loads from common control room panels, (since they will all, now, be one channel.) This facilitates interlocking of such loads. (Note that channeling and isolation considerations for the four safety grade channels, in regard to control room panels, still remain).

2. Simplification of the design of the cable spreading area.

Conclusions Relative to Principal Criteria

The above described addition of a non-1E channel to the on-site power system complies entirely with current PSAR commitments relative to Regulatory Guide 1.75. The Applicant's position is, therefore, that the addition of the non-1E channel to the on-site power system does not represent a non-conservative departure from the principal architectural and engineering criteria described in the applications for construction permits and operating licenses for WNP-1/4.

Attachment D-1

Channel Contents Four Channel System

Channel A - Class 1E circuits for Safety Related Loads
- Associated Circuits for LOOP loads

RPS A
NIS A (Conduit only)
ESFAS Analog A
ESFAS Digital A
ECI X
Safety Train A (Decay Heat Removal ((Low Pressure Injection)) pumps, Make-up ((High Pressure Injection)) pumps, Containment Spray Pumps, Emergency Shutdown Water Pumps, etc.)
LOOP loads A, including LOOP loads not required during ESFAS

Channel B - Class 1E circuits for Safety Related Loads
- Associated Circuits for LOOP loads

RPS B
NIS B (Conduit only)
ESFAS Digital B
ECI Y
Safety Train B
LOOP loads B, including LOOP loads not required during ESFAS
IMS (conduit)

Channel C - Class 1E Safety Related Loads
- Balance of Plant Circuits (partial)

RPS C
NIS C (Conduit only)
ESFAS Analog C
ICS
1055/BN Computer System
BOP Power Control & Instrumentation

Channel D - Class 1E Safety Related Loads
- Balance of Plant Circuits (partial)

RPS D
NIS D (Conduit only)
ESFAS Analog B
NNI
Turbine Generator Controls
BOP Power Control & Instrumentation

Attachment D-2

Channel Contents Five Channel System

Channel A - Class 1E Circuits for Safety Related Loads
- Associated Circuits for Loads required for LOOP

RPS A
NIS A (Conduit only)
ESFAS Analog A
ESFAS Digital A
ECI-X
Safety Train A (DHR pumps, Make-up pumps, CSS pumps, etc.)
LOOP loads A

Channel B - Class 1E Circuits for Safety Related Loads
- Associated circuits for loads required for LOOP

RPS A
NIS B (Conduit only)
ESFAS Digital B
ECI-Y
Safety Train B
LOOP loads B

Channel C - (Conduit only) - Class 1E Safety Related Only

RPS C
NIS C
ESFAS Analog C

Channel D - (Conduit only) - Class 1E Safety Related Only

RPS D
NIS D
ESFAS Analog B

Channel X - No Class 1E Circuits

NNI
IMS
ICS
All BOP 15KV/5KV/480V/120V AC, 125V DC and instrumentation circuits
1055/BN Computer
T/G Controls
LOOP loads powered from Class 1E systems via Isolation Devices

PRESENT DESIGN

CHANNEL A

cable tray system

Class 1E circuits for safety related loads. Associated circuits for loads required for LOOP.

- RPS "A"
- ESFAS Analog "A"
- ESFAS Digital "A"
- ECI "X"
- Safety Train "A" (DHR Pumps, Makeup Pumps, CSS Pumps, etc.)
- LOOP Loads

conduit system

- NIS A (Conduit only)

CHANNEL B

cable tray system

Class 1E circuits for safety related loads. Associated circuits for loads required for LOOP

- RPS "B"
- ESFAS Digital "B"
- ECI "Y"
- Safety Train "B"
- LOOP Loads

conduit system

- IMS
- NIS B (Conduit only)

CHANNEL C

cable tray system

- Power and control for a portion of plant non-safety related items
- ESFAS analog C
- ICS
- RPS
- Computer Inputs

conduit system

- NIS C (Conduit only)

CHANNEL D

cable tray system

- Power and control for remainder of the balance of plant
- ESFAS Analog B
- NNI
- RPS "D"
- T/G controls

conduit system

- NIS D (Conduit only)

PROPOSED DESIGN

CHANNEL A

cable tray system

Class 1E circuits for Safety Related loads. Associated circuits for load required for LOOP

- RPS "A"
- ESFAS Analog "A"
- ECI "X"
- Safety Train "A"
- (DHRpumps, Makeup Pumps, CSS Pumps, etc.)
- LOOP Loads

conduit system

- NIS A (Conduit only)

CHANNEL B

cable tray system

Class 1E Circuits for Safety Related loads. Associated circuits for loads required for LOOP.

- RPS
- ESFAS Digital "B"
- ECI "Y"
- Safety Train "B"
- LOOP Loads

conduit system

- NIS B (Conduit only)

CHANNEL C

conduit system

NIS "C"
ESFAS Analog C
RPS "C"

cable tray system deleted

CHANNEL D

conduit system

NIS "D"
ESFAS Analog B
RPS "D"

cable tray system deleted

CHANNEL X

former C & D cable trays redesignated as X

All BOP 15KV/5KV/480V/120V A-C and 125 DC Circuits
NNI
IMS Computer Inputs
ICS T/G Controls
LOOP Loads (Powered from 1E System via Isolation Devices

Comparison of Four Channel & Five Channel Wiring Raceway Systems WNP 1&4 Figure 1