Docket No.: 50-206 MAY 28 1986

Mr. Kenneth P. Baskin, Vice President
Nuclear Engineering
Safety and Licensing Department
Southern California Edison Company
2244 Walnut Grove Avenue
P.O. Box 800
Rosemead, California 91770

Dear Mr. Baskin:

SUBJECT: VERIFICATION OF TELECON INFORMATION

Re: San Onofre Nuclear Generating Station, Unit 1

During the staff's review of your recent submittals dated April 8, and May 1, 1986, conference calls to obtain further information were held on May 2 and May 7, 1986. Enclosed is the staff's record of the questions and answers provided during the discussions. Please review the enclosure to verify the responses. Any needed corrections should be formally transmitted to the NRC.

The reporting and/or recordkeeping requirements contained in this letter affect fewer than ten respondents; therefore, OMB clearance is not required under P.L. 96-511.

Sincerely,

/s/

Richard F. Dudley, Project Manager
Project Directorate #1
Division of PWR Licensing-A

Enclosure:
As Stated

8606040339 860528
PDR ADOCK 05000206
S PDR

Office:	LA/PAD#1	PM/PAD#1	PD/PAD#1
Surname:	PShuttleworth	RDudley/tg	GLear
Date:	05/ /86	05/27/86	05/28/86

Mr. Kenneth P. Baskin
Southern California Edison Company

San Onofre Nuclear Generating Station
Unit No. 1

cc
Charles R. Kocher, Assistant
General Counsel
James Beoletto, Esquire
Southern California Edison Company
Post Office Box 800
Rosemead, California 91770

David R. Pigott
Orrick, Herrington & Sutcliffe
600 Montgomery Street
San Francisco, California 94111

Mr. Stephen B. Allman
San Diego Gas & Electric Company
P. O. Box 1831
San Diego, California 92112

Resident Inspector/San Onofre NPS
c/o U.S. NRC
P. O. Box 4329
San Clemente, California 92672

Mayor
City of San Clemente
San Clemente, California 92672

Chairman
Board of Supervisors
County of San Diego
San Diego, California 92101

Director
Energy Facilities Siting Division
Energy Resources Conservation &
Development Commission
1516 - 9th Street
Sacramento, California 95814

Regional Administrator, Region V
U.S. Nuclear Regulatory Commission
1450 Maria Lane
Walnut Creek, California 94596

Joseph O. Ward, Chief
Radiological Health Branch
State Department of Health
Services
714 P Street, Office Bldg. 8
Sacramento, California 95814

Mr. Hans Kaspar, Executive Director
Marine Review Committee, Inc.
531 Encinitas Boulevard, Suite 105
Encinitas, California 92024

MAY 28 1986

Distribution Copies:
Docket No(s)
NRC PDR
Local PDR
TNovak, Actg. DD
NThompson, DHFT
ELD
EJordan
BGrimes
JPartlow
GLear
RDudley
PShuttleworth
ACRS (10)
PAD#1 r/f
PAD#1 p/f

ENCLOSURE

REACTOR SYSTEMS BRANCH QUESTIONS AND LICENSEE RESPONSES PERTINENT TO THE SONGS 1 WATER HAMMER ACCIDENT OF NOVEMBER 21,1985

The Reactor Systems Branch (RSB) reviewed a portion of the documentation pertinent to the accident which occurred at SONGS 1 on November 21, 1985. This review, and discussions with Plant Systems Branch personnel, resulted in the generation of twenty questions. These served as a tentative agenda for telephone conference calls. Additional questions were generated in the interval between preparation of the tentative agenda and the conduct of the conference calls. Further questions resulted from the discussion during the conferences. This document provides the initial questions (Nos. 1 - 20), the additional questions, the licensee responses, and additional staff comments (where appropriate). Note the responses have been prepared by the staff, and represent the staff understanding of the licensee response. They may not be considered to represent a licensee response until the licensee has had an opportunity to review and correct them.

Telephone conference calls were conducted on May 2 and May 7, 1986. The May 2 conference participants were W. Fournoy, L. Bennett, and R. Ornelas representing SONGS 1; and R. Dudley, C. Tinkler, J. Shapaker, and W. Lyon from NRC. Participants in the May 7 call were W. Fournoy, L. Bennett, R. Dudley, C. Tinkler, and W. Lyon.

REFERENCE 1: "Investigation Report, San Onofre Unit 1 Water Hammer Event of November 21, 1985", Docket No. 50-206, Southern California Edison, April 1986.

1. <u>STAFF.</u> We understand SONGS 1 now has three AFW pumps. All information we have is applicable to the old two pump configuration. Please provide the following pertinent to the three pump configuration:

a. P&ID,

b. Flow rate information as a function of various pumping configurations, and

c. Preliminary operational information such as anticipated conditions of use, approximate steps necessary to initiate use and approximate timing of pump initiation, valve orientations, and operating procedures insofar as they differ from the two pump configuration.

It is not the staff's intent to review the new AFW configuration at this time. Rather, the intent is to obtain a preliminary assessment of the potential influence on plant response and accident mitigation insofar as comparison of analyses, the November 21 accident, and the existing plant configuration is concerned.

SONGS. The third AFW pump does not affect the licensing basis at this time or during the upcoming fuel cycle. Operation is completely manual. Interconnections are immediately upstream of the normal AFW flow control valves. The flow rate is approximately twice that of the existing AFW pumps, with similar shutoff head characteristics. Procedure preparation is incomplete, but it is known that they will include operation of this pump and its associated diesel generator in an extreme emergency (beyond design basis).

The rated flow rate of the electrically driven AFW pump is 208 gpm, and that of the turbine driven AFW pump is 235 gpm. The new (third) pump rated flow rate is approximately 400 gpm.

STAFF RESPONSE. This is sufficient information to establish that the third AFW pump will not impact the design basis analysis, and the potential benefits can be realized in extreme beyond basis conditions if it can be placed in operation in time. Information to be discussed later in this document indicates this is a reasonable assumption. It is not the intention of the staff to further investigate the beyond design basis aspects of this plant improvement at this time.

2. STAFF. Damage to valves is described in which parts were not recovered. The only mention we have found in regard to the potential location of these parts is on page 6-188, which contains the statement "Investigation of the steam generator feedring was performed to verify that the water hammer was not initiated from the feedring and that some of the missing parts of the check valves did migrate to the feedring."

a. What were the results of the examination of the feedring with respect to the missing parts?
b. What parts (or pieces) have not been recovered?
c. What assurance can be provided that the above parts, if any, are not in a steam generator?
d. If we cannot be assured there are no parts (or pieces) in a steam generator, please provide an assessment of the potential impact with respect to steam generator tube rupture.

SONGS. A careful inventory was taken of all parts which should have been found and all have been located. Those which have not been physically recovered have been identified as located in the SG feed rings. The "A" steam generator feedring contains a 1½" diameter hex nut with a piece of fractured stud inside. The "B" steam generator feedring contains 2 1½" diameter hex nuts, one of which contains a piece of fractured stud, and a 2½" - 3" flat washer. One of the hex nuts was previously identified in 1982 and tack welded to the feedring. The weld is still intact. The nuts and the washer are stainless steel. The steam generator feedrings are made of ferritic steel, with 1" diameter flow holes. All parts, with the exception of one small cotter pin, are too large to get through the feed ring holes. Assessment of this pin is that even if it gets into the steam generator secondary side, it is too small to be of concern. All available evidence is that all parts will remain in the feed rings where they are presently located. They are in low flow rate regions, and they are not expected to be influenced by feed water flow.

In addition to the above inspections, the secondary side of SGs A and C have been examined for presence of foreign objects. Nothing was found.

3. STAFF. The main feedwater pumps are described as having direct drive lubrication systems with an electrically driven backup system. What are the sources of electrical power for the electrically driven lubrication system?

SONGS. Power is provided from the same electrical bus that provides power to the respective feed water pump motors. Hence, emergency power should be available to the lubrication system if power is available to operate the pump.

4. STAFF. A modification to the main feedwater flow control valves is described on the bottom of page 6-138 and the top of page 6-139. Is there a potential impact upon re-initiation of main feedwater if such action is desired under accident conditions?

SONGS. This modification does not affect the design basis for the plant. If desired, the valve trip can be over-ridden from the control room and feed water re-initiated.

5. STAFF. The Reference report contains a discussion of response to loss of all AC power on pages 6-238 and 6-239. The discussion addresses RCP thermal response and design temperature limits, and analyses are mentioned in which the inlet water to the component cooling water heat exchanger will not reach 200^{o}F for three hours. This is cited as justification for allowing two hours for restoration of AC power (provided the steam driven auxiliary feedwater pump is in operation). Please provide additional information pertinent to the analyses. We understand the CCW system has a battery powered DC pump which we assume would be operated to achieve the cited thermal response. Is this correct? What thermal energy rate is input to the fluid by the pump and was this considered in the analysis? What is the response of valves in the CCW system to the plant conditions, including the loss of all AC power. For example, do any of the valves automatically close under loss of all AC power conditions? What is the initiation time for start of the DC pump and is its operation addressed in the EOIs?

SONGS. The staff is correct in the belief that the CCW system has a battery powered DC pump which can be used under loss of all AC power conditions. The thermal energy input rate from the pump is approximately 3 HP, which is negligible in comparison to the thermal energy received from the RCP. An analysis performed in 1981 showed that it took 2 hrs 59 min for the CCW heat exchanger to reach 200 ^{o}F under loss of all AC power conditions.

No valve movements occur under these conditions which would impede flow. There are several control valves associated with the CCW system, but the valves of potential concern fail open and therefore should not impede flow under the conditions of interest here. The total volume of the CCW was not used since the appraisal of the analyst was that not all of the CCW system would be involved in the flow path. Pump initiation is automatic upon detection of low CCW pump discharge pressure. This DC backup pump is addressed in the EOIs.

6. STAFF. The case 6 analysis of loss of main feedwater with the single failure of the motor-driven AFW pump is cited as equivalent to a station blackout (page 6-7). In this analysis, the SG level at 10 minutes is stated to be 50%. What was the level at the time of loss of feedwater, and how does this compare to technical specification requirements?

   SONGS. There is no technical specification or trip action applicable to SG level. The nominal operating level of 30% of the narrow range (approximately 38000 lbs) was used for the analyses. There is an alarm at 26%.

7. STAFF. The Westinghouse St. Lucie Cooldown Study is cited as the source of information showing a pressurizer cooldown rate of 10^{o}F/hr. Please justify application of this value to the SONGS pressurizer. Address size, thermal insulation and heat loss rate, inventory, sensible heat addition and removal due to flow, and the ratio of heat loss rate to heat capacity. (Page 6-8)

SONGS. A submittal to NRC pertinent to Appendix R has plant cooldown information and includes detail pertinent to pressurizer behavior for SONGS 1. For example, initial temperature was taken as 643 ^{o}F, and the temperature at 2.1 hrs was 640 ^{o}F. Pressurizer water level was taken as a constant. Further information can be obtained from the letter of May 21, 1985, calculation number 0310-027-1372, the table on page 42 of 84 and Section 3.4, page 15 of 84.

8. STAFF. What RCS leak rate was assumed for the six cases identified on Page 6-9?

   SONGS. Zero.

9. STAFF. Item 6 on Page 6-10 indicates no credit for charging or letdown. Was letdown assumed to be terminated?

   SONGS. There was zero charging and zero letdown. No credit was taken for the pressurizer control system.

10. STAFF. Page 6-14 contains the statement "However, SG inventory is slowly increasing and would condense any voids generated" (in the RCS). Since the RCS appears to be saturated and there is no charging, how will this be accomplished? Stated a different way, is there sufficient inventory in the RCS that voids can be condensed without loss of pressurizer level?

    SONGS. The pressurizer inventory is sufficient to allow for condensation of voids in the RCS without loss of pressurizer inventory. However, voids would remain in the upper head if RCPs were not running.

11. STAFF. Item 2 on the same page references no bulk boiling in the hot leg. Does this mean no two phase condition? Stated differently, it is difficult to understand how bulk boiling can take place in the hot leg since there is no source of energy, unless one means the term to apply to a depressurization condition where there is flashing and energy is considered to flow from the metal pipe into the fluid. What is meant?

SONGS. No bulk boiling in the hot leg means no loss of subcooling and no bulk boiling in the RCS.

12. STAFF. Bleed and feed cooling via the charging pumps and the pressurizer PORVs is identified on page 6-19. What is the capacity of the charging pumps and the PORVs? What is the meaning of the assumption of double-ended breaks in the SI lines?

SONGS. The flow rate for one charging pump is 315 gpm at 2000 psig and 230 gpm at 2200 psig. There are two charging pumps. PORV setpoints are approximately 2200 psi and the pressurizer safety valves are set to approximately 2500 psi. PORV steam flow rate capacity is 30 lbs/sec per valve. Charging pump shutoff head is approximately 2460 psi.

STAFF RESPONSE. The last sentence of item 12 was deleted by the staff since the answer is obvious.

13. STAFF. Page 6-19 has the loop mixture level at the hot leg elevation with steam relief via the PORVs at 1900 sec, yet page 6-20 has the upper head beginning to void at 2400 sec. Are these correct? If so, please explain how the level has dropped to the hot legs without voiding the upper head.

SONGS. The timing is correct. It may be useful to consider the upper head as draining rather than voiding, and the only way the upper head can drain is via the guide tubes. Westinghouse advises that steam binding occurs and prevents initiation of draining at the time of initial core bulk boiling. It also takes some time for the upper head to completely drain.

14. STAFF. The boundary conditions for the RETRAN simulation referenced on page 6-21 are not clear. Please clarify.

SONGS. These are further clarified in an internal report which describes the analyses in detail. A copy will be provided to the staff.

STAFF RESPONSE. The staff received a copy of the report on May 8, 1986 (Ting, Y. P., "RETRAN Analysis of SONGS1 11/21/85 Water Hammer Event", internal Southern California Edison Co. report, February, 1986.). This contained all of the information needed by the staff.

15. STAFF. The staff does not understand the discussion on page 6-22 with respect to starting RCP B which resulted in decrease of T_{hot} and T_{ave} which reduced heat transfer from the primary to secondary systems. Steam flow rates are stated to have fallen to near zero and this is stated to have occurred due to the modeling of the header as one volume and a SG as another volume. Please discuss.

SONGS. Further consideration of the behavior has led to the conclusion that the probable cause of decreased steam flow is introduction of cold (70^{0}F) AFW which depresses the pressure. Starting RCPs increases heat flow rate from the primary to secondary sides, as does the introduction of cold AFW.

REFERENCE 2: Letter dated May 1, 1986, M. Medford (SCE) to G. Lear (NRC)

16. STAFF. Page 6-13 of the previous reference contains a discussion of a MFLB which is stated to be conservatively modeled with no equipment failures. The electrically driven AFW pump appears to trip at run-out at roughly one minute. Operator action is assumed at ten minutes to restart the pump, and five minutes are assumed for refill of the feed lines under these conditions. What is the feedline volume and what is the actual time required to fill that volume consistent with the other analysis assumptions? What are the comparable assumptions with respect to the MFLB in the May 1 analysis? Operation of the pump at 375 gpm appears to be close to runout conditions. What is the substantiation that the pump can be operated for an extended time at this condition? (Emphasis should be upon test data.)

SONGS. The electrically driven AFW pump trips at a pressure of 675 psig which corresponds to a flow rate somewhat in excess of 375 gpm, which is achieved at 775 psig. (The original approach was to use 400 gpm rather than 375, but the pump manufacturer expressed concern that 400 gpm might result in pump instability. The manufacturer stated that there would be no problem at 375 gpm.) Thus, AFW trip can be expected under any condition which causes rapid depressurization of the SGs. Of course, loss of SG pressure also results in loss of the turbine driven AFW pump.

Operator response to AFW trip on runout is to assume manual control and close all AFW flow control valves. Each is then opened slowly until a flow rate of 125 gpm is attained in the AFW pipe leading to each of the three SGs. Of course, if the affected SG is identified, one would not continue to send 125 gpm to that SG, but identification is not a requirement of the response.

For analysis purposes, the assumption is made that all three feedwater lines are empty. The total volume is approximately 1500 gal (about 73 ft^3/line on average), and thus it takes about 5 min to refill the lines of the unaffected SGs. The A and C SG lines each take about 4 min, whereas SG B requires about 6 min due to its longer length.

The analysis described in Reference 2 is based upon AFW restart at 15 min, with effective flow to the SGs at 20 min, with 5 min allowed for feedline refill.

The pump has not been extensively tested at the SONGS 1 plant. A bench test is presently under way to establish pump characteristics, but this will not provide long term operational information. Additional information in regard to the test will be provided to the staff.

17. STAFF. Does SONGS 1 have an auxiliary pressurizer spray capability? (This question is relative to Item C where the sprays are assumed available, but the RCPs have been tripped.) Were sprays assumed to be used in the analysis?

SONGS. The plant has auxiliary spray capability. Sprays were assumed used when available since this adds water to the pressurizer and therefore is a conservatism when one is concerned with filling the pressurizer. Normal operation is spray initiation at a pressure of 2125 psi, with maximum spray rate at 2175 psi. Approximately 40 ft^3 of water was added in the analysis of interest here.

Spray was not considered in the accident analysis. The behavior evident in some of the figures that is similar to what one would expect from spray operation is due to perturbations such as from charging, letdown, and pressurizer heater operation. Note the maxima in the curves is slightly below the point of spray initiation.

18. STAFF. Please expand on the statement that "1979 ANS 5.1 Decay Heat is modelled" and include sufficient information that an independent analyst could reproduce the results.

SONGS. A finite operating period corresponding to three cycles, with 430 effective full power days/cycle, was used. Energy from heavy elements was included.

19. STAFF. Are the SONGS PORVs considered to be safety related?

SONGS. No. No credit is taken for use of PORVs in design basis calculations. Where they are used, they are used because they make the transient worse.

PORVs are used for RCS feed and bleed, which is beyond the plant basis. Note the PORVs and block valves are air operated, with nitrogen backup in case of loss of air. The PORVs fail closed and the block valves fail open. The nitrogen system is seismically qualified. Battery backup for controls is provided in case of loss of all AC power.

20. STAFF. The feedline rupture analysis text indicates that RCS subcooling is not lost, but Figure 2-3 does not appear to support this conclusion.

Please explain. If the calculations were run further in time, it would appear that subcooling may be lost in the upper head, with voiding following. Please address.

SONGS. This should reflect that subcooling may be lost, but this occurance will not happen prior to achieving turnaround in which AFW flow rate is sufficient to more than compensate for the heat generation rate. Thus, subcooling will be regained later in the transient. The staff observations in regard to upper head voiding are correct.

21. STAFF. Figure 7 of Reference 1 shows the pressurizer liquid volume as essentially constant at approximately 1300 ft^3 for a finite time. During a portion of this time, the pressurizer safety valves appear to be open due to a constant pressure of 2500 psia being presented in the figure. Does this mean the pressurizer is liquid full and the valves are passing liquid or a two phase mixture? If so, are the valves qualified for this duty? Use of the PORVs under similar conditions should also be addressed. (See, for example, page 6-19 where the discussion identifies water relief via the PORVs.)

SONGS. These are older analyses and the staff should address the Reference 2 analyses as representing SONGS 1 for purposes of this review. The new analyses clearly show that no water will be passed through the relief valves under the transient conditions being represented here.

The Figure 7 computer results have been carefully studied by Westinghouse, who performed the analyses. These show that the pressurizer reaches a level corresponding to 1299 ft^3 and therefore no liquid is passed since the pressurizer is not filled. The termination of pressurizer volume increase is stated to be due to turnaround of the temperature increase.

The PORVs have been tested for relief of steam, two phase mixture, and water at both low and high pressures. They are qualified for relief at these conditions.

STAFF RESPONSE. The staff continues to believe that the 1299 ft^3 value may be something the computer does to represent a full pressurizer in the numerical sense, and that this value should not be used to conclude that the pressurizer is not full and that no liquid is being passed. The staff believes that the two discontinuities in the first derivative of the pressurizer inventory curve support this view. A temperature turnaround is not expected to result in these discontinuities.

However, the staff agrees with the licensee that the new analyses should be used for the evaluations pertinent to the November 21 accident. This, and the ability of the PORVs to operate under the conditions of concern, makes this question academic. It is not necessary to pursue this item further as part of the SONGS 1 review.

22. STAFF. Figure 7 shows a value of 15 min associated with a flow rate of 250 gpm. Is this the time of initiation of AFW flow or the time of flow becoming effective after filling the feed lines?

    SONGS. The latter. AFW was actually initiated at 10 min, and approximately 5 min are required to fill the feed lines.

23. STAFF. Figure 6.1-3 of Ref. 1 shows increasing and decreasing pressure behavior in the vicinity of 30 to 45 minutes. Does this correspond to operation of pressurizer sprays?

    SONGS. See Item 17 response.

24. STAFF. Does SONGS 1 have auxiliary spray capability?

    SONGS. Yes.

25. STAFF. Were there any differences in initial conditions between the loss of all AC power analyses described in Section 6.1.2.1 and the LMFW analyses?

SONGS. No.

26. STAFF. Were the Section 6.1.2.2 analyses previously approved by the staff?

SONGS. Yes.

27. STAFF. What is the relationship of feed line ruptures to the design basis of SONGS 1?

SONGS. These are not identified in the SER and are not part of the design basis for Unit 1.

28. STAFF. What RCP heat input rate into the fluid is used for cases 1 - 6?

SONGS. 5 MW/pump. Cases 1 and 2 of the November 1981 submittal to NRC were based on 3 MW/pump. Actual measured pump heat is 2.3 MW/pump.

29. STAFF. Page 6-10 contains "AFW flow was insufficient to match the SV steam flow necessary to remove decay heat and RCP heat and SG water levels continued to drop." Please clarify.

SONGS. This may be clearer as "... RCP heat, and therefore, SG ...."

30. STAFF. Were steam line break analyses performed for large breaks?

SONGS. Yes.

31. STAFF. What are the boundary conditions for the analyses with RETRAN which are identified on page 6-21?

SONGS. See Item 14 response.

32. STAFF. Is the design of the SG such that AFW sprays on the tubes, or does it flow into the region of the bottom of the tubes?

SONGS. AFW flows into the tube region from the bottom of the downcomer and hence does not spray onto the tubes. Heat transfer depends upon partial filling of the SG so that the bottom of the tubes is in contact with water.

33. STAFF. What is the rated flow for the AFW pumps?

SONGS. The motor driven AFW pump rated flow rate is 208 gpm and the turbine driven pump is rated at 235 gpm. The new AFW pump has a rated flow rate of approximately 400 gpm.

34. STAFF. Are the original Cases 2 - 6 limiting?

SONGS. No. The analyses only establish that these are workable situations. No attempt was made to establish limiting cases as part of that analysis program.

35. STAFF. What is the basis for the 10 min value for initiation of AFW in Case 6?

SONGS. This can be considered as a short duration station blackout.

36. STAFF. What steam conditions were assumed for analyses of the line breaks involving the steam generators and was superheated steam considered?

SONGS. Dry steam was assumed for the breaks. Superheat has not been addressed with respect to RCS response, although it has been addressed with respect to equipment qualification.

37. STAFF. Please provide additional information pertinent to Cases 1 and 2 of Reference 2.

SONGS. These two cases are closer to reality than the prior Case 1 - 6 submittal. Note the main feed line break is still not a design basis accident.

The 30 minutes time for initiation of AFW in the main feed line break is considered as bounding with respect to the prior analyses. This actually assumes the operator takes action at 29 minutes, and full flow results at 30 minutes. (SONGS 1 personnel assume one minute maximum for the motor driven pump to achieve full flow. The actual time is approximately 45 sec.) Thirty minutes is considered to be the maximum time available for the operator to respond for this event.

The main feed line break value of 20 minutes is the time at which full flow is achieved which continues for as long as it is desired by the operator. In actuality, there is some flow early in the event, and then flow is lost. The operators are assumed to start the AFW pump 15 min from the initiation of the event, and useful flow is achieved in 20 min following filling of the feed lines.

Note the feed and bleed analyses of the RCS show that approximately 50 min are available prior to core uncovery. The above analyses are based on not voiding the hot leg.

38. STAFF. What is the status of single failure with respect to steam line breaks?

SONGS. A schedule of meeting single failure criteria for steam line breaks has been agreed to between SONGS and the staff. This shows meeting the requirements prior to power in cycle 10. The upcoming cycle is cycle 9.

39. STAFF. What is the status of the third AFW system with respect to use of RCS feed and bleed as an emergency heat removal mechanism?

SONGS. Existing operator instructions and training are to use any possible mechanism to provide water to the steam generators prior to resorting to RCS feed and bleed. This, the freshness of the November 21 event, and recent addition of the third AFW system, would result in use of the third AFW system prior to feed and bleed.

Use of the third AFW system prior to use of feed and bleed will be specifically incorporated into emergency procedures early in the upcoming operation cycle.

40. STAFF. Several serial number 100 valves are associated with the SG blowdown system. Why are some of these valves used for automatic isolation and others not? (See Figure 4-5, page 4-23 of Reference 1.)

SONGS. Valve CV 100B is in series with valve CV 100A, and therefore it is not necessary that both be actuated to obtain isolation.