JUN 2 2 1981

Docket Nos.: STN 50-528/529/530

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. E. E. Van Brunt, Jr.
Vice President - Nuclear Projects
Arizona Public Service Company
P. O. Box 21666
Phoenix, Arizona 85036

Dear Mr. Van Brunt:

SUBJECT: PALO VERDE NUCLEAR GENERATING STATION, UNITS 1, 2 AND 3 - REQUEST FOR ADDITIONAL INFORMATION

In order to complete our review of the Palo Verde FSAR, we find that additional information is required. The enclosed request for information covers those review areas for which the Reactor Systems Branch has primary responsibility.

Many of these questions are within the CESSAR System 80 scope of review and, therefore, APS is not required to respond to those questions.

Within 7 days of receipt of this letter, please identify those questions which will be responded to on the Palo Verde docket and provide a schedule for responding to those items. The remaining questions will be addressed to combustion engineering and will be responded to on the CESSAR docket.

Please contact us if you have any questions on this matter.

Sincerely,

Original signed by
Robert L. Tedesco

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
Request for Information

cc: w/enclosure
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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PALO VERDE QUESTIONS

440.1 (5.2.2) A description of the design features which will be used to mitigate the consequences of overpressurization events while operating at low temperatures is not provided in the CESSAR System 80 FSAR. Provide a description of the features which will be provided on the CESSAR System 80. Specific design criteria regarding overpressurization protection while operating at low temperatures are as follows:

1. Operator Action: No credit can be taken for operator action for 10 minutes after the operator is aware of a transient.

2. Single Failure: The system must be designed to relieve the pressure transient given a single failure in addition to the failure that initiated the pressure transient.

3. Testability: The system must be testable on a periodic basis consistent with the system's employment.

4. Seismic and IEEE 279 Criteria: Ideally, the system should meet seismic Category 1 and IEEE 279 criteria. The basic objective is that the system should not be vulnerable to a common failure that would both initiate a pressure transient and disable the overpressure mitigating system. Such events as loss of instrument air and loss of offsite power must be considered.

An alarm must be provided to monitor the position of the pressurizer relief valve isolation valves to assure that the overpressure mitigating system is properly aligned for shutdown conditions.

In demonstrating that the mitigation system meets these criteria, the applicant should include the following information in his submittal:

1. Identify and justify the most limiting pressure transients caused by mass input and heat input.

2. Show that overpressure protection is provided (do not violate Appendix G limits) over the range of conditions applicable to shutdown/heatup operation.

3. Identify and justify that the equipment will meet pertinent parameters assumed in the analyses (e.g., valve opening times, signal delay, valve capacity).

4. Provide a description of the system including relevant P&I drawings.

5. Discuss how the system meets the criteria.

6. Discuss all administrative controls required to implement the protection system.

440.2 (5.2.2) Provide details of your proposed preoperational and initial startup test program to show that they are consistent with the requirements of Regulatory Guide 1.68.

440.3 (5.2.2) Check valves in the discharge side of the high pressure safety injection, low pressure safety injection, RHR, and charging systems perform an isolation function in that they protect low pressure systems from full reactor pressure. The staff will require that these check valves be classified ASME IWV-2000 category AC, with the leak testing for this class of valve being performed to code specifications. It should be

noted that a testing program which simply draws a suction on the low pressure side of the outermost check valves will not be acceptable. This only verifies that one of the series check valves is fulfilling an isolation function. The necessary frequency will be that specified in the ASME Code, except in cases where only one or two check valves separate high to low pressure systems. In these cases, leak testing will be performed at each refueling after the valves have been exercised. Identify all check valves which should be classified Category AC as per the position discussed above. Verify that you have the necessary test lines to leak test each valve. Provide the leak detection criteria that will be in the Technical Specifications.

440.4 (5A) On page 5A-2, it is indicated that a negative Doppler coefficient of $-.8 \times 10.^{-5}$ $\Delta K/K/F$ is assumed in the bounding overpressure transient (loss of load). It is our position that overpressure protection of system be demonstrated without taken credit for either doppler or moderator temperature reactivity feedback (SRP 5.2.2, Section III.6). Reanalyze the bounding overpressure transient without credit for doppler feedback, demonstrating that primary system pressure does not exceed 110% of the design pressure.

440.5 (5A) On page 5A-1, it is indicated that the worst case transient, loss of load, in conjunction with a delayed reactor trip, is the design basis for the primary safety valves. It is our position that the high pressure reactor trip or second safety grade trip signal, whichever occurs later, should be used for sizing the primary system safety valves. Confirm that the CESSAR System 80 safety valves are sized sufficiently to accommodate a reactor trip on the second safety grade trip signal.

440.6 Palo Verde must have the capability to take the plant from full

(5.4.7) power to a cold shutdown using only safety grade equipment, per the requirements of BTP RSB 5-1. Address your compliance with all provisions of that position and respond to the detailed questions below.

1. Describe the sequence for achieving a cold shutdown condition within 36 hours, assuming the most limiting single failure with only onsite power availability. Identify all manual actions inside or outside containment that must be performed and discuss the capability of remaining at hot standby until manual actions (or repairs) can be performed.

   a. If the steam generator dump valves, operators, air and power supplies are not safety grade, justify how you would cool down the primary system in the event of loss of offsite power and an SSE.

   b. Describe the sequence for depressurizing the primary system using only safety-grade systems, assuming a single failure. Identify all manual actions inside or outside containment that must be performed.

   c. Discuss the boration capability using only safety-grade systems, assuming a single failure. Identify all manual actions inside or outside containment that must be performed. If the proposed boration method utilizes the charging pumps (assuming a letdown line failure is proposed), provide an evaluation of this approach with regard to concentration of boron source and liquid volume in primary system.

2. Discuss the provisions for collection and containment of RHR pressure relief valve discharge.

3. Describe tests which will demonstrate adequate mixing of the added borated water and cooldown under natural circulation conditions with and without a single failure of a steam generator atmospheric dump valve. Specific procedures for plant cooldown under natural circulation conditions must be available to the operator. Summarize these procedures.

4. Discuss the availability of the seismic Category I auxiliary feedwater supply for at least 4 hours at hot shutdown plus cooldown to the RHR system cut-in based on longest time for the availability of only onsite or only offsite power and assuming a single failure. If this cannot be achieved, discuss the availability of an adequate alternate seismic Category I water source.

5. What provisions in natural circulation cooldown methods have been made to account for possible upper head void formation.

440.7 (5.4.7) Provide detailed information on the sizing criteria used to determine the relief capacity of the SDCS suction line pressure relief valves.

Did the version of the ASME code that the SDCS relief valves were sized to require establishing liquid or two-phase relief capacity wth testing? If so, describe in detail the test program and results. If the liquid or two-phase relief capacity was not established by test, show that the difference between the rated and maximum required relief capacity is more

than sufficient to bound liquid and two-phase relief rate uncertainties.

Provide details on the alarms and indications which would inform the operators that a SDC suction line isolation valve has closed while the plant is in shutdown cooling. Is there any common failure which would result in both valves being closed while in shutdown cooling.

When LPSI pump mini flow isolation valves are closed during shutdown cooling, what would prevent pump damage if a pressure transient were to occur which caused RCS pressure to exceed LPSI deadhead pressure.

When the plant is in the SDCS mode, is there any single failure which could cause the suction of both SDC pumps to be switched from the hot leg piping to the dry sumps?

440.8 (5.4.7) Provide the following information related to pipe breaks or leaks in high or moderate energy lines outside containment associated with the RHR system when the plant is in a shutdown cooling mode:

1. Determine the maximum discharge rate from a pipe break in the systems outside containment used to maintain core cooling.

2. Determine the time available for recovery based on these discharge rates and their effect on core cooling.

3. Describe the alarms available to alert the operator to the event, the recovery procedures to be utilized by the operator, and the time available for operator action.

A single failure criterion consistent with Standard Review Plan 3.6.1 and Branch Technical Position APCSB 3-1 should be applied in the evaluation of the recovery procedures utilized.

440.9 (5.4.7) Indicate whether there are any systems or components needed for shutdown cooling which are de-energized or have power locked out during plant operation. If so, indicate what actions have to be taken to restore operability to the components or systems.

It is the staff's position that all operator actions necessary to take the plant from normal operation to cold shutdown (SDCS entry) should be performed from the control room. If the present design does not meet this position, please commit to revise it accordingly.

440.10 (5.4.7) Provide additional information regarding the power sources supplied to the SDCS isolation valves. The staff's position is that a single failure of a power supply will not prevent isolation of the SDCS when RCS pressure exceeds its design pressure. Additionally loss of a single power supply cannot result in the inability to initiate at least one 100 percent shutdown cooling train.

440.11 (6.3) Discuss the provisions and precautions for assuring proper system filling and venting of ECCS to minimize the potential for water hammer and air binding. Address piping and pump casing venting provisions and surveillance frequencies.

440.12 (6.3.3) Section 6.3.3.2.2 states that the worst single failure for the large break LOCA is the failure of one of the low pressure pumps to start which will result in a minimum amount of safety injection water available to the core. Explain why the single failure of a diesel generator, which results in loss of one

HPSI train and one LPSI train, is not the worst single failure for the large break LOCA with respect to the amount of safety injection water available to the core in post LOCA operation.

440.13 (6.3) Identify all ECCS valves that are required to have power locked out and confirm they are included under the appropriate Technical Specifications, with surveillance requirements listed.

440.14 (6.3) Consideration should be given to the possibility that local manual valves (handwheel), could go undetected in the wrong position until a postulated accident occurs. Appropriate administrative controls or valve position indication are examples of methods to be considered to minimize this possibility. Provide a list of all critical manual valves and address the actions that will be implemented to assure all critical valves are properly positioned.

Identify all manual valves which have locking provisions.

It is our position that limit switches which enable valve position to be indicated in the control room should be installed on all manually operated and normally locked ECCS valves.

In addition a recent event (Docket 50-320, LER 78-20/3L, 4/21/78) has brought to our attention that the automatic operation of some motor operated valves can be disabled when the manual handwheel pins are engaged. Identify all critical motor operated valves associated with the CESSAR 80 design that have this design feature and describe the controls and procedures utilized to prevent the inadvertent disablement of the automatic operation of these valves.

440.15 (6.3) Identify the plant operating conditions under which certain automatic safety injection signals are blocked to preclude unwanted actuation of these systems. Describe the alarms available to alert the operator to a failure in the primary or secondary system during this phase of operation and the time available to mitigate the consequences of such an accident.

440.16 (6.3) The information in the CESSAR 80 FSAR regarding post-LOCA passive failures is not complete. It is the Reactor Systems Branch position that detection and alarms be provided to alert the operator to passive ECCS failures during long-term cooling which allow sufficient time to identify and isolate the faulted ECCS line. The leak detection system should meet the following requirements:

1. Identification and justification of maximum leak rate should be provided.

2. Maximum allowable time for operator action should be provided and justified.

3. Demonstration should be provided that the leak detection system will be sensitive enough to initiate (by alarm) operator action, permit identification of the faulted line, and isolation of the line prior to the leak creating undesirable consequences such as flooding of redundant equipment or excessive radioactive fluid. The minimum time to be considered is 30 minutes.

4. It should be shown that the leak detection system can identify the faulted ECCS train and that the leak is isolatable.

5. The leak detection system must meet the following standards:

   a. Control Room Alarm

   b. IEEE 279-1971, except single failure requirements

440.17 (6.3) The acceptance criteria in the Standard Review Plan for Section 6.3 states the ECCS should retain its capability to cool the core in the event of a single active or passive failure during the long-term recirculation cooling phase following an accident. Demonstrate that CESSAR 80 ECCS design has this capability.

440.18 (6.3) A reported event has raised a question related to the conservatism of NPSH calculations with respect to whether the absolute minimum available NPSH has been considered. In the past, the required NPSH has been taken by the staff as a fixed number supplied through the applicant by either the architect engineer or the pump manufacturer. Since a number of methods exist and the method used can affect the suitability or unsuitability of a particular pump, it is requested that the basis on which the required NPSH was determined be branded (i.e., test, Hydraulic Institute Standards) for all the ECCS pumps and the estimated NPSH variability between similar pumps including the testing inaccuracies be provided.

440.19 (6.3) Provide the basis for ECCS lag times. Are these times calculated or verified by test. If calculated, are they verified during preoperational tests, and periodically reverified?

440.20 (6.3) Provide in the Technical Specifications, (1) the range of nitrogen cover gas pressure for the SIT, and (2) the ECCS pump discharge pressures.

440.21 (6.3) Provide a time reference for each action in the sequence of action included in the changeover from injection to recirculation. Indicate the time required to complete each action and what other duties the operator would be responsible for at this point in the accident. How much time does the operator have to assure that the system is realigned to the recirculation mode before RWST water is exhausted if the RWSP isolation valves are not closed? Consider the required pump NPSH in your response.

If the operator fails to close the RWST isolation valves, demonstrate that the HPSI will continue to adequately cool the core during the recirculation mode.

440.22 (6.3) Recently, another plant has indicated that a design error existed in the sizing of their RWST. This error was discovered during a design review of the net positive suction head requirements for the containment spray and residual heat removal pumps. The review showed that there did not appear to be sufficient water in the RWST to complete the transfer of pump suctions from the tank to the containment sump, before the tank was drained and ECCS pump damage occurred.

It was reported that in addition to the water volume required for injection following a LOCA, an additional volume of water is required in the RWST to account for:

1. <u>Instrument error</u> in RWST level measurements.

2. Working allowance to assure that normal tank level is sufficiently above the minimum allowable level to assure satisfaction of technical specifications.

3. Transfer allowance so that sufficient water volume is available to supply safety pumps during the time needed to complete the transfer process from injection to recirculation.

4. Single failure of the ECCS system which would result in larger volumes of water being needed for the transfer process. In this situation, the worst single failure appears to be failure of a single ECCS train to realign to the containment sump upon low RWST signal. This result in the continuation of large RWST outflows and reduces the time available for the manual recirculation switchover, before the tank is drawn dry and the operating ECCS pumps are damaged.

5. Unusable volume in the tank is present because once the tank suction pipes are reached, the pumps lose suction and any remaining water is unusable. Additionally, some amount of water above the suction pipes may also be unusable due to NPSH considerations and vortexing tendencies with the tank.

Preliminary indications are that approximately an additional 100,000 gallons of RWST capacity were needed to account for these considerations. It is our understanding that the design parameters for instrument error, transfer allowance and single failure have changed since the original sizing of the tank.

In light of the above information, discuss the adequacy of your Refueling Water Storage Tank. Provide a discussion of

the necessary water volumes to accommodate each of the five considerations indicated above. Justify your choice of volumes necessary to account for each consideration. Provide drawings of your RWST, showing placement and elevation of tank suction lines, and level sensors. Also, provide operator switchover procedures for aligning to the recirculation mode, with estimates of the time required for each action.

440.23 (6.3) Provide a discussion on specific methods of detecting, alarming and isolating passive ECCS failures during long-term cooling to include valve leakage. Show that there is sufficient time for the operator to take corrective action and maintain an acceptable water inventory for recirculation. Justify the basis for the assumed leak rates. Describe how the contaminated water would be handled if one ECCS train must continue to operate with a leak.

440.24 (6.3) Assume a maximum passive failure flow rate of 50 GPM in each ECCS pump room and discuss the effects of the passive failure to each ECCS pump operation and demonstrate that adequate protection is provided for ECCS pumps from possible flooding.

440.25 (6.3) In the event of early manual reset of the safety injection actuation signal (SIAS) followed by a loss of offsite power during the injection phase, operator action may be required to reposition ECCS valves and restart some pumps. The staff requires that operating procedures specify SIAS manual reset not be permitted for a minimum of 10 minutes after a LOCA. Provide the administrative procedures to ensure correct load application to the diesel generators in the event of loss of offsite power following an SIAS reset.

440.26 (6.3) Describe the instrumentation for level indication in the containment emergency sump. Also, provide detailed design

drawings of the containment emergency sump including the design provisions which preclude the formation of air entraining vortices during recirculation cooling. Confirm that the containment emergency sump design meets the requirements of Regulatory Guide 1.82.

440.27 (6.3) Recent plant experience has identified a potential problem regarding the operability of the pumps used for long-term cooling (normal and post-LOCA) for the time period required to fulfill that function. Provide the pump design lifetime (including operational testing) and compare to the continuous pump operational time required during the short- and long-term of a LOCA. Submit information in the form of tests or operating experience to verify that these pumps will satisfy long-term requirements.

440.28 (6.3) Describe the means provided for ECCS pump protection including instrumentation and alarms available to indicate degradation of ECCS pump performance. Our position is that suitable means should be provided to alert the operator to possible degradation of ECCS pump performance. All instrumentation associated with monitoring the ECCS pump performance should be operable without offsite power, and should be able to detect conditions of low discharge flow.

440.29 (6.3) Describe the instrumentation available for monitoring ECCS performance during post-LOCA operation (injection mode and recirculation mode).

440.30 (6.3) Provide a commitment that Palo Verde will perform preoperational and startup tests to meet the requirements of Regulatory Guide 1.68 and 1.79.

440.31 Provide a commitment that Palo Verde will perform tests of ECCS
(6.3) as installed to confirm that the actual ECCS flow rates are greater than the values assumed in the LOCA analyses.

440.32 Expand Table 15.0-6, the list of single failure considered in
(15.0) transient and accident analyses, to include the following:

1. one primary safety valve stuck closed

2. one secondary safety valve fail to open or fail to close

3. loss of offsite power

4. failure of one diesel to operate (for the events with loss of offsite power being treated as a consequential result of the event).

5. failure to achieve fast transfer

440.33 For all analyses of transients with concurrent single failures,
(15.0) provide a reference to the sensitivity study which shows that the failure selected is the worst case single failure.

440.34 Confirm that during the preoperational or startup test phase you
(15.0) intend to verify the valve discharge rates and response times (such as opening and closing times for main feedwater, auxiliary feedwater, turbine and main steam isolation valves, and steam generator and pressurizer relief and safety valves) to show that they have been conservatively modeled in the Chapter 15.0 analyses.

440.35 (15.0) The method that you have used for calculating the amount of failed fuel after an accident has not been approved. It is our position that fuel failures be recalculated using the criteria that any fuel rod which has a CE-1 DNBR less than the minimum DNBR value determined in Section 4.4 fails. Radiological consequences should be calculated accordingly.

440.36 (15.0) Verify that for each transient analyzed in Chapter 15, if operator action is not discussed then no operator action is required. In particular, consider events in which the ECCS is actuated or RCP trip would be required based on present procedures.

440.37 (15.0) For each accident, discuss non-safety grade equipment which was assumed to operate and could result in the transient becoming more severe or verify that no non-safety grade equipment operating would produce a more severe transient. For example, the pressurizer heaters being energized for a transient resulting in high RCS pressures could tend to worsen the effects of the transient. Likewise, pressurizer spray could be detrimental for a transient resulting in low RCS pressure.

440.38 (15.0) Plant operators are instructed to trip the reactor coolant pumps (RCPs) during ECCS actuation. For a steam line break, tripping of the RCPs at varying times into the transient has not been addressed. Demonstrate, by analysis or otherwise, that the consequences of tripping the RCPs during a steam line break transient are bounded by the analyses already performed.

440.39 (15.0) One of the key parameters in LOCA analyses is peak clad temperature. For non-LOCA transients, minimum DNBR (departure from nucleate boiling ratio) is of primary importance. For those transients analyzed in Section 15 of the FSAR, provide graphical output of the DNBR as a function of time.

440.40 (15.0) As part of the CESEC review, the NRC intends to perform audit evaluations of feedwater line breaks, steam line breaks, and large- and small-break LOCAs (as part of the FSAR and TMI Action Plan Item II.K.3.30 and II.K.3.31 reviews). In order to perform these audits, we require the following data, as outlined in the "PWR Information Request Package."

440.41 (15.0) The current CESEC model does not properly account for steam formation in the reactor vessel. Therefore, for all events in which (a) the pressurizer is calculated to drain into the hot leg, or (b) the system pressure drops to the saturation pressure of the hottest fluid in the system during normal operation, we require the applicant to reanalyze these events with an acceptable model or otherwise justify the acceptability of Palo Verde Chapter 15 analyses conclusions performed with CESEC.

440.42 (15B) Figure 15B-19 shows the primary system pressure exceeding 110% of the design pressure. This figure also indicates a substantial pressure differential between the pressurizer and reactor vessel. The standard review plans typically limit the pressurization of the RCS to 110% of the design pressure. However, the ASME pressure vessel code permits exceeding the 110% limit to approximately 120% for very low probability events. The NRC will accept the limiting pressurization transient (i.e., feedwater line break) as calculated for System 80 if we can be assured that the analysis performed is conservative and that a small break in the feedwater line is a very low probability event.

As such, we request the following information be provided:

(1) Verification of CESEC to predict pressurization transients. This should include the developed pressure differential across the pressurizer surge line.

(2) Demonstrate that the probability of a small break in the feedwater system is not significantly more probable than the large break. Include the consideration of ancilliary line breaks.

(3) Section 15B.3 references a sensitivity study for RCS overpressurization transient to plant initial conditions. Provide the results to this study in graphical form. Specifically, include DNBR and pressure as a function of time.

(4) It is expected that increasing the break area for a feedwater line break would increase the degree of primary system pressurization. A larger break area should result in an earlier loss of heat sink and corresponding higher decay heat for system pressurization. Figure 15B-1 indicates that the limiting feedwater line break is not a doubleended guillotine break (1.4 ft^2), but a 0.2 ft^2 break. Provide greater details as to why this occurs. Is this behavior considered realistic or a consequence of a modeling assumption? Provide additional graphical explanations, including heat transfer coefficient, heat flow, secondary side inventory, all secondary side flow rates, and any additional data required to demonstrate the reasons for the 0.2 ft^2 break being the limiting break size.

(5) Figure 15B-10 provides the relationship between the maximum RCS pressure to initial steam generator inventory. Provide additional information which explains in detail

functional behavior of this curve. Provide the RCS pressure curves for the cases of initial SG inventory of 95,000 and 170,000 lbm. Describe the SG heat transfer occurring throughout these events.

Page 15B-5 states: "...the initial RCS pressure can be adjusted to provide simultaneous reactor trip signals from high pressurizer pressure and low water level in the intact steam generator and hence the plateau of maximum RCS pressure." Provide greater details of the analyses and assumptions made in order to achieve coincident trip signals from the pressurizer and SG.

(6) For Figure 15B-11 (and page 15B-6), how does raising the degree of feedwater subcooling increase the maximum RCS pressure? It would appear that raising the degree of subcooling would result in a larger heat sink, and, therefore, a lower peak pressure.

(7) What decay heat model does CESEC use? Does this model assume infinite irradiation?

(8) Provide details of the core and steam generator heat transfer models used in CESEC.

(9) Utilizing a one-node representation of the steam generator secondary side, how is the low liquid level trip analyzed?

(10) Provide verification of the CESEC <u>pressurizer</u> model for pressurization transients (resulting in the opening of a safety valve or PORV) with data from experiments and operating plant transients. Of interest is level and pressure as a function of time. Document the assumptions made in analyzing these tests.

(11) Document the sensitivity of a feedwater line break with and without loss of offsite power.

440.43 (15B) For the feedwater line break analysis, provide the pressurizer liquid and mixture level as a function of time.

Provide detailed plots for the following parameters during the initial 50 seconds of the transient:

1. Pressurizer Pressure

2. Surge line flow

3. Pressurizer mixture level

4. Pressurizer Safety Valve flow and quality.

440.44 We requie additional information regarding the steam generator behavior during a feedwater line break. Provide the steam generator secondary side coolant inventory, mixture level, heat transfer coefficients, energy removed by each steam generator (Btu) and secondary side flow as a function of time.

It is our understanding that the limiting heat transfer modeling technique utilized in CESEC assumes an approximately constant heat transfer coefficient between the primary and secondary systems until all the liquid mass in the secondary system is depleted (i.e., $\Delta M = 0$). It is not clear why the limiting modeling technique was not the case where the heat transfer was degraded as the secondary side inventory began uncovering the tubes. Please explain.

Discuss differences in the steam generator secondary heat transfer modeling between a feedwater line break and a steam line break.

440.45 (15.1.3.2.2) The stuck-open atmospheric dump valve analysis assumed operator action to scram the core 1200 seconds into the transient. Justify the time of manual action. Provide details of the plant symptoms which will alert the operator of the stuck-open dump valve. When will the plant automatically scram without operator invervention? Discuss the failures assumed in the analysis.

Question 440.41 addresses concerns with the capability of the CESEC code to properly account for primary system voiding. Address the concerns of this position as they related to your analysis of the stuck-open atmospheric dump valve event.

Provide graphical output of the mass flow rate exiting the dump valve as a function ot time.

When analyzing a stuck-open dump valve, operator action was required to isolate the feedwater from the affected steam generator. Justify the conservatism of time for operator action assumed in the analysis. What signals do the operators receive signifying that the feedwater should be isolated? When assuming tech-specs limits for the steam generator tube leakage, describe how CESEC accounts for the primary to secondary mass depletion. In the analysis, the primary system was initialized to design operating conditions. Address the conservatism of this assumption when compared to off-nominal tech-specs limits and hot standby conditions.

440.46 (15.0) Accidents resulting in containment isolation also isolate the component cooling water to the reactor coolant pumps. This

can potentially lead to RCP seal damage which may result in a LOCA. Address the time available for the operators to restore the coolant to the seals. Has consideration been given to not isolating component cooling water to the RCP seals on containment isolation? If pump seal integrity cannot be maintained, evaluate the consequential failure of the pump seals for the limiting accident.

440.47 (15.1.4.2) Section 15.1.4.2 addresses small steam line breaks outside containment (SSLBOC). The following questions relate to this section:

(1) Justify why a SSLBOC is limited to 11.5% of full power turbine flow.

(2) Update Table 15.1.4.2-1 to include Safety Injection Tank (SIT) initiation time. Also, provide SIT and HPI flow as function of time.

(3) During a small steam line break, the reactor core initially responds to a load demand. What break size results in the highest power excursion? For the limiting break size, provide graphical output of the system pressure, core power, and DNBR as a function of time.

(4) Explain why the liquid mass within the broken steam generator increases after 1080 seconds. Isn't the steam generator isolated? If not, why not?

(5) Why was the open dump valve accident (Section 15.1.3) analyzed at full power and the small steam line break (Section 15.1.4) analyzed at zero power? Assuming a tech-spec steam generator tube leakage of 1 gpm for both analyses, why wasn't the resulting dosage the same?

(6) What was the single failure assumed for the small steam line break? Justify the single failure selection as resulting in the limiting conditions.

(7) Provide graphical output of the ECC flows as a function of time and indicate when boron began to penetrate the primary system. How is the time to boron injection derived?

(8) Address the consequence of loss of AC power during the transients analyzed.

(9) Question 440.41 addresses concerns with the capability of the CESEC code to properly account for primary system voiding. Address the concerns of this question as they relate to your analysis of small steam line break outside of containment.

(10) Provide diagrams of the reheater offlines (include dimensions, loss coefficients, interconnections between the steam lines). This data should be sufficiently detailed to enable the NRC to conduct an audit of a steam line break coincident with a failure of an MSIV to close. Provide results for this accident (i.e., system pressure, pressurizer level, DNBR ratios, ECCS flows, steam generator flows, etc.) assuming with and without loss-of-offsite power. Address the consequence of losing offsite power during the steam line break.

(11) Analysis of an inadvertent opening of a turbine bypass valve has not been provided. For this accident, will the DNBR fall below 1.19 as it did for Waterford? If not, discuss the differences between the plants which cause the DNBR limit to be exceeded for one plant and not the

other. If the DNBR limit is exceeded, provide a detailed analysis for this event.

Provide a list of all accidents (excluding primary system LOCAs) which result in a DNBR less than 1.19.

(12) Compare the steam flow model utilized in CESEC with the Moody slip flow model.

440.48 (15.0) Provide a list of transients which result in opening of the pressurizer safety valves.

440.49 (15.0) The staff has been informed that the CESEC-III computer program is best suited to analyze transients which void the upper head of the reactor vessel. As such, we request that the following information be provided:

(1) Documentation of the CESEC-III code. As part of the documentation, address the differences between the different versions of CESEC (I, II, and III).

(2) Provide comparative analyses with the different versions of the CESEC programs (used for licensing) to demonstrate the adequacy of previous analyses.

(3) Provide verification of CESEC-III against plant and experimental data for pressurization and depressurization transients (such as the ANO-2 experiments and the St. Lucie I cooldown experience).

(4) For those transients which result in primary system voiding, provide graphical output of the upper head mixture level as a function of time. Discuss operator

actions/guidelines for detecting and mitigating primary system void formation.

(5) Show, by analysis or otherwise, that the allowable cooling rate (for cold shutdown conditions) will not result in primary system voiding.

440.50 (15.0) Do all CE steam generator designs incorporate a flow restrictor in the steam generator outlet nozzles?

440.51 (15C.3.1) Section 15C.3.1.3.3 is confusing. Provide greater detail of the reactor vessel mixing model. How do the various versions of CESEC evaluate assymetric temperatures between the loops during a FWLB and a SLB (assuming with and without loss of offsite power)? Provide experimental verification for these models.

440.52 (15C.3.3) Section 15C.3.3 implies that during a SLB, concurrent with loss-of-offsite power, the reactor trips on a low DNBR signal. It is our understanding that CESEC does not calculate DNBR. How is the time of reactor trip calculated?

440.53 (15.1) The inadvertent opening of an atmospheric dump valve event is considered as a moderate frequent event per SRP 15.1.1. Confirm that the analysis performed for this event in Section 15.1.3.2 is the limiting case identified by a qualitative comparison from the events in the same category group specified in SRP 15.1.1 (e.g., decrease in feedwater temperature, increase in feedwater flow, increase in steam flow, and inadvertent opening of a steam generator relief or safety valve). The qualitative analyes for each of the events in this group should be presented in the FSAR for staff review. Also, the results of analyses should be presented in the FSAR

for each event with their worst single failure combination and the limiting case is identified.

440.54 (15.0) The depressurization transients analyzed for System 80 were conducted utilizing the CESEC-II computer program. This program does not account for steam formation in the upper head of the reactor vessel nor for steam formation in the primary system after the pressurizer empties. Neglecting these effects can result in the improper evaluation of the system pressure and hydraulic behavior. The importance of this phenomenon was demonstrated by the St. Lucie I natural circulation cooldown event of June 11, 1980.

The modeling deficiency in CESEC-II described above has the potential for providing unacceptable results for the depressurizing transients analyzed in the FSAR. As such, for all transients which empty the pressurizer or may result in saturated conditions elsewhere in the primary system, the CESEC-II computer program must be verified to demonstrate it can correctly calculate system thermal-hydraulic responses. The staff requires the applicant to demonstrate the acceptability of the CESEC-II program to properly account for the thermal-hydraulic phenomena in question, and to demonstrate compliance with NRC regulations. In addition, we require a description of th SESEC code's ability to calculate the asymmetric cooldown between the intact and broken loops. Overlay plots of the hot leg and cold leg temperatures in the intact and broken loops should be provided.

440.55 (15.6) For small-break LOCAs, containment isolation may occur. It is our understanding that component cooling water to the RCP seals will be isolated upon containment isolation. Demonstrate that the RCP seals will remain intact and maintain the pressure boundary for the duration of the accident. Address expected

RCP operation. If seal integrity cannot be maintained, seal failure must be assumed. Discuss the maximum seal leakage rates based on operating experience. If the consequences of seal failure are assumed to be covered by the analyzed break spectrum, justify the differences in the break locations from the locations analyzed.

440.56 (6.3.3) The LOCA break spectrum analyses presented are stipulated to be applicable to any System 80 plant that conforms to the interface requirements specified within Section 6.3.3. The submittal for the LOCA analyses does not address the effects of steam generator tube plugging. The effect of a decrease in steam generator tube flow area is an increase in the peak cladding temperature (when the peak occurs during the reflood portion of the transient). If the analyses provided are considered to support generators with plugged tubes, describe the intent of the plugging the analyses support and the method used to account for the plugging. If steam generator tube plugging was not considered, the applicant will be required to perform additional ECCS analyses prior to operation with plugged generator tubes. In either case, the applicant is required to include an interface requirement on the validity of the LOCA analyses (acceptance criteria of 10 CFR 50.46) and the Technical Specification limit for the number (or percentage) of allowable plugged steam generator tubes.

440.57 (15.6.2.2) In light of recent operating experiences (the St. Lucie Unit 1 natural circulation cooldown event of June 11, 1980, and re-analyses of SAR Chapter 15 design bases events by St. Lucie in February 1981) a potential deficiency has been identified with the CESEC computer program and NSSS model. As the pressurizer cools down and the system pressure decreases, steam can form in the reactor vessel upper head due to flashing of the hot coolant in this stagnant region. The steam bubble

in the reactor vessel upper head displaces coolant from the reactor vessel into the pressurizer and the steam in the vessel head will determine the system pressure. The CESEC model used for the steam generator tube rupture event does not account for this occurrence. Further, CESEC analyses which predict that the pressurizer will empty, or that the reactor coolant system saturates, do not appear to correctly calculate the system thermal-hydraulic response and are not justified for use. These events are to be re-analyzed with a suitable model or additional justification is to be provided for the CESEC analyses to demonstrate that the computer program conservatively accounts for the formation of steam in the reactor coolant system.

440.58 (15.6.2.2) The analysis for a steam generator tube rupture does not address tube leakage in the unaffected steam generator. Provide an interface requirement for the allowable steam generator tube leakage and reference the Technical Specification limit. Confirm the analyses were performed using this allowable limit or provide justification why this leakage term can be excluded from the analyses.

440.59 (15.6.2.2) The analysis for a steam generator tube rupture is for a double-ended rupture. Provide the analyses used to determine that this is the limiting case. If a partial area break is considered, such that the steam generator relief valves open at a longer time into the transient is more primary coolant leaked to the secondary and out the SRVs, resulting in an increased dose rate.

440.60 (15.6.2.2) SRP 15.6.3 acceptance criteria requires that this event be analyzed with a concurrent loss of offsite power. Provide an analysis for the limiting case which includes a concurrent loss of offsite power.

440.61 (15.6.2.2) For the SGTR event, what prevents steam from the affected steam generator being used to drive the steam-driven auxiliary feedwater pump and exhausted to the environment? If operator action is required, confirm that no credit for operator action was given for 30 minutes, consider with your assumption for isolation of the affected steam generator. If credit was given for operator action in less than 30 minutes, provide justification why this credit can be given, or reanalyze the event assuming steam from the faulted steam generator is used to drive the steam-driven AFW pump and is exhausted to the environment.

440.62 (15.6.3, 15.6.4, 15.6.5) Provide a description of the CESEC model used to model the CVSC from the reactor coolant system to the break point. Include a description of the environmental conditions at the break point (pressure, enthalpy, break flow model used).

440.63 (15.6.3, 15.6.4, 15.6.5) Discuss the single failure assumed for these analyses. What analyses/evaluations were performed to justify that the single failures chosen were the most limiting?

440.64 (15.2) In this section, you have selected the turbine trip without a single failure as the limiting reactor coolant system pressure and the limiting radiological release event for the moderate frequent event category in the decreased heat removal by secondary system group. However, these limiting cases were not selected by a qualitative comparison of similar initiating events specified in SRP 15.2.1 through SRP 15.2.7 (e.g., loss of external load, turbine trip, loss of condenser vacuum, steam pressure regulator failure, loss of normal AC power and loss of normal feedwater flow). Provide a qualitative analysis in the FSAR for each of the initiating events in the same group per the SRP, and identify the limiting cases for the

group. Provide a detail quantitative analysis for each of the limiting cases including the limiting RCS pressure, limiting fuel performance, and the limiting radiological release.

440.65 (15.2) In this section, you have provided the loss of condenser vacuum with a fast transfer failure and technical specification steam generator tube leakage as the limiting RCS pressure and the limiting radiological release event for the limiting fault event category in the decreased heat removal by secondary system group. Although, these limiting cases may be the candidates for the limiting cases for the infrequent event category in the group, they were not selected by a qualitative comparison of similar initiating events plus a single failure specified in SRP 15.2.1 through 15.2.7. Provide a qualitative analysis in the FSAR for each of the initiating event plus a single failure in the same group per the SRP, and identify the limiting cases for the group. Provide a detailed quantitative analysis for each of the limiting cases including the limiting RCS pressure, limiting fuel performance, and the limiting radiological release. Confirm that the results of the analyses meet the acceptance criteria for these events per SRP 15.2.1.

440.66 (15A) Provide tabulations of the sequence of events, disposition of normally operating systems, utilization of safety systems, and a transient curve of primary system pressure for the total loss of primary coolant flow event. Also provide an analysis of the total loss of primary coolant flow with a single failure event. Confirm that the results of these analyses meet the acceptance criteria for these events per SRP 15.3.1.

440.67 (15.3) In Section 15.3.5 you have provided the single reactor coolant pump shaft seizure with loss of offsite power following turbine trip and with technical specification tube leakage as the limiting RCS pressure and radiological release event for the

limited fault event category. This postulated event is classified as an infrequent event per SRP 15.3.3. Confirm that the results of the analysis meet the acceptance criteria for these events per SRP 15.3.3, using the criteria stated in Question 440.35 to calculate the amount of failed fuel in this event. State the amount of failed fuel in the results of the analysis. Radiological consequences should be calculated accordingly.

440.68 (15.0) Provide results of an analysis of the reactor coolant pump shaft break as required by SRP 15.3.4 for staff review. The event should consider loss of offsite power following turbine trip and with technical specification steam generator tube leakage. The criteria stated in Question 440.35 should be used for the calculation of the amount of failed fuel for this event. State the amount of failed fuel in the results of the analysis. Radiological consequences should be calculated accordingly. Confirm that the results of the analysis meet the acceptance criteria for these events per SRP 15.3.4 which classifies this event as an infrequent event.

440.69 (15.5) In this section, you have provided the pressurizer level control system malfunction (PLCSM) with a fast transfer failure and the PLCSM with a loss of offsite power at turbine trip with technical specification steam generator tube leakage as the limiting RCS pressure and radiological release event for the limiting fault event category in the increase in reactor coolant system inventory group. However these limiting cases were not selected by a qualitative comparison of similar initiating events plus a single failure specified in SRP 15.5.1 (e.g., inadvertent operation of high pressure ECCS or a malfunction of the CVCS). Provide a qualitative analysis in the FSAR for each of the initiating events (with and without a single active failure) in the same group per the SRP, and

identify the limiting cases for the group. Provide a detailed quantitative analysis for each of the limiting cases including the limiting RCS pressure, limiting fuel performance, and the limiting radiological release. Confirm that the results of the analyses meet the acceptance criteria for these events per SRP 15.5.1.

440.70 (15.0) Provide tabulations of the sequence of events, disposition of normally operating systems, utilization of safety systems, and all necessary transient curves for the startup of an inactive reactor coolant pump event. The comparison to peak RCS pressure acceptance criteria should be included in the analysis. Also provide the results of an analysis of this event with a single failure. Confirm that the results of these analyses meet the acceptance criteria for these events per SRP 15.4.4.

440.71 (15.D) You have provided, in Section 15D, the results of an inadvertent boron dilution event without a single failure under plant cold shutdown conditions. This information is not sufficient. You should provide results of analyses for all possible boron dilution events under various plant operational modes (e.g., refueling, startup, power operation, hot standby and cold shutdown). Also provide the results of analyses of these events with a single failure. Confirm that the results of these analyses meet the acceptance criteria for these events per SRP 15.5.1. In particular, the available times per operator action between time of alarm and time to loss of shutdown margin should be shown to meet the SRP guidelines. The results of the analyses should be presented in the FSAR including tabulations of sequence of events, disposition of normally operating systems, utilization of safety systems, and all necessary transient curves for the events.

In your analysis, indicate for all modes of operation what alarms would identify to the operators that a boron dilution event was occurring. Consider the failure of the first alarm. Provide the time interval from this alarm to when the core would go critical. If a second alarm is not provided, show that the consequences of the most limiting unmitigated boron dilution event meet the staff criteria and are acceptable.

440.72 (15.2) As explained in Issue No. 1, NUREG-0138, credit is taken for closure of nonsafety-grade valves such as turbine stop valves, control valves, and bypass valves downstream of the MSIV to limit blowdown of a second steam generator in the event of a steam line break upstream of the MSIV. In the Palo Verde Plant design there are various flow paths located between the MSIV and the turbine stop valves (Figure 10.2-4) that serve various unidentified functions. To confirm satisfactory performance after a steam line break provide the following information, as applicable, related to these various flow paths that branch off between the MSIV's and the turbine stop valves:

(1) the function of the various flow paths and their maximum steam flow

(2) the type of valves

(3) the size of valves

(4) the quality of the valves

(5) design code of the valves

(6) the closure time of the valves

(7) the actuation mechanism of the valves

(8) the closure signal including sensor

(9) quality of power sources to valves and sensors

(10) quality of air supply to air-operated valves

(11) identify the valves that will remain open during main steam isolation

In addition, provide justification or analysis that the failure of an MSIV and the additional blowdown paths result in a less severe accident than that analyzed in Chapter 15.

440.73 (15.D) Several recent LERs indicate there has been a deficiency in the inadvertent boron dilution analysis at some plants. Provide an analysis of the dilution event when the RCS is drained to the hot leg.

440.74 (15.D) Recently, an operating PWR experienced a boron dilution incident due to inadvertent injection of NaOH into the reactor coolant system while the reactor was in a cold shutdown condition. Discuss the potential for a boron dilution incident caused by dilution sources other than the CVCS.

440.75 (15.6) Discuss the transient resulting from a break of an ECCS injection line. In particular, describe the flow splitting which will occur in the event of a single failure and verify that the amount of flow actually reaching the core is consistent with the assumptions used in the analysis.

440.76 (15.8) The NRC is currently considering what actions may be necessary to reduce the probability and consequences of anticipated transients without Scram (ATWS). Until such time as the Commission determines what plant modifications are necessary, we have generally concluded that pressurized water plants can continue to operate because the risk from anticipated transient without scram events in a limited time period is acceptably small. However, in order to further reduce the risk from anticipated transient without scram events during the interim period before completing the plant modifications determined by the Commission to be necessary, we have required that the following actions be taken:

1. Develop emergency procedures to train operators to recognize anticipated transient without scram events, including consideration of scram indicators, rod position indicators, flux monitors, pressurizer level and pressure indicators, pressurizer relief valve and safety valve indicators, and any other alarms annunciated in the control room with emphasis on alarms not processed through the electrical portion of the reactor scram system.

2. Train operators to take actions in the event of an anticipated transient without scram, including consideration of manually scramming the reactor by using the manual scram button, prompt actuation of the auxiliary feedwater system to assure delivery to the full capacity of this system, and initiation of turbine trip. The operator should also be trained to initiate boration by actuation of the high pressure safety injection system to bring the facility to a safe shutdown condition.

Describe how you will meet the above requirements, and provide a schedule for submittal of the ATWS procedures for staff review.

440.77 (6.3) Palo Verde only	List all ECCS valve operators and controls that are located below the maximum flood level following a postulated LOCA or main steam line break. If any are flooded, evaluate the potential consequences of this flooding both for short and long-term ECCS functions and containment isolation. List all control room instrumentation lost following these accidents.
440.78 (6.3 and 9.3.4) Palo Verde only	Because of freezing weather conditions, blocking of the vent line on the refueling water tank (RWT) has occurred on at least one operating plant. Describe design bases and features that preclude this condition from occurring in the Palo Verde Plant.
440.79 (6.3) Palo Verde only	It is our position that the SIS hotleg injection valves should be locked closed with power removed during normal plant operation in order to prevent premature hotleg injection following a LOCA.
440.80 (6.3) Palo Verde only	Your sump test program described in Section 6.2.2 is not in sufficient detail. The experimental program must demonstrate that sufficient margin in available NPSH over that required for each pump with all pumps at runout or maximum post-LOCA flow. The test must demonstrate that the design precludes conditions adverse to safety system operation. Test parameters must include: (1) minimum to maximum containment water level, (2) minimum to maximum safety system flow range in various combinations (this includes transients associated with start-up, shutdown, or throttling of a train or pump), (3) random blockage of up to 50 percent of the screens and grids, (4) approach flow for each dominant direction and combinations thereof, and (5) simulation of break flow or drain flow

impinging or originating within line of sight of the sump and its approaches.

If adverse conditions are encountered, the model configuration must be revised until an acceptable configuration is developed and demonstrated to perform over the full range of variables.

Since you choose to conduct a model test, provide details of the test program. Include information on the model size, scaling principles utilized, comparison of model parameters to expected post-LOCA conditions, and a discussion on how all possible flow conditions and screen blockages will be considered in the model tests. Whenever a reduced scale model is tested, all tendencies for vortex formation must be suppressed. Rotational flow patterns and surface dimples which might be acceptable in full scale tests, probably would not be accepted in a model program. Model testing must include some in-plant testing to demonstrate experimentally that NPSH margin exists for each pump.

440.81 (6.3) Palo Verde only

During our reviews of license applications we have identified concerns related to the containment sump design and its effect on long-term cooling following a Loss of Coolant Accident (LOCA).

These concerns are related to (1) creation of debris which could potentially block the sump screens and flow passages in the ECCS and the core, (2) inadequate NPSH of the pumps taking suction from the containment sump, (3) air entrainment from streams of water or steam which can cause loss of adequate NPSH, (4) formation of vortices which can cause loss of adequate NPSH, air entrainment and suction of floating debris into the ECCS and (5) inadequate emergency procedures and operator training to enable a correct response to these

problems. Preoperational recirculation tests performed by utilities have consistently identified the need for plant modifications.

The NRC has begun a generic program to resolve this issue. However, more immediate actions are required to assure greater reliability of safety system operation. We therefore require you take the following actions to provide additional assurance that long-term cooling of the reactor core can be achieved and maintained following a postulated LOCA.

1. Establish a procedure to perform an inspection of the containment, and the containment sump area in particular, to identify any materials which have the potential for becoming debris capable of blocking the containment sump when required for recirculation of coolant water. Typically, these materials consist of: plastic bags, step-off pads, health physics instrumentation, welding equipment, scaffolding, metal chips and screws, portable inspection lights, unsecured wood, construction materials and tools as well as other miscellaneous loose equipment. "As licensed" cleanliness should be assured prior to each startup.

   This inspection shall be performed at the end of each shutdown as soon as practical before containment isolation.

2. Institute an inspection program according to the requirements of Regulatory Guide 1.82, Item 14. This item addresses inspection of the containment sump components including screens and intake structures.

3. Develop and implement procedures for the operator which address both a possible vortexing problem (with consequent pump cavitation) and sump blockage due to debris. These procedures should address all likely scenarios and should list all instrumentation available to the operator (and its location) to aid in detecting problems which may arise, indications the operator should look for, and operator actions to mitigate these problems.

4. Pipe breaks, drain flow and channeling of spray flow released below or impinging on the containment water surface in the area of the sump can cause a variety of problems; for example, air entrainment, cavitation and vortex formation.

   Describe any changes you plan to make to reduce vortical flow in the neighborhood of the sump. Ideally, flow should approach uniformly from all directions.

5. Evaluate the extent to which the containment sump(s) in your plant meet the requirements for each of the items previously identified; namely debris, inadequate NPSH, air entrainment, vortex formation, and operator actions.

The following additional guidance is provided for performing this evaluation.

1. Refer to the recommendations in Regulatory Guide 1.82 (Section C) which may be of assistance in performing this evaluation.

2. Provide a drawing showing the location of the drain sump relative to containment sumps.

3. Provide the following information with your evaluation of debris:

   a. Provide the size of openings in the fine screens and compare this with the minimum dimensions in the pumps which take suction from the sump (or torus), the minimum dimension in any spray nozzles and in the fuel assemblies in the reactor core or any other line in the recirculation flow path whose size is comparable to or smaller than the sump screen mesh size in order to show that no flow blockage will occur at any point past the screen.

   b. Estimate the extent to which debris could block the trash rack or screens (50 percent limit). If a blockage problem is identified, describe the corrective actions you plan to take (replace insulation, enlarge cages, etc.).

   c. For each type of thermal insulation used in the containment, provide the following information:

      (1) type of material including composition and density,

      (2) manufacturer and brand name,

      (3) method of attachment,

      (4) location and quantity in containment of each type,

(5) an estimate of the tendency of each type to form particles small enough to pass through the fine screen in the suction lines.

d. Estimate what the effect of these insulation particles would be on the operability and performance of all pumps used for recirculation cooling. Address effects on pump seals and bearings.

440.82 (15.0) Palo Verde only

Section 15D.2.2.2 of the CESSAR System 80 FSAR states that the loss of instrument air event impact on the plant systems and components will be addressed in the applicant's FSAR.

Discuss the loss of instrument air for Palo Verde showing that it meets the appropriate acceptance criteria for a moderate frequency event. Causes and potential systems interactions should be addressed and the loss of instrument air should be considered during all phases of reactor operation. Also, present your plans and capability for preoperational or startup tests to substantiate the analyses.

440.83 (II.B.1) Palo Verde only

Your response to Item II.B.1 of NUREG-0737 requirements is not sufficient. Provide the following:

1. Provide diagrams and a description of the vent discharge vicinity. Verify that adequate ventilation is provided and that equipment in this area is capable of withstanding discharge of gases and liquids from the vents.

2. What size are the flow limiting orifices and what are the calculated flow rates through the vent system for both gas mixtures and liquids at operating pressures?

3. Provide drawings of the piping system from the vessel head and pressurizer through the discharge paths. In particular, show the location of the solenoid operated valves and consider potential missile hazards from them.

440.84 (II.K.3.17) Palo Verde only

Your response to Item II.K.3.17 of NUREG0737 is not complete. Provide a commitment that you will establish a program prior to fuel loading for data collection on information regarding ECCS outages. The information will contain: (1) outage dates and duration of outages; (2) cause of the outages; (3) EECS systems or components involved in the outage; and (4) collective action taken.

440.85 (II.K.3.25) Palo Verde only

Your response to Item II.K.3.25 of NUREG-0737 states that the reactor coolant pump normal cooling water system (nonsafety grade nuclear cooling water system) is backed up by the essential seals during loss of offsite AC power. Describe the manual action involved and the manual action time required for transferring the cooling water supplies. Also, state that your operating procedure allows enough time to restore the cooling water supplies to the RCP seals before you trip the RCPs. After the RCP trip, you may still need essential cooling water supply to the RCP seals.

ENCLOSURE (Attachment to Q 440.40)

PWR INFORMATION REQUEST PACKAGE

Introduction

The purpose of this package is to request specific Pressurized Water Reactor (PWR) information, both physical and operational, that will allow the plant to be modeled using advanced computer codes, such as TRAC or RELAP. These plant models will be used in the study of various transients of concern to the nuclear industry.

For organizational purposes the plant modeling has been subdivided into 7 main components:

1. Reactor Vessel
2. Steam Generator
3. Reactor Coolant Pumps
4. Pressurizer
5. Emergency Core Coolant Systems
6. Primary Coolant Piping
7. System Valves
8. Fuel Rod Design

The forward and reverse flow energy loss coeffients, required in this package, are used to describe the influence of coolant volume geometry upon coolant flow energy. For example, a 90° elbow will not only change the direction of coolant flow but will cause the coolant to lose energy as well. The coefficient is dimensionless and is a function of the friction factor and the equivalent L/D of the piping features at the junction or in the volume. The basic equation is:

$K = f\ L/D$ with f = friction factor
L = pipe length
D = pipe diameter

Features with equivalent L/Ds to be considered include (but are not restricted to) abrupt area changes, plenum volumes, moisture separators, valves and elbows. The terms 'forward' and 'reverse' apply to the normal and reversed direction of flow, respectively.

Operational information such as controls, operating conditions and alarm setpoints are critical to the correct modeling of the PWR. There are many factors that affect this information. Of prime concern, are the delays in system actuation caused by instrument deadbands, uncertainties and attached electronics. These are not specifically written in this package due to the plant specific nature of these factors. However, this information is needed and, is therefore, requested.

Due to the generalized nature of this questionaire, some of the information requested may not be applicable for a particular plant. In these cases the requested parameter should be ignored.

In addition to the requested data in this package, schematic drawings depicting each of the major components should be included.

1.0 REACTOR VESSEL

I. Inlet nozzles

A.	Inside diameter at nozzle inlet	ft
B.	Inside diameter at nozzle outlet	ft
C.	Distance from nozzle inlet to nozzle outlet	ft
D.	Forward flow energy loss coefficient	
E.	Reverse flow energy loss coefficient	
F.	Inside surface roughness	k/D

II. Downcomer

A.	Flow area as a function of elevation, relative to inlet nozzle centerline	ft^2
B.	Full power inlet temperature	°F
C.	Full power inlet pressure	psia
D.	Elevation of top of downcomer relative in inlet nozzle centerline	ft
E.	Forward flow energy loss coefficient	
F.	Reverse flow energy loss coefficient	
G.	Surface roughness	k/D
H.	Hydraulic diameter as a function of elevation, relative to inlet nozzle centerline	ft

III. Lower Plenum (below flow distributor)

A.	Flow area as a function of elevation, relative to inlet nozzle centerline	ft^2
B.	Total volume including structural material	ft^3
C.	Metal-to-water volume ratio	

D.	Forward flow energy loss coefficient	
E.	Reverse flow energy loss coefficient	
F.	Average roughness	k/D
G.	Fractional composition of structural components (e.g., SS-306 26.4%, etc)	
H.	Hydraulic diameter as a function of elevation relative to inlet nozzle centerline	ft

IV. Lower plenum flow distributor

A.	Flow area	ft^2
B.	Forward flow energy loss coefficient	
C.	Reverse flow energy loss coefficient	
D.	Composition	
E.	Axial elevation at center and at edge	ft
F.	Thickness	ft

V. Lower plenum between distributor and lower core plate

A.	Flow area	ft^2
B.	Hydraulic diameter	ft
C.	Forward flow energy loss coefficient	
D.	Reverse flow energy loss coefficient	
E.	Roughness	k/D
F.	Material composition	
G.	Total volume including structural material	ft^3
H.	Metal-to-water volume ratio	

VI. Reactor core

A. Fuel assembly

1.	Flow area	ft^2
2.	Hydraulic diameter	ft
3.	Forward flow energy loss coefficient at grid spacer	
4.	Reverse flow energy loss coefficient at grid spacer	
5.	Roughness	k/D
6.	Material composition	
7.	Total volume	ft^3
8.	Metal-to-water volume ratio	
9.	Axial elevations of center of grid spacers, relative to inlet nozzle centerline	ft

B. Control rod assembly

1.	Flow area	ft^2
2.	Hydraulic diameter	ft
3.	Forward flow energy loss coefficient at grid spacer	
4.	Reverse flow energy loss coefficient at grid spacer	
5.	Roughness	k/D
6.	Material composition	
7.	Total volume	ft^3
8.	Metal-to-water volume ratio	
9.	Axial elevations of center of grid spacers, relative to inlet nozzle centerline	ft

C. Fuel assembly with instrument

1.	Flow area	ft^2
2.	Hydraulic diameter	ft
3.	Forward flow energy loss coefficient	
4.	Reverse flow energy loss coefficient	
5.	Roughness	k/D
6.	Material composition	
7.	Total volume	ft^3
8.	Metal-to-water volume ratio	

D. Core bypass flow path(s)

1.	Flow area	ft^2
2.	Hydraulic diameter	ft
3.	Forward flow energy loss coefficient	
4.	Reverse flow energy loss coefficient	
5.	Roughness	k/D
6.	Total volume	ft^3
7.	Percentage bypass of core full power flow	%

E. Core power distributions (axial and radial for each of the following conditions)

1. Normal full power
2. Control rods 50% inserted
3. Control rods fully inserted with most significant control rod assembly stuck out

F. Reactor protective system interactions with core

G. Engineered safeguards protective system interactions with core

H. Coolant temperature at core inlet as a function of core power — °F

I. Coolant pressure at core inlet as a function of core power — psia

J. Coolant temperature at core top as a function of core power — °F

K. Coolant pressure at core top as a function of core power — psia

L. Quantity of gamma heating in core as a function of core power and axial elevation — kW

M. Reactor kinetics--beginning of life

1. Scram rod reactivity insertion as a function of time for:
   a. all rods drop
   b. all but most reactive rod drops
2. Reactivity change as a function of moderator density
3. Density reactivity change as a function of boron concentration
4. Reactivity change as a function of fuel temperature (Doppler)
5. Boron worth as a function of boron concentration and moderator temperature

N. Reactor kinetics - end of life

1. Scram rod reactivity insertion as a function of time for:
   a. all rods drop
   b. all but most reactive rod drops
2. Reactivity change as a function of moderator density
3. Density reactivity change as a function of boron concentration
4. Reactivity change as a function of fuel temperature (Doppler)

5. Boron worth as a function of boron concentration and moderator temperature

VII. Upper core plenum from top of outlet nozzles to bottom of vessel head

A.	Flow area	ft^2
B.	Hydraulic diameter	ft
C.	Forward flow energy loss coefficient	
D.	Reverse flow energy loss coefficient	
E.	Roughness	k/D
F.	Material composition	
G.	Total volume	ft^3
H.	Metal-to-water volume ratio	
I.	Inside diameter of plenum shroud orifices	ft

VIII. Upper head

A.	Flow area	ft^2
B.	Hydraulic diameter	ft
C.	Forward flow energy loss coefficient	
D.	Reverse flow energy loss coefficient	
E.	Roughness	k/D
F.	Material composition	
G.	Total volume	ft^3
H.	Metal-to-water volume ratio	

IX. Outlet nozzles

A.	Inside diameter at nozzle inlet	ft
B.	Inside diameter at nozzle outlet	ft
C.	Distance from nozzle inlet to nozzle outlet	ft
D.	Forward flow energy loss coefficient	
E.	Reverse flow energy loss coefficient	
F.	Roughness	k/D
G.	Elevation relative to inlet nozzle	ft

2.0 STEAM GENERATOR

All elevations should be relative to inlet of the steam generator reactor coolant inlet nozzle, whose elevation relative to the centerline of the reactor vessel cold leg inlet should be included. If the steam generators differ from each other the requested information should be provided for each generator. All elevations are relative to reactor vessel cold leg inlet nozzle centerline.

I. Primary Loop

A. Primary coolant inlet nozzle

1.	elevation of inlet nozzle centerline at the entrance to the inlet plenum	ft
2.	inside diameter at nozzle inlet	ft
3.	inside diameter at plenum inlet	ft
4.	length of nozzle at nozzle centerline	ft
5.	angular orientation of nozzle centerline relative to horizontal	
6.	forward flow energy loss coefficient	
7.	reverse flow energy loss coefficent	
8.	inside surface roughness	
9.	flowrate at nozzle entrance at full load	lbm/sec
10.	coolant temperature at nozzle entrance at full load	°F
11.	coolant pressure at nozzle entrance at full load	psi

B. Primary coolant inlet plenum

1.	elevation of tube sheet bundle entrance	ft
2.	plenum volume	ft^3

3.	area of plenum at entrance to tube bundle	ft^2
4.	forward flow energy loss coefficient	
5.	reverse flow energy loss coefficient	
6.	roughness	
7.	wall thickness	ft
8.	width of plenum divider (UTSG)[a] at tube sheet entrance	ft

C. Tube bundle

1.	number of flow tubes	
2.	tube ID	ft
3.	tube OD	ft
4.	elevation of tube bundle exit once through steam generator (OTSG)[b]	
5.	lengtn of flow tubes (OTSG) average length u-tube steam generator (UTSG)	ft
6.	Heat transfer area	ft^2
7.	Heat transfer area including curved section (UTSG only)	ft^2
8.	forward flow energy loss coefficient	
9.	reverse flow energy loss coefficient	
10.	internal roughness	
11.	total volume in tubes	ft^3
12.	maximum/minimum and average tube elevation (UTSG)	ft

---

a. U-tube steam generator

b. Once through steam generator

D. Primary Coolant Outlet Plenum

1.	elevation of tube sheet exit	ft
2.	plenum volume	ft^3
3.	area of plenum at tube exit	ft^2
4.	forward flow energy loss coefficient	
5.	reverse flow energy loss coefficient	
6.	roughness	
7.	wall thickness	ft

E. Primary Coolant Outlet Nozzle

1.	elevation of inlet nozzle centerline at the entrance to the inlet plenum	ft
2.	inside diameter at nozzle outlet	ft
3.	inside diameter at nozzle inlet	ft
4.	length of nozzle at nozzle centerline	ft
5.	angular orientation of nozzle centerline relative to horizontal	
6.	forward flow energy loss coefficnent	
7.	reverse flow energy loss coefficnent	
8.	inside surface roughness	
9.	flowrate at nozzle entrance at full load	lbm/sec
10.	coolant temperature at nozzle entrance at full load	°F
11.	coolant pressure at nozzle entrance at full load	psi

II. Secondary Loop

A. Feedwater Supply

1.	feed flow at full load	lbm/sec
2.	feed flow as a function of	lbm/sec
	a. load	
	b. mixture level	
3.	feedwater temperature	°F
4.	feedwater pressure	psi
5.	auxiliary feed systems	
	a. initiating setpoints	
	b. flow rate each type as a function of pumps running	lbm/sec
	c. aux feed temperature (each type)	°F
	d. number of pumps (each type)	
	e. aux feed pressure (each type)	psi
	f. aux feed inlet elevation	ft
6.	main feed inlet elevation	ft

B. Downcomer (Preheater) Section

1.	downcomer (OTSG)	
	a. flow area as function of elevation above top of lower tube plate	ft^2
	b. roughness	
	c. forward flow energy loss coefficient	
	d. reverse flow energy loss coefficient	
	e. downcomer shroud ID	ft
	f. downcomer shroud OD	ft

g.	baffle shroud ID	ft
h.	baffle shroud OD	ft
i.	shell ID	ft
j.	shell OD	ft
k.	elevation of baffle shroud bottom above lower tube plate	ft
l.	elevation of downcomer shroud top above lower tube plate	ft
m.	total volume in section	ft^3
n.	heat transfer area	ft^2

2. preheater (UTSG)

a.	elevation of	
	(1) top of preheater section	ft.
	(2) bottom of preheater section	ft
b.	flow area as function of height of section	ft^2
c.	forward flow energy loss coefficient	
d.	reverse flow energy loss coefficient	
e.	roughness external tubes/ baffles	
f.	heat transfer surface	ft^2
g.	coolant volume of section	ft^3
h.	metal volume in section	ft^3

3. operating conditions

a.	outlet temperature	°F
b.	outlet pressure	psi
c.	outlet flow if different from inlet flow	lbm/sec

4. outlet to tube bundle (boiler)

a.	flow area	ft^2
b.	forward flow energy loss coefficient	
c.	reverse flow energy loss coefficient	

C. Tube Bundle (Boiler)

1. flow area through tubes as a function of height — ft^2
2. total heat transfer area — ft^2
3. forward flow energy loss coefficient
4. reverse flow energy loss coefficient
5. roughness (external tubes)
6. elevations
   a. top of baffle assembly — ft
   b. bottom of upper tube plate — ft
   c. top of baffle shroud — ft
7. height of top of nucleate boiling region as a function of load (above lower tube plate) — ft
8. height of top of film boiling region as a function of load (above lower tube plate) — ft

9. flow losses in baffle region
10. metal volume in region — ft^3
11. operating conditions
    a. outlet temperature — °F
    b. outlet pressure — psi
    c. quality as a function of height in region
12. total free volume in region — ft^3

D. Superheat Steam Downcomer (OTSG)[a]

1. steam generator ID — ft
2. roughness
3. forward flow energy loss coefficient
4. reverse flow energy loss coefficient
5. elevations:
    a. bottom of downcomer — ft
    b. steam outlet centerline — ft
6. flow area as a function of height — ft^2
7. steam outlet nozzle ID — ft
8. steam conditions at exit
    a. outlet temperature — °F
    b. outlet pressure — psi
    c. flow rate as a function of load — lbm/sec
9. heat transfer area — ft^2
10. total free volume in region — ft^3

E. Steam Dome (UTSG)[b]

---

a. Once through steam generator
b. U-tube steam generator

1.	flow area as a function of height (top of tubes to swirl vane moisture separator(s) (SVMS)	ft^2
2.	elevations	
	a. top of tube bundle	ft
	b. bottom of SVMS	ft
	c. top of SVMS	ft
	d. bottom of steam dryers	ft
	e. top of steam dryers	ft
	f. steam outlet	ft
3.	SVMS (Steam Separators - CE)[a]	
	a. forward flow energy loss coefficient	
	b. reverse flow energy loss coefficient	
	c. roughness	
	d. flow area through SVMS	ft^2
	e. recirc. flow as a function of load	lbm/sec
	f. number of swirl vanes	
	g. number of steam separators (CE)	
4.	steam dryers	
	a. forward flow energy loss coefficient	
	b. reverse flow energy loss coefficient	
	c. roughness	
	d. flow area through dryers	ft^2
	e. recirc. flow as a function of load	
5.	total free volume in region	ft^3
6.	total metal volume	ft^3
7.	operating conditions	
	a. steam outlet pressure	psi
	b. steam outlet temperature	°F
	c. steam flow as a function of load	lbm/sec

a. Combustion Engineering

8. steam outlet nozzle ID ft

F. SG Material Composition

1. SG vessel
2. baffles
3. downcomer shroud (OTSG)
4. tube support plates
5. primary tubes
6. SVMS (steam separators)
7. steam dryers

G. Valves

1. main steam isolation valve
   a. valve diameter ft
   b. control setpoints
   c. distance from SG outlet nozzle
      I. vertical ft
      II. horizontal ft
2. relief valves
   a. valve diameters ft
   b. control setpoints
   c. distance from SG outlet nozzle
      I. vertical ft
      II. horizontal ft
3. atmospheric dump valves
   a. valve diameters ft
   b. control setpoints
   c. distance from SG outlet nozzle
      I. vertical ft
      II. horizontal ft

3.0 REACTOR COOLANT PUMPS

I. GEOMETRY

A. Pump volume — ft^3

B. Effective pump volume flow area — ft^2

C. Effective pump volume hydraulic diameter — ft

D. Pump volume flow length — ft

E. Pump volume height — ft

F. Pump volume elevation — ft

G. Pump inlet (suction)

1. flow area — ft^2
2. hydraulic diameter (ft) — ft
3. elevation — ft
4. forward flow energy loss coefficient
5. reverse flow energy loss coefficient

H. Pump outlet (discharge)

1. flow area — ft^2
2. hydraulic diameter (ft) — ft
3. elevation — ft
4. forward flow energy loss coefficient
5. reverse flow energy loss coefficient

2. PERFORMANCE

A.	Rated angular velocity	rev/min
B.	Rated volumetric flow	ft^3/sec
C.	Rated head	ft
D.	Rated pump torque	$ft\text{-}lb_f$
E.	Rated pump motor torque	$ft\text{-}lb_f$
F.	Rated density	lb_m/ft^3
G.	Operating parameters for normal steady state at 100% rated plant conditions	
	1. angular velocity	rev/min
	2. volumetric flow	ft^3/sec
	3. head	ft
	4. pump torque	$ft\text{-}lb_f$
	5. pump motor torque	$ft\text{-}lb_f$
	6. density	lbm/ft^3
H.	Pump and pump motor moment of inertias	lb_m/ft^2 lb_m/ft^2
I.	Pump motor torque vs. pump motor speed table	$ft\text{-}lb_f$ rev/min
J.	Pump frictional torque coefficients as a function of pump angular velocity	

K. Maximum forward and reverse pump rotational velocities rev/min

rev/min

L. Single phase nomologous pump data

1. Require 16 data tables of the independent variable vs. each dependent variable with definition of terms in variables given in Table 1:

TABLE 1. PUMP HOMOLOGOUS CURVE DEFINITIONS

Regime Number	Regime Mode ID	Name	α	v	v/α	Independent Variable	Dependent Variable: Head	Dependent Variable: Torque
1	HAN	Normal	≥ 0	≥ 0	≤ 1	v/α	h/α^2	β/α^2
2	HVN	Pump	≥ 0	≥ 0	≥ 1	α/v	h/v^2	β/v^2
3	HAD	Energy	≥ 0	≤ 0	≥ -1	v/α	h/α^2	β/α^2
4	HVD	Dissipation	≥ 0	≤ 0	≤ -1	α/v	h/v^2	β/v^2
5	HAT	Normal	≤ 0	≤ 0	≤ 1	v/α	h/α^2	β/α^2
6	HVT	Turbine	≤ 0	≤ 0	≥ 1	α/v	h/v^2	β/v^2
7	HAR	Reverse	≤ 0	≥ 0	≥ -1	v/α	h/α^2	β/α^2
8	HVR	Pump	≤ 0	≥ 0	≤ -1	α/v	h/v^2	β/v^2

α = Rotational velocity ratio. (actual rotational velocity/rated rotational velocity).

v = Volumetric flow ratio. (actual volumetric flow/rated volumetric flow).

h = Head ratio. (actual head/rated head).

β = Torque ratio. (actual torque/rate torque)

M. Four quadrant curves (required only if single phase homologous curves of Item 2-L of above are not available).

1. require data tables (or plots if data are not available) describing the pump characteristics in terms of:

volumetric flow	ft^3/sec
rotational velocity	rev/min
head	ft
and torque	$ft\text{-}lb_f$

N. Two phase pump data

(1) Require fully degraded two phase homologous pump data (consisting of 16 data tables in the same format as that described in Item 2-L) with a specific correlation between void fraction and two phase head and torque relative to single phase head and torque

(2) If the data and correlation of Item 2-N-1 of above are not available, provide any available two phase pump data and related correlation(s) with complete explanatory information

III. THERMALHYDRAULIC (CONDITIONS FOR NORMAL STEADY STATE AT 100% RATED PLANT CONDITIONS

A. Pump volume

1.	average pressure	lb_f/in^2 absolute
2.	average temperature	°F
3.	average quality	%

B. Pump suction and discharge junctions

1.	mass flow	lb_m/sec

IV. CONTROL LOGIC

A. Require all trip setpoints, logic, and interlocks associated with the tripping off of each pump

B. Require all reactor system information needed to interpret the requested information of Item 4-A

4.0 PRESSURIZER

I. Tank

A.	OD	ft
B.	ID	ft
C.	Height	ft
D.	Total internal volume including structural materials	ft^3
E.	Flow area as a function of height	$ft^2 ft$
F.	Composition	

II. Surge line

A.	OD pipe	ft
B.	ID pipe	ft
C.	Rougnness	k/D
D.	Forward flow energy loss coefficient	
E.	Forward flow energy loss coefficient	
F.	Pipe length	ft
G.	Number of elbows	
H.	Elevations	
	1. Hot leg connection	ft
	2. Pressurizer connection	ft

III. Surge line nozzle

A. Elevation of nozzle inlet centerline at entrance to pressurizer

B. ID nozzle inlet

C. ID pressurizer inlet

D. Length of nozzle at nozzle centerline

E. Forward flow energy loss coefficient at pressurizer inlet

F. Reverse flow energy loss coefficient at pressurizer inlet

G. Inside surface roughness

IV. Safety nozzles

A.	Flow area	ft^2
B.	Flow resistance	
C.	Maximum nozzle capacity	lbm/sec
D.	Operational setpoint	psia

V. Relief nozzle

A.	Flow area	ft^2
B.	Flow resistance	
C.	Maximum nozzle capacity	lbm/sec
D.	Operational setpoint	psia

VI. Pressurizer heaters

A.	Number of rods	
B.	Outer diameter of rod	ft
C.	Total heat transfer surface area	ft^2
D.	Power input	kW
E.	Composition	
F.	Heater setpoints	
	1. Normal operation	
	2. Transient operation	

VII. Operational

A.	Water volume as a function of load	ft^3
B.	Operating conditions as a function of load	
	1. Pressure	psia
	2. Temperature	°F
	3. Quality	%
	4. Boron concentration	ppm

C. Spray line volume flow as a function of ΔP

D. Spray line setpoints

1. Normal operation psi

2. transient operation psi

Accumulator

A. Tanks

1.	Number of tanks	
2.	OD	ft
3.	ID	ft
4.	Total volume	ft^3
5.	Height	ft
6.	Flow area as a function of height	ft^2/ft
7.	Composition	

B. Surge line

1.	Junction flow area	ft^2
2.	Pipe OD	ft
3.	Pipe ID	ft
4.	Total length	ft
5.	Forward flow energy loss coefficient	
6.	Reverse flow energy loss coefficient	
7.	Roughness	k/D
8.	Elevations	
	a. Tank connection	ft
	b. Cold leg connection	ft

C. Operational (Both nominal and upper and lower limits)

1.	Liquid level	ft
2.	Liquid volume	ft^3

3. Operating conditions
   a. Pressure — psi
   b. Temperature — °F
   c. Boron concentration — ppm
4. Fill gas
   a. Composition
   b. Volume

II. HPIS

1. Injection liquid conditions
   a. Pressure — psia
   b. Temperature — °F
   c. Boron concentration — ppm
2. Flow rate as a function of primary system pressure and # pumps — lbm/sec
3. Number of pumps
4. Operational setpoints — psia

III. LPIS

1. Injection liquid conditions
   a. Pressure — psia
   b. Temperature — °F
   c. boron concentration — ppm
2. Flow rate as a function of primary system pressure and # pumps — lbm/sec
3. Number of pumps
4. Operational setpoints — psia
   a. actuation time delays — s

IV. Charging system

1.	Injection liquid conditions	
	a. Pressure	psia
	b. Temperature	°F
	c. Enthalpy	BTU/lbm
2.	Flow rate as a function of primary system pressure and # pumps	lbm/sec
3.	Number of pumps	
4.	Operational setpoints	psia

6.0 PRIMARY COOLANT PIPING SYSTEM

The primary coolant system consists of the piping leading from the reactor vessel to the steam generators, from the steam generators to the reactor coolant pumps, and from the coolant pumps to the reactor vessel inlet nozzles. To adequately model the piping, information concerning flow direction changes, the presence of valves, elbows, tees and changes in flow areas must be adequately described.

In the attached tables the locations where flow conditions change in a piping section are requested using a cylindrical coordinate system. The origin of the coordinate system is at the intersection of the reactor vessel axial centerline with a utility-designated reactor vessel inlet nozzle horizontal centerline. Angular references are counterclockwise and will be with respect to the referenced inlet nozzle. The relationship of the cylindrical coordinate system to the reactor vessel is shown schematically in Figure 1.

An example of how piping component locations would be specified is as follows. A section of primary piping connecting a reactor coolant pump to a reactor inlet nozzle is shown schematically in Figure 2. For simplicity, assume the piping does not have any flow area reductions, changes in inner

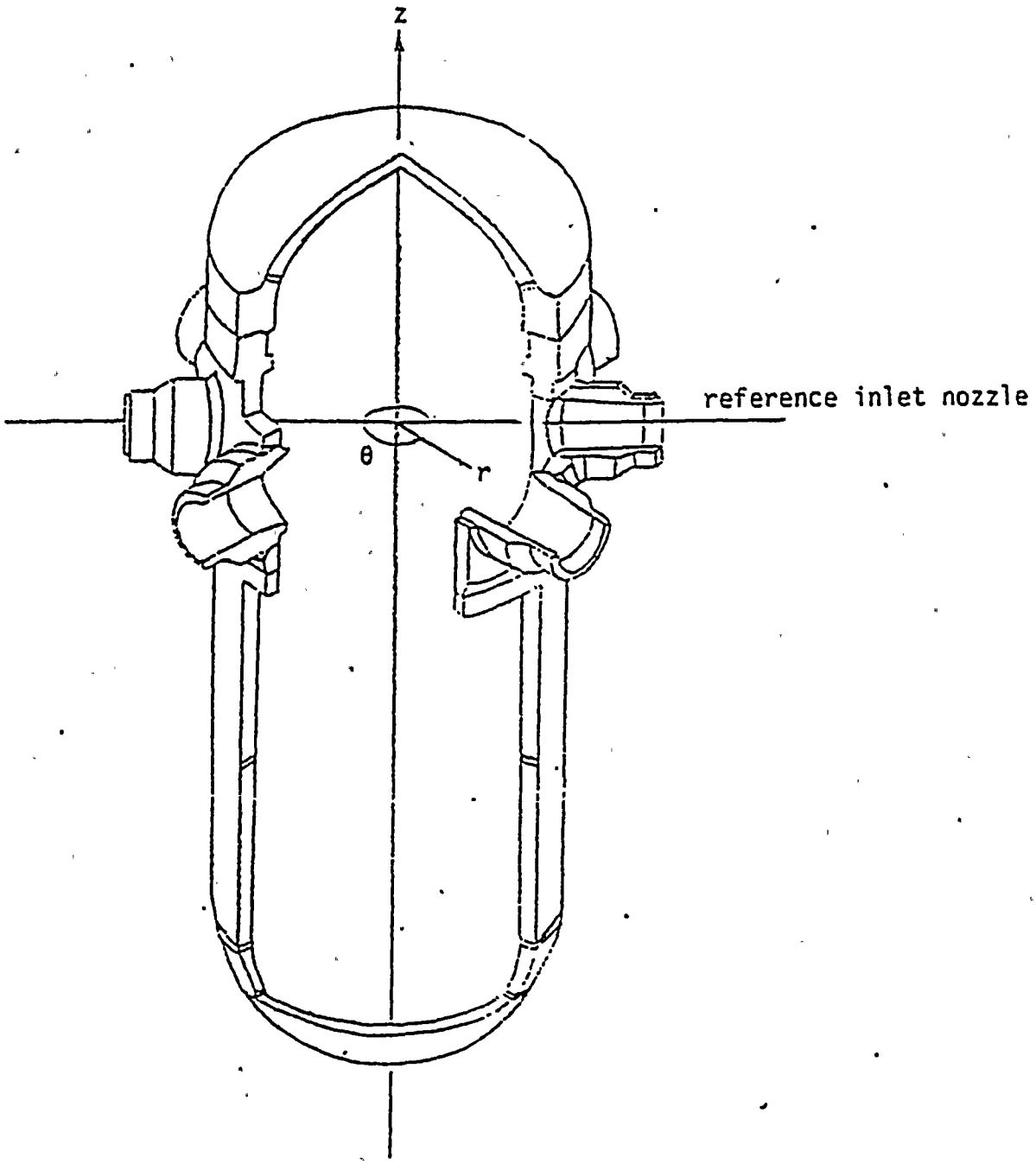

Figure 1. Piping system cylindrical coordinate system.

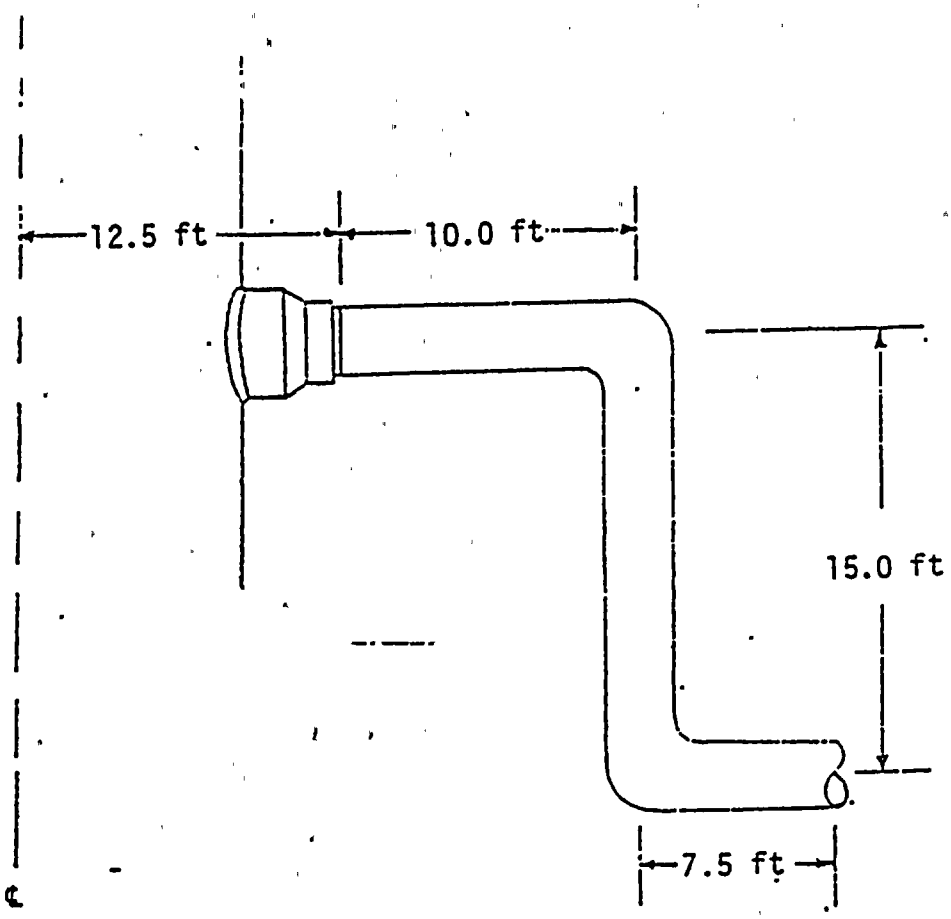

Figure 2. Piping to reactor vessel inlet nozzle.

surface roughness, valves, or piping penetrations. The angular orientation of the inlet nozzle with respect to the reference inlet nozzle is 270°. The various coordinate locations should be systematically specified, preferably starting where the coolant enters this section of piping and ending at the inlet to the reactor vessel inlet nozzle. The convention of starting at the normal coolant flow inlet and ending at the normal coolant flow exit should be followed througnout the reactor coolant system piping description.

The first coordinate location to be specified is at the coolant pump discharge nozzle exit. This location is also at 270° orientation, and is 30.0 ft from the reactor vessel axial centerline and 15.0 ft below the inlet nozzle centerline. Its coordinate location is therefore 270°, 30.0 ft, -15.0 ft.

The next coordinate location of interest is the 90° elbow where the coolant flow direction changes from horizontal to vertical. As shown in the figure, the only coordinate that has changed is the distance, r, from the reactor vessel centerline. The location of this 90° elbow is therefore 270°, 22.5 ft, -15.0 ft.

The coordinate location of the next elbow is 270°, 22.5 ft, 0.0 ft. This is assuming all inlet nozzles on the reactor vessel are at the same elevation. Note that the only coordinate value to change was the elevation (-15.0 ft to 0.0 ft).

The final coordinate location is the inlet to the reactor vessel inlet nozzle, which is at 270°, 12.5 ft, 0.0 ft.

For changes in coolant flow direction of greater than 90°, the section of piping should be divided into two or more sections, such that no one section represents more than a 90° change in direction.

In addition to information concerning the physical layout of the primary piping, several other items of data are needed to accurately model the system at the specified points of interest. These are:

1. Flow area

2. Pipe inside roughness

3. Hydraulic diameter

4. Forward flow energy loss coefficient(s)

5. Reverse flow energy loss coefficient(s)

6. If an area change, abrupt or smooth

7. Normal condition coolant pressure

8. Normal condition coolant temperature

9. Normal condition coolant fluid component flow rate

10. Normal condition coolant vapor component flow rate

11. Piping material, e.g. SS-306, inconel X750, etc

12. Piping thickness

13. Reason for description, e.g. motor valve, pipe penetration, piping tee, etc

Table 2 is provided to simplify the presentation of the data. The first column is for a user-supplied reference number in the event there is a need for further information. For example, if there is a primary loop

Type of Area Change	Coolant Pressure (psi)	Coolant Temperature (°F)	Coolant Fluid Flow Rate (lbm/sec)	Coolant Vapor Flow Rate (lbm/sec)	Piping Material	Piping Thickness (ft)	Comments

isolation valve, additional information would be required. The location of the valve would be specified using the user-supplied item number. It is requested that the item numbers be uniquely specified to eliminate possible misintrepretations of data.

7.0 SYSTEM VALVES

A separate section is provided for description of the various valves in the primary and secondary systems. This has been done so that the additional information required for describing the various valves does not cause unnecessary clutter.

The basic information required for all valves is as follows:

1. Location of valve in system

   a. component name, or

   b. item number, if located in primary piping (refer to section on primary piping)

2. Valve type

   a. check valve
   b. inertial check valve (flapper)
   c. motor valve
   d. servo valve

3. Valve flow area in full open position

4. Forward flow energy loss coefficient(s)

5. Reverse flow energy loss coefficient(s)

6. Presence of any flow area change

7. Subcooled discharge coefficient

8. Two-phase discharge coefficient

9. Normal conditions fluid flow rate

10. Normal conditions vapor flow rate

Specific information related to a particular type of valve is given below.

1. Check valves

   a. presence or absence of hysteresis
   b. normal valve position--open or closed
   c. closing backpressure
   d. leak ratio--fraction of valve area when valve is normally closed

2. Inertial check valves (see Figure 3)

   a. repeatability of operation
   b. initial valve position--open or closed
   c. closing backpressure (P)
   d. leak ratio--fraction of valve area when valve is normally closed
   e. initial flapper angle (θ_o)
   f. minimum flapper angle (θ_{min})
   g. maximum flapper angle (θ_{max})
   h. moment of inertia of flapper
   i. initial angular velocity (ω)
   j. moment arm length of flapper (L)
   k. radius of flapper disk
   l. mass of flapper (W)

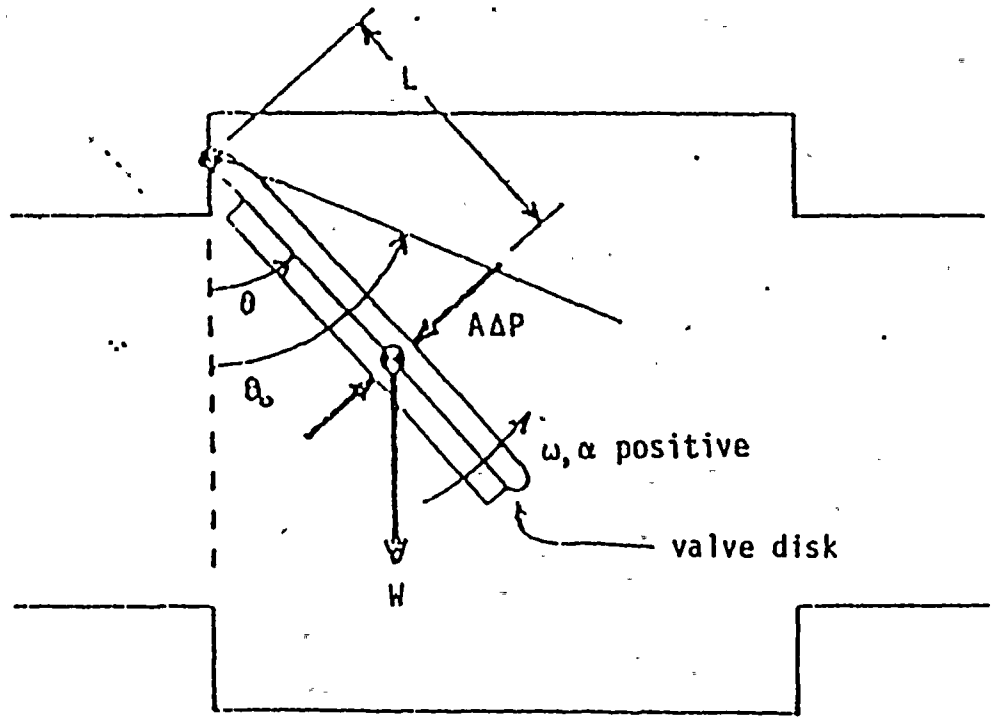

Figure 3. Inertial check valve.

3. Motor valve

a. conditions initiating motion
b. conditions terminating motion
c. valve change rate--either

(1) rate of change of the normalized valve area as the valve opens and closes, or

(2) rate of change of the normalized valve steam position

d. initial position of the valve

e. If 3.c.2 is given--normalized valve area as a function of normalized stem position

4. Servo valve--use one of the following

a. normalized valve area as a function of the controlling parameter(s)
b. normalized stem position as a function of the controlling parameter(s)

5. Motor and Servo valves--for smooth area changes only

a. forward flow energy loss coefficient(s) as a function of normalized stem position
b. reverse flow energy loss coefficient(s) as a function of normalized stem position

8.0 FUEL ROD DESIGN

I. Fuel Pellet Data

A.	Composition	
B.	Enrichment(s)	% U-235
C.	Cold state temperature for fuel dimensions	°F
D.	Density	lbm/ft^3
E.	Fuel pellet height	ft
F.	Diameter	ft
G.	Pellet dish spherical radius	ft
H.	Pellet dish depth	ft
I.	Pellet dish diameter	ft
J.	Burnup at end of each cycle	MWd/MTU
K.	Fuel sintering temperature	°F
L.	O/M ratio	
M.	Fuel surface roughness	μm
N.	Radial power distribution across pellet such that	

$$\sum_{n=1}^{N} \frac{P_n (r_{n+1}^2 - r_n^2)}{r_f^2} = 1$$

where

r_f = radius to outside of fuel pellet

r_n = inner radial coordinate of nth mesh spacing

r_{n+1} = outer radial coordinate of nth mesh spacing

P_n = power profile factor for n^{th} mesh spacing

N = number of mesh spacings in fuel

II. Fuel Rod Data

A.	Fuel stack height	ft
B.	Fuel stack insulating pellets	
	1. composition	
	2. length	
	a. top pellet	ft
	b. bottom pellet	ft
C.	Upper plenum volume including spring	ft^3
D.	Plenum spring	
	1. composition	
	2. number of coils	
	3. uncompressed height	ft
	4. uncompressed outer diameter	ft
	5. spring wire diameter	ft
E.	Fill gas composition	
F.	Fill gas pressure at cold state	psia
G.	Fill gas temperature at cold state	°F
H.	Fuel rod cladding	
	1. composition	
	2. inside diameter	ft
	3. outside diameter	ft
	4. fuel rod length	ft
	5. arithmetic mean roughness	μm
I.	Axially averaged and time averaged fast neutron flux cladding exposed to during lifetime. Fast neutron lower threshold is 1 MeV.	neutrons/m^2-sec
J.	Axially averaged and time averaged thermal neutron flux cladding exposed to during lifetime	neutrons/m^2-sec
K.	Time span of cladding neutron exposure	days
L.	Fuel rod pitch	ft

III. Fuel Rod/Assembly Thermalhydraulic Data

A.	Hydraulic diameter, nominal channel	ft
B.	Rod average linear heat rate	kW/ft
C.	Peak to average heat flux factors as a function of axial elevation	
D.	Hot channel and hot spot parameters	
	1. maximum heat flux	BTU/hr-ft^2
	2. maximum linear heat rate	kW/ft
	3. fuel maximum temperature	°F
	4. cladding maximum temperature	°F
	5. hot channel outlet temperature	°F
	6. hot channel outlet enthalpy	BTU/lbm
	7. DNB ratio (W-3 correlation), steady state	

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