VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

October 12, 1995

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Serial No. 95-474
NL&OS/ETS
Docket Nos. 50-280
50-281
License Nos. DPR-32
DPR-37

Gentlemen:

**VIRGINIA ELECTRIC AND POWER COMPANY**
**SURRY POWER STATION UNIT NOS. 1 AND 2**
**INSERVICE TESTING PROGRAM ANOMALIES**

The NRC safety evaluation report for Surry Power Station's ASME Section XI Inservice Testing Program Plan was received by letter dated October 20, 1994. Appendix A to this report identified 14 anomalies in the program plan. Although a response was requested for only one of the anomalies (Anomaly 3), we have provided a response for each of the 14 anomalies. The responses are provided in Attachment 1 to this letter.

As a result of the anomalies identified in the NRC safety evaluation report, certain IST program changes have been made. IST program changes have also resulted from system configuration changes and from engineering evaluations of system and component function. These changes and the revised program pages are included in Attachments 2 and 3, for Surry Units 1 and 2, respectively.

If you have any questions concerning our response or IST program changes, please contact us.

Very truly yours,

James P. O'Hanlon

James P. O'Hanlon
Senior Vice President - Nuclear

Attachments

130104



AD47

cc: U.S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N.W.
Atlanta, Georgia 30323

Mr. Morris W. Branch
NRC Senior Resident Inspector
Surry Power Station

50-280 SURRY 1 VE&PCO

NRC SAFETY EVALUATION OF INSERVICE TESTING PROGRAMS REV. 0 RESPONSES TO 1st PROGRAM ANOMALIES

REC'D W/LTR DTD 10/12/95...9510200251

# ATTACHMENT 1

# NRC SAFETY EVALUATION OF SURRY UNITS 1 AND 2 INSERVICE TESTING PROGRAMS REVISION 0

# RESPONSES TO IST PROGRAM ANOMALIES

# RESPONSE TO
# IST PROGRAM ANOMALIES

1. NRC SER Anomaly The IST program does not include a description of how the components were selected and how testing requirements were identified for each component. The review performed for this Safety Evaluation (SE)/TER did not include verification that all pumps and valves within the scope of 10 CFR 50.55a and Section XI are contained in the IST program, and did not ensure that all applicable testing requirements have been identified. Therefore, the licensee is requested to include this information in the IST program. The program should describe the development process, such as a listing of the documents used, the method of determining the selection of components, the basis for the testing required, the basis for categorizing valves, and the method or process used for maintaining the program current with design modifications or other activities performed under 10 CFR 50 59.

   Virginia Power Response A description of the IST program development process for the third inspection interval is given below. This description will be included in the next change to the IST program.

   General Program Development

   ASME B&P Code, Section XI (hereby referred to as 'the Code') requires that the owner of each nuclear power plant prepare and submit a "plan" for testing and inspection of systems and components under the jurisdiction of the Code and in compliance with Title 10, Part 50 of the Code of Federal Regulations (Para. 50.55a). With respect to the elements of that plan related to the testing of pumps and valves, Section XI, Subsections IWP and IWV direct each licensee to comply with the applicable portions of ASME/ANSI OM Parts 6 and 10. The NRC directed via the Federal Register, Vol. 57, No. 152 dated August 6, 1992 that pump and valve testing should be performed in accordance with ASME/ANSI OM-1987 including OMa-1988 Addenda. Specifically, Part 1 of OM-1987 and Paragraphs 1.1 of OMa-1988 Addenda, Parts 6 and 10, establish the IST program scope with the provision that the rules apply only to ISI Classes 1, 2 and 3 as stated by the NRC in the Federal Register, Vol. 56, No. 21 dated January 31, 1991.

   In accordance with ASME/ANSI OM-1987 and OMa-1988 Addenda, the following are required to be included in the testing program:

   1) Centrifugal and positive displacement pumps that are provided with an emergency power source and required to perform a specific function in shutting down the reactor to the cold shutdown condition, maintaining the cold shutdown condition or mitigating the consequences of an accident.

2) Active or passive valves (and their actuating and position indicating systems) which are required to perform a specific function in shutting down the reactor to the cold shutdown condition maintaining the cold shutdown condition or mitigating the consequences of an accident.

3) Pressure relief devices that protect systems or portions of systems which perform a required function in shutting down the reactor to the cold shutdown condition maintaining the cold shutdown condition or mitigating the consequences of an accident.

In addition to the general Code requirements outlined above, there are other interpretations and positions that have come about as a result of past regulatory initiatives and guidance including Generic Letter 89-04 and NUREG-1482. Other than these guides, there is no specific guidance for developing the IST Program scope of testing. Therefore, a set of rules was established by which the scope of the Surry ASME Section XI IST Program is determined including components that are to be included and the extent and type of testing required for each. Based on these rules, the philosophy and assumptions used in determining the test requirements for selected pumps and valves were documented.

Program Scope

In the course of developing the Program scope, each of the significant safety systems included within the ISI Class boundaries and certain safety systems outside of the ISI Class boundaries (such as the emergency diesel fuel oil transfer system) were evaluated with respect to the function of each component and the need for its operability as it relates to the scope of Section XI. Supporting documents used include,

Final Safety Analysis Report (FSAR),
Technical Specifications,
Past program correspondence,
Operating Procedures (normal, emergency and abnormal) and
Plant System Descriptions.

The sequence followed during the development effort was as follows:

1) Each of the plant systems was subjected to an overview to determine any potential active safety function as described in the scope Statement. Those systems with no obvious safety functions were then excluded from further consideration. Plant documents as well as operating staff comments were utilized in this phase.

2) For the remaining systems, flow diagrams were studied and any component that could possibly have an active or passive safety function (other than simply

maintaining the pressure boundary) were identified for further evaluation.

3) The function of each component identified from the flow diagrams was determined based on available documentation, staff review or general experience of the evaluator. Testing requirements were derived based on the component function(s) and the applicable rule(s).

4) Available documents were reviewed and specific or implied component operational requirements were compared to the component functions.

5) The results of the steps described above were reviewed by several knowledgeable members of the plant staff and evaluated for accuracy and consistency, and compiled in an IST basis document. Based on this review, the final program scope was derived and the IST Program Plan developed.

Program Update

During the third 10-year interval it is expected that the scope of the Program will occasionally be modified in response to unrelated activities including, but not limited to:

1) Plant design changes,

2) Changes in operating conditions (e.g. normal valve lineup) and

3) Changes in accident mitigating procedures philosophy.

As a result, it is expected that the IST Program may be revised to ensure continued compliance with the Code requirements relating to the scope of the test program. The supervisor responsible for maintaining the IST Program is provided copies of all plant modifications that are designated by Engineering to have a potential IST/ISI impact. Should a change require a Program revision, the IST Coordinator would then implement the change to the Program Plan and the appropriate test procedure(s) in a timely manner.

2. NRC SER Anomaly Several of the licensee's relief requests (V-5, -20, -43, and -50) are approved by GL 89-04 and are not evaluated in this TER. The licensee indicates compliance with GL 89-04, but does not specifically address all aspects of the Generic Letter provisions in the requests. In these cases, it is assumed that the licensee is complying with all of the requirements of the applicable GL 89-04 positions. Relief is not granted for the above relief requests for testing that deviates from that prescribed in GL 89-04. Whether the licensee complies with the provisions of GL 89-04 is subject to NRC inspection. If the licensee intends to deviate from a GL 89-04 position, a revised relief request specifically stating the deviation from GL 89-04

guidance must be submitted for review and approval prior to implementing the testing. For example, it does not appear that valves 1(2)-SW-108, -113, and -130 in relief request V-50 meet the grouping criteria of GL 89-04, Position 2. Valves 1-SW-108 and -113 are Code Class 3 valves and are at the discharge of the charging pump service water pumps. Check valve 1-SW-130 is a non-Code Class valve that is in the combined return line to the circulating water discharge tunnel. The licensee should disassemble and inspect valve 1-SW-130 as a separate group or revise request V-50 to justify the deviation from the GL 89-04 grouping criteria.

Virginia Power Response The internals for valve 1(2)-SW-130 have been removed. Therefore, valve 1(2)-SW-130 no longer serves a safety function and was deleted from the IST program. Unless noted in the relief requests, Surry Power Station complies with the provisions in GL 89-04.

3. NRC SER Anomaly Valve relief requests V-5, -20, -41, -42, -43, -46, and -50 are for check valves that may not be practically verified closed using system pressure or flow or full-stroke exercised open with flow per GL 89-04, Position 1. The licensee proposes to full-stroke exercise these valves by sample disassembly, inspection, and a manual exercise. The NRC considers valve disassembly and inspection to be a maintenance procedure and not a test equivalent to the exercising produced by fluid flow. This procedure has some risk, which make its routine use as a substitute for testing undesirable when some method of testing is possible. Disassembly and inspection, to verify the full-stroke open or closure capability of check valves is not a recommended option when exercising can be practically performed by system pressure, flow, or other positive means. Check valve disassembly is a valuable maintenance tool that can provide much information about a valve's internal condition and as such should be performed under the maintenance program at a frequency commensurate with the valve type and service.

Some test method may be feasible to full-stroke exercise these valves. The licensee should consider methods such as using nonintrusive techniques (e.g., acoustics, ultrasonics, magnetics, radiography, and thermography) to verify a full-stroke exercise of the subject check valves. This testing may only be practical at cold shutdowns or refueling outages. The licensee should perform their investigation and if a test method is found to be practicable, the IST requirements of the applicable valves should be satisfied by testing instead of disassembly and inspection. If testing is not practicable and disassembly and inspection is used, it must be performed in accordance with GL 89-04, Position 2. The licensee should respond to this concern.

Virginia Power Response Relief request V-5 addresses closure testing the main feedwater isolation check valves 1(2)-FW-10, 41 and 72. The use of radiography to verify closure will be investigated. This investigation will be completed in 1996.

Relief request V-20 addresses check valves 1(2)-SI-47 and 56 on the suction line from the containment sump to the low head safety injection pumps. As stated in the relief request, flow cannot be established in the suction line without introducing untreated water into the safety injection system. Without flow there is no motive force to move the disk to the open position. Therefore, the only way to move the disk is by manual manipulation following disassembly.

Relief request V-41 addresses check valves 1(2)-FW-144, 159 and 174 on the auxiliary feedwater pump minimum flow lines and check valves 1(2)-FW-148, 163 and 178 on the auxiliary feedwater pump oil cooler lines. Techniques other than disassembly and inspection will be investigated. This investigation will be completed in 1996.

Relief request V-42 addresses check valves 1(2)-MS-176, 178 and 182 on the main steam supply header to the turbine driven auxiliary feedwater pump. Radiography techniques will be evaluated for testing these valves to the closed position. This evaluation will be completed in 1996.

Relief request V-43 addresses check valves 1-CS-13, 24, 105 and 127, 1-RS-11 and 17, 2-CS-13, 24, 104 and 105, and 2-RS-11 and 17 on the containment spray discharge lines. As stated in the relief request exercising these valves with flow will introduce spray to the containment.

Valves 1-CS-105 and 127, and 2-CS-104 and 105 do not have external lever arms. Therefore, the only way to move the disk on these valves is by manual manipulation following disassembly.

Valves 1(2)-CS-13, 24 and 1(2)-RS-11 and 17 do have external lever arms. As stated in the relief request the measurement of torque on these lever arms would not be repeatable from test to test. A combination of manipulating the disk with the lever arm and verifying a disk strike with acoustic techniques was considered. However, without trending per the Code requirements the torque necessary to move the disk, there is no assurance that flow can move the disk. Therefore, disassembly and inspection is the best method for verifying full-stroke capability.

Relief request V-46 (Unit 1) addresses check valves 1-SW-313, 323 and 2-SW-333 on the service water supply to the main control room air conditioning system. Flow instrumentation has been installed. Therefore, these valves will be full flow tested every quarter. Relief request V-46 (Unit 1) is being withdrawn.

Relief request V-46 (Unit 2) addresses the service water vent valves to the recirculation spray heat exchanges, 2-SW-247, 249, 251 and 253. The only time when these valves would open other than during an accident is when the recirculation spray heat exchanges are tested with flow. This test is performed on an infrequent basis (one train every other outage or both trains every fourth outage). Also, the valves may not open completely when the heat exchanges are tested because the valves open just enough to vent the air from the heat exchangers as they fill with water. Therefore, acoustic monitoring would not verify full-stroke. Disassembly and inspection is the best method for testing these valves.

Relief request V-50 addresses check valves 1(2)-SW-108 and 113, 1-SW-262 and 268, and 2-SW-442 and 445 on the discharge lines of the charging pump service water pumps. A full flow acceptance criterion has been determined for the charging pump service water pumps. Therefore, design flow can be verified for the discharge check valves. Relief request V-50 is being withdrawn.

4. <u>NRC SER Anomaly</u> In relief request P-1 (see Section 2.1.1.1) the licensee requests relief from the pump vibration measurement reference value requirements for all pumps in the IST program. The licensee proposes to set the vibration velocity reference values for pumps with a measured vibration velocity below 0.05 in/sec. at 0.05 in/sec. The alternative is authorized on an interim basis pursuant to 10 CFR 50.55a(a)(3)(ii) with the following provision. Prior to assigning the 0.05 in/sec. as a minimum reference value, the licensee should review each application and the manufacturers' recommendations to ensure that the proposed minimum reference value is appropriate. Once the OM Code Committee comes to a consensus and changes the Code with guidance for smoothly-running pumps, the licensee should adopt the guidance or develop and justify a reasonable alternative to the Code.

<u>Virginia Power Response</u> Surry Power Station will adopt the Code guidance for smoothly-running pumps when it is approved by the OM Code committee. The technical manuals provided by the various manufacturer's do not provide recommendations pertaining to minimum levels of vibrations. These pumps have been in service for approximately 20 years and have undergone refurbishment and maintenance. Knowledge concerning levels of expected and acceptable vibration levels lies at the site and not with the manufacturers. Polling the manufacturers for recommendations would not provide additional meaningful information.

The vibration test data has been reviewed for each pump. From this review we concluded that the vibration data does trend as expected and that there are no cases where rigidity of the foundation suppresses the pump vibration to the point that it is not trendable. As a matter of practice, the vibration data is frequently reviewed to identify adverse trends so that maintenance can be initiated before a pump enters into the alert or required action ranges.

5. <u>NRC SER Anomaly</u> In relief request P-11 (Unit 1) (see Section 2.3.1.1) the licensee requests relief from the reference value and differential pressure (d/p) measurement requirements for the emergency service water pumps. The licensee proposes to conduct tests of these pumps within the tide level limits of a pump reference curve. The pump flow will be compared to acceptance criteria based on the reference curve

and the ranges given in OM Part 6, Table 3b. Discharge pressure will not be measured. Testing using a reference curve can be acceptable if the seven elements listed in Section 2.3.1.2 are incorporated into the IST program and procedures for developing and implementing the curve(s). However, traditional curve testing does not appear to be feasible in this case, as this is a fixed-resistance system with no provision for varying d/p or Q.

Given that traditional curve testing may not be appropriate, and considering the limited information provided in the relief request regarding the licensee's approach to curve testing, the reviewer cannot fully assess the proposal and determine whether relief should be granted or the alternative authorized as provided for in 10 CFR 50.55a. The licensee's proposed testing gains information that can be considered to assess the operational readiness of these pumps. Therefore, their proposal provides an adequate alternative to the Code requirements for an interim period. Interim relief should be authorized pursuant to 10 CFR 50.55a(f)(6)(i) for a period of one year or until the next refueling outage, whichever is longer. Relief should not be authorized beyond that point. By the end of that period the licensee should either comply with the Code or develop and a method of monitoring the condition of these pumps that provides a reasonable alternative to the Code. The proposed method, if different than the Code should be submitted to the NRC for review and approval.

Virginia Power Response The request to only measure suction pressure and not discharge/differential pressure is being withdrawn from P-11. Differential pressure is currently calculated and will continue to be calculated using the measured tide level and discharge pressure.

Surry Power Station complies with the elements discussed in the SE for using a pump curve instead of a point on the curve, except for element c which requires a minimum of five points to establish the reference curve. The request to use three points instead of a minimum of five points has been addressed by the NRC. By letter dated October 22, 1993 (TAC NOS. M86961 AND M86962), the NRC agreed with our position that three points are adequate in the case of the emergency service water pumps. The NRC conclusion was based on additional information sent by Virginia Power by letter dated June 29, 1993 (Serial No. 93-206).

Quoting from the Virginia Power letter,

> "Relief Request P-11 for the emergency service water pumps describes the difficulty of performing the quarterly ASME test for these pumps. The hydraulic test loop is a fixed resistance system which is affected by tide level. To eliminate waiting for the proper tidal conditions, use of a pump curve based on the changing tidal conditions has been developed. The NRC evaluation concluded that the use of a pump curve is acceptable, provided that certain elements are incorporated into the IST Program. One of these

elements required by the SER was that the curves be based on an adequate number of points, with a minimum of five. If defining the whole pump curve were required, we would agree that at least five points would be necessary to adequately establish the pump curve. However, the portion of the pump curve affected by the change in tide level is small. A maximum change of $\pm 3$ feet in tide level produces approximately a 1000 gpm change in flow. The reference value is approximately 17,000 gpm. This portion of the pump curve can be adequately represented by three points instead of five.

Requiring that the pump curve be regenerated with a minimum of five points after every major maintenance activity would be an unnecessary burden because the curve cannot be generated during one test as with most pumps. High and low tides occur approximately eleven hours[1] apart. Not only must the high and low tide points be determined, but points in between must also be generated over a long period of time. Given the small portion of the pump curve affected and the difficulty in gathering the data, three data points would be more then adequate to develop the pump curve. Based on this additional information, Surry Power Station requests concurrence from the NRC to use three points to develop the appropriate portion of the pump curve for the emergency service water pumps. Relief Request P-11 applies only to Surry Unit 1."

Note 1: The actual duration between high and low tides is six hours. The original text was in error. However, the time of six hours between tests still presents a significant burden when gathering data for the pump curve.

The NRC letter concluded that,

"Relief request P-11 seeks approval to use a pump curve that accounts for changes in tide level to eliminate delays during testing in waiting for the proper tide level. You proposed to use three testing points to modify a small portion of the pump curve affected by changes in the tide level. Based on our review, we conclude that the use of three data points to modify the appropriate portion of the pump curve to account for changes in the tide level is acceptable."

Based on the response in the October 22, 1993 NRC letter, we consider the use of three points to be approved by the NRC. There was no mention of the June 29, 1993 letter from Virginia Power or the October 22, 1993 letter from the NRC in the current NRC SE. Therefore, we conclude the NRC reviewer was unaware of this correspondence.

Another area of concern addressed in the current NRC SE is that,

> "when using reference pump curves, it may be more difficult to identify instrument drift or to trend changes in component condition."

Surry Power Station has the ability to normalize the test data and trend the data from test to test. By knowing the polynomial equation that describes the reference pump curve discussed in P-11, a reference value can be calculated for the dependent variable using the value of the independent variable. The actual test result is divided by the reference value to yield a normalized test result which can then be used to more easily identify instrument drift or to trend the changes in component condition.

As described above, differential pressure will continue to be calculated using the measured tide level and discharge pressure. With the calculated differential pressure as the independent variable, a reference flow is determined (the dependent variable) from the pump curve. The measured flow is then compared to acceptance criteria based on the reference flow. This method of monitoring provides a reasonable alternative to the Code.

6. <u>NRC SER Anomaly</u> In relief request P-16 (Unit 1) (see Section 2.4.1.1) the licensee requests interim relief from the requirements to measure and evaluate Q for the main control room (MCR) air conditioning pumps. The licensee proposes to determine pump Q by measuring the d/p across the chiller condensers. The alert value will be set at 10% above the minimum Q (240 gpm) or 264 gpm. The pump will be declared inoperable if a Q of 240 gpm cannot be achieved via system adjustments. Additionally, if a pump discharge pressure of at least 30 psig cannot be achieved with a shut backwash valve, the system and pump will be investigated. Inlet pressure and Q measuring instruments will be installed by the end of the Unit 1, cycle 12 refueling outage, which is scheduled for the second quarter of 1994. Once the modifications are made, the licensee will comply with the Code requirements. Interim relief should be authorized pursuant to 10CFR50.55a(f)(6)(i) for a period of one year or until the next refueling outage, whichever is longer. Relief should not be granted beyond that point. By the end of that period the licensee should either comply with the Code or develop and implement a method of monitoring the condition of these pumps that provides a reasonable alternative to the Code.

   <u>Virginia Power Response</u> By letter dated June 2, 1994 (Serial No. 94-324), Virginia Power indicated that extensive testing of the newly installed ASME Section XI flow instrumentation revealed that the instruments were not reading full flow and that an engineering effort had been initiated to redesign the instrument configuration. The letter also requested an extension of P-16 to the end of the next Surry Unit 1 outage, currently scheduled to start in the third quarter of 1995. The NRC agreed to the extension by letter dated July 20, 1994 (TAC NO. M89859). The flow

instrumentation has been installed. Therefore, relief request P-16 is being withdrawn.

7. NRC SER Anomaly In relief request P-17 (Unit 1) (see Section 2.4.2.1) the licensee requests relief from the requirement to obtain reference values at repeatable points of operation for the MCR air conditioning system chilled water circulating pumps. The licensee proposes to use a straight line approximation method to determine d/p reference points as a function of flow between the two test points. The measured d/p will be compared to the upper required action limit which is set at 110% of the reference d/p ($P_{rdiff}$), and the lower required action limit at 90% of $P_{rdiff}$. No alert range will be assigned. Relief should be granted from this Code requirement pursuant to 50.55a(f)(6)(i) with the following provision. The licensee should follow the seven guidelines identified in Section 2.3.1.1.2 of this report for using reference curves, if practicable. Where it is not practicable to follow these guidelines, the licensee should identify the specifics of their alternative and justify the deviations and show the adequacy of their proposed testing.

Virginia Power Response Surry Power Station complies with the elements discussed in the SE for using a portion of the pump curve instead of a point on the curve, except for element c which requires a minimum of five points to establish the reference curve. The NRC reviewer expressed concern in Section 2.4.2.1 that,

> "For any shape of curve, the acceptance criteria at the endpoints will be the most accurate, however, for a convex downward curve, the acceptance criteria based on the straight line will be generally less conservative than at the actual reference points."

The endpoints for the straight line approximation are from approximately 240 gpm to 270 gpm, which represents less then 11% of the total curve. This portion of the pump curve is to the far right on the curve. A review of the manufacturer's pump curve shows that between the points of interest the curve is almost a straight line and the difference between a straight line and the curve at the midpoint is visually undetectable. Also, the change in differential pressure over the flow range is only 5 psid as compared to typical pump differential pressures over the same flow range of between 91 to 96 psid, which makes the error introduced at the midpoint even less significant.

The NRC reviewer expressed another concern that, "it may be more difficult to identify instrument drift or to trend changes in component condition." Surry Power Station has the ability to normalize the test data and trend the data from test to test. By knowing the equation of the line between the two points, a reference value can be calculated for the dependent variable using the value of the independent variable. The actual test result is divided by the reference value to yield a normalized test result which can then be used to more easily identify instrument drift or to trend the changes in component condition. Therefore, for the reasons given above, the straight line approximation provides an adequate alternate to element c in the NRC evaluation.

8. NRC SER Anomaly In relief requests P-16 (Unit 2) and P-19 (Unit 1) (see Section 2.5.1.1) the licensee requests interim relief from the requirement to measure flow and differential pressure at repeatable points of operation for the component cooling pumps. The licensee proposes to use a straight line approximation method to determine d/p reference points as a function of flow between the two test points. The measured d/p will be compared to the upper required action limit, which is set at 110% of $P_{rdiff}$, and the lower required action limit at 90% of $P_{rdiff}$. No alert range will be assigned. Relief should be granted from this Code requirement pursuant to 50.55a(f)(6)(i) with the following provision. The licensee should follow the seven elements identified in the preceding paragraph for using reference curves, if practicable. Where it is not practicable to follow these guidelines, the licensee should identify the specifics of their alternative and justify the deviations and show the adequacy of their proposed testing.

Virginia Power Response As a point of clarification, Surry Power Station is requesting permanent relief to use a straight line approximation and not interim relief as indicated in Anomaly 8.

Surry Power Station complies with the elements discussed in the SE for using a portion of the pump curve instead of a point on the curve, except for element c which requires a minimum of five points to establish the reference curve. The NRC reviewer expressed concern in Section 2.5.1.1 that,

> "For any shape of curve, the acceptance criteria at the endpoints will be the most accurate, however, for a convex downward curve, the acceptance criteria based on the straight line will be generally less conservative than at the actual reference points."

The endpoints for the straight line approximation are from approximately 8600 gpm to 9500 gpm, which represents less then 10% of the total curve. This portion of the pump curve is to the far right on the curve. A review of the manufacturer's pump curve shows that between the points of interest the curve is slightly convex downward. However, the difference between a straight line and the curve at the midpoint is so small as to be visually unquantifiable. Also, the change in differential pressure over the flow range is less then 6 psid as compared to typical pump differential pressures over the same flow range of between 86 to 92 psid, which makes the error introduced at the midpoint even less significant.

The NRC reviewer expressed another concern that, "it may be more difficult to identify instrument drift or to trend changes in component condition." Surry Power Station has the ability to normalize the test data and trend the data from test to test. By knowing the equation of the line between the two points, a reference value can be calculated for the dependent variable using the value of the independent variable. The actual test result is divided by the reference value to yield a normalized test result which can then be used to more easily identify instrument drift or to trend the changes in component condition. Therefore, for the reasons given above, the straight line approximation provides an adequate alternate to element c in the NRC evaluation.

9. NRC SER Anomaly V-47 requests relief from the stroke time measurement method and acceptance criteria requirements of OM-10 for the listed valves and proposes to measure the stroke times of these valves by observing the valve stems locally and not apply the acceptance criteria of Paragraph 4.2.1.8. The licensee's proposed testing provides a relatively inaccurate measurement of valve stroke times. The measurement inaccuracies and relaxed acceptance criteria decrease the likelihood of detecting degradation unless the valve is sufficiently degraded that the limiting value of full-stroke time is exceeded. Position 5 of GL 89-04 provides guidance for developing limiting values of full-stroke times for power-operated valves. If these limits are not set in accordance with these guidelines, the proposed testing is incapable of detecting degradation and is unacceptable.

If not already done, the licensee should investigate alternate testing methods that could provide more accurate measurements of stroke times for these valves. These methods could range from procedural changes (e.g., removing a power source to a controller or using calibration equipment to insert a signal that would cause a valve to move to its open or closed position at maximum speed) to using nonintrusive diagnostics to measure the stroke times (e.g., using a hall-effect probe or gauss detector to detect when current is interrupted to a solenoid valve coil and detect when the slug has moved from the seat into the coil). If a more accurate test method is found to be practicable, it should be employed to test the applicable valves. Valve diagnostic programs can yield significant information about the valve assembly, including the valve and actuator. When meaningful inservice testing is impractical, a periodic verification performed using valve diagnostic techniques may be an adequate alternative method for monitoring these valves for degrading conditions. Therefore, this alternative can ensure an acceptable level of quality and safety if the licensee has an established program of periodic diagnostic testing. (See Section 3.1.1.1).

Virginia Power Response Surry Power Station will continue to stroke the valves listed in V-47 as often as practicable because frequent exercising increases operational readiness particularly for the valves in the service water system. The service water valves are subject to sediment laden process fluid that tends to foul the working mechanisms of the valve. Valve diagnostic techniques (such as provided by the Air Cet system) are typically used either following maintenance or when an air operated valve fails the IST surveillance test. Therefore, valve stroke time is currently used as a gross indicator of valve health that can lead to a more sophisticated application of diagnostic techniques.

10. NRC SER Anomaly V-51 requests relief from the leak rate corrective action requirements of OM-10 for all CIVs in the IST program and proposes to allow an evaluation of leakage rates above the allowable limits instead of repair or replacement as long as the overall containment leakage is less than $0.6L_a$. The licensee did not provide details about the evaluation that would be performed. The evaluations should be performed in a manner that provides a high level of assurance that delaying the repair or replacement of valves with high leakage rates will not result in exceeding the $0.6L_a$ limit before the next leakage rate tests. The licensee should document in the program plan how these evaluations will be performed and what will be included (see Section 3.1.2.1).

Virginia Power Response An evaluation that returns a valve to service if it exceeds it's permissible leakage rate would typically include a determination of the cause for the leakage. The evaluation would also address the effect of the degradation mechanism for the valve on the ability of the containment to maintain overall leakage below $0.6L_a$ during the subsequent 24 month interval. Evaluations are documented in the plant records and are available for review.

11. NRC SER Anomaly V-26 requests relief from the test frequency requirements of OM-10 for the accumulator discharge check valves, 1(2)-SI-107, -109, -128, -130, -145, and -147, and proposes to verify a full-stroke exercise of these valves using nonintrusive techniques on a sampling basis during refueling outages. The licensee's proposed alternate testing appears to comply with most of the guidelines for using nonintrusives on a sampling basis in Section 3.2.1.1.1 of this report, however, it is unclear from the submittal if all of these conditions are met. The licensee should verify that the testing of the subject valves complies with all of these guidelines. The proposed grouping in this request does not appear to comply with the GL 89-04 requirement that group valves have the same service conditions. Valve 1(2)-SI-109 is the second check valve (closer to the RCS) in the injection line from the accumulator to the RCS while the other three group valves are the first check valves (closer to the accumulators). Differences in service conditions may affect the corrosion, erosion, wear, etc. for this valve such that it is not representative of the other valves in the proposed group. The licensee should justify the proposed grouping or bring it into compliance with the grouping criteria of GL 89-04.

Virginia Power Response Anomaly 11 questions the placement of the check valve 1(2)-SI-109 on the A loop in the same group as the three accumulator discharge check valves closest to the accumulator. It is assumed in the discussion that the valve closest to the RCS is the valve that normally seats against RCS pressure (i.e., valve 1(2)-SI-109), and is normally subjected to RCS pressure, water chemistry, and possibly elevated temperatures while the other valves in the group do not normally experience these conditions.

Valve 1(2)-SI-109 in the A loop was originally placed with the valves closest to the accumulators because the valves closest to the RCS on B and C loops experience RHR flow during shutdowns whereas the valve on the A loop does not. Therefore, it was concluded that valve 1(2)-SI-109 was subject to different service conditions.

However, if only normal operating conditions are considered, valve 1(2)-SI-109 should be placed in the group with the other two valves closest to the RCS. The other two valves 1(2)-SI-130 and 147 experience RHR flow only during shutdowns, which accounts for a small percentage of the total operating conditions.

The other alternative would be to place 1(2)-SI-109 in it's own group, which would require acoustic monitoring every outage. Numerous disassembly and inspections revealed that these check valves show virtually no degradation. Therefore, the difference in service conditions due to RHR flow during outages produces no additional detectable degradation in the valves in B and C loops. Given the lack of degradation observed in the valves during the disassembly and inspections, the added burden of acoustically monitoring valve 1(2)-SI-109 every outage is not justified. Therefore, this valve will be placed in the same group as valves 1(2)-SI-130 and 147.

12. NRC SER Anomaly V-27 requests relief from the exercising method requirements of OM-10 for the safety injection to RCS hot legs check valves and proposes to exercise these valves closed as pairs instead of individually at the frequency described in TS Table 4.1-2A. The licensee's proposed alternate testing appears to comply with several of the conditions for testing series check valves as pairs listed in Section 3.2.1.2.1 of this report, however, it is unclear from the submittal if all of these conditions have been met. The licensee should document their review of the plant safety analysis and the determination that only one of the two valves is credited in the safety analysis (that is, one valve could be removed without creating an unreviewed safety question or creating a conflict with regulatory or license requirements). The basis for the test acceptance criteria should also be documented. This documentation should be maintained on site for inspection by the staff.

Virginia Power Response The review of the plant safety analysis and the determination that only one of the two valves is credited in the safety analysis is documented and maintained on site for inspection by the staff. The test acceptance criterion for the back seat test is based on 5 gpm, which is the leakage limit for the Event V pressure isolation valves given in the Technical Specifications.

Separately, the issue of sensitized stainless steel affects the disposition of relief request V-27. In 1971 the United States Atomic Energy Commission set forth criteria to be followed regarding the use of sensitized steel piping. As a result of these criteria, the upstream valves 1(2)-SI-238, 239 and 240 were added during construction to provide double isolation between the RCS and sensitized stainless steel piping. However, the double isolation barriers were not designed to be individually tested and there is no requirement to test the integrity of each barrier.

13. <u>NRC SER Anomaly</u> In relief requests V-41, -42, and 46 (Units 1 and 2) the licensee proposes to disassemble and inspect the subject valves on a sampling basis at a refueling outage interval, but not necessarily during the refueling outage. These requests seek the option of performing the disassembly and inspection of the subject valves during power operations. There are several issues involved with this proposed test schedule:

- The acceptability of the refueling interval frequency.
- Entering a TS LCO action statement to perform PM.
- Appropriate corrective actions if sample disassembly is done during power operations.

Disassembly of a check valve may render the associated safety system train inoperable, which could result in entry into a TS LCO action statement. NRC Inspection Manual, Part 9900: Technical Guidance, titled "Maintenance - Voluntary Entry into Limiting Conditions for Operation Action Statements to Perform Preventative Maintenance" provides guidance on performing PM when the maintenance requires rendering the affected system or equipment inoperable. The NRC considers check valve disassembly and inspection to be an intrusive maintenance procedure and not a test. Even though an LCO action statement can be entered to perform surveillance testing, an action statement should not be entered routinely to perform PM activities unless it is justified in accordance with the NRC Inspection Manual, Part 9900. Therefore, if the proposed disassembly and inspection is to be performed during power operation and requires entry into an LCO action statement, the licensee should consider the following guidelines paraphrased from Part 9900:

a. There is a reasonable expectation that the on-line disassembly and inspection would improve safety by ensuring the operational readiness of the valves. The increase in reliability should exceed the effect of the increase in system unavailability.

b. The disassembly and inspection activity should be carefully planned to prevent repeatedly entering and exiting LCO action statements.

c. Other related equipment should not be removed from service during the performance of the on-line maintenance activity.

d. Maintenance should not be performed on-line unless confidence in the operability of the redundant subsystem is high. If equipment is degraded or trending towards a degraded condition in one train of a safety system, the redundant train should not be removed from service to perform on-line disassembly and inspections.

e. While performing an on-line maintenance activity, avoid performing other testing or maintenance that would increase the likelihood of a transient. There should be a reasonable expectation that the facility will continue to operate in a stable manner.

GL 89-04, Position 2, states "If the disassembled valve is not capable of being full-stroke exercised or there is binding or failure of valve internals, the remaining valves in that group must also be disassembled, inspected, and manually full-stroke exercised during the same outage." Until the operational readiness of the other group valves is verified by disassembly and inspection or by testing, their continued capability to perform their function should not be assumed. If a valve disassembled during power operation is found to be failed or excessively degraded, the licensee should immediately (before the end of the shift during which the failure is discovered) analyze the valve failure to determine the degradation mechanism and the likelihood that the other group valves are affected significantly by this mechanism. If the licensee's evaluation indicates that continued dependance on the operational readiness of these valves is not warranted, all group valves should be immediately declared inoperable and the appropriate TS required actions be followed. If the licensee determines that continued dependance on the operational readiness of these valves is warranted, the valves need not be immediately declared inoperable. However, all group valves should be disassembled and inspected or have their operational readiness verified by testing within the TS action statement time specified for one train of the safety system being inoperable. If the option of disassembly and inspection during power operations is be used, the licensee should document how the corrective action will be implemented for each specific valve group and the justification for performing this testing at a schedule different than the schedule approved by GL 89-04. The licensee's justification should consider the guidelines listed above as paraphrased from the NRC Inspection Manual, Part 9900. This documentation should be maintained on site for inspection by the staff.

<u>Virginia Power Response</u> The provision that, "If the licensee's evaluation indicates that continued dependance on the operational readiness of these valves is not warranted, all group valves should be disassembled and inspected or have their operational readiness verified by testing within the TS action statement time specified for one train of the safety system being inoperable" is impractical to implement. Because the provision given above is impractical and the operational readiness of the valves cannot be verified by testing, we are withdrawing the portion of the relief requests which deals with disassembly while the station is at power.

14. <u>NRC SER Anomaly</u> Valves 1(2)-SI-88, 91, 94, 238, 239, and 240 (refer to relief request V-27 and TER Section 3.2.1.2) are in the safety injection lines to the RCS hot legs. There are two of these valves in series in each injection line (e.g., 1-SI-88 and -238). These valves are ASME Code Class 1 and form the boundary with the connected ASME Code Class 2 piping. The systems that are connected to these

injection headers are the chemical and volume control (CVCS) and low head safety injection (LHSI) systems. Portions of those systems are low pressure (i.e., the relief valves on the LHSI headers are set to open at 220 psig). The subject valves are not identified as PIVs in the plant TS or in the licensee's response to Generic Letter 87-06 (GL 87-06), dated June 12, 1987. The GL 87-06 response identifies motor operated valves (MOVs) 1-SI-MOV-1869A, -1869B, -1890A, and -1890B as PIVs in the hot leg injection headers. These MOVs provide one barrier between the RCS and the interconnected low pressure systems. The licensee's response to GL 87-06 does not identify a second barrier in the hot leg injection headers.

The licensee's letter dated June 12, 1987, lists valves 1(2)-SI-79, 82, 85, 241, 242, and 243 as PIVs that are tested in accordance with the plant TS. These valves are in the injection headers to the RCS cold legs in an arrangement that is similar to the hot leg injection header valves in relief request V-27. It appears that the hot leg injection header check valves (1(2)-SI-88, 91, 94, 238, 239, and 240) perform an Event V pressure isolation function similar to the cold leg injection header check valves. The licensee should evaluate the function of these valves to determine if they perform a pressure isolation function and have been erroneously omitted from the TS and the GL 87-06 response.

Attachment 2 of the licensee's response to GL 87-06, dated June 12, 1987, lists several valves as PIVs that are not categorized A or A/C in the IST program and that are not leak rate tested to assure their leak tight integrity. Only three of these valves (1-SI-109, -130, and -147) are exercised to the closed position by the IST program, the remainder are exercised and stroke timed (for the power-operated valves) to only the open position. OM-10, Paragraphs 4.2.1.2(a) and 4.3.2.2(a) require valves to be exercised to the position(s) required to fulfill their function(s). Since these valves perform a pressure isolation function in the closed position (as identified in the licensee's response to GL 87-06), they should also be exercised to the closed position and the power-operated valves stroke timed to the closed position. The licensee should make the changes to the testing of these valves necessary to comply with the Code requirements or submit requests for relief where compliance is impractical or constitutes a hardship without a compensating increase in the level of quality and safety. In addition, the licensee should document their determination that the activities performed in lieu of leak rate testing these valves adequately assures the integrity of an independent barrier at the reactor coolant pressure boundary.

<u>Virginia Power Response</u> There are two statements in Anomaly 14 which need to be addressed before we discuss testing the valves identified as pressure isolation valves in our response to GL 87-06, and a third statement which needs clarification. The first statement is as follows.

> "The GL 87-06 response identifies motor operated valves (MOVs) 1-SI-MOV-1869A, -1869B, -1890A, and -1890B as PIVs in the hot leg injection headers. These MOVs provide one barrier between the RCS and

> the interconnected low pressure systems. The licensee's response to GL 87-06 does not identify a second barrier in the hot leg injection headers."

Valves 1-SI-MOV-1869A and B form the barrier to the high pressure high head safety injection (HHSI) system and not to a low pressure system as described above. Also, our response to GL 87-06 identified check valves 1-SI-226, 227, 228 and 229 in conjunction with valves 1-SI-MOV-1869A and B, and 1890A and B as the second barrier for penetrations 23, 60, 62 and 113. Refer to Figures 1 (Unit 1) and 2 (Unit 2). However, the response did not give any testing requirements, either a leak test or a back seat test, for these valves.

The second statement is as follows.

> "The licensee's letter dated June 12, 1987, lists valves 1(2)-SI-79, 82, 85, 241, 242, and 243 as PIVs that are tested in accordance with the plant TS. These valves are in the injection headers to the RCS cold legs in an arrangement that is similar to the hot leg injection header valves in relief request V-27. It appears that the hot leg injection header check valves (1(2)-SI-88, 91, 94, 238, 239, and 240) perform an Event V pressure isolation function similar to the cold leg injection header check valves."

As shown in Figures 1 and 2, the valve configurations are different in that the pressure isolation valves (PIVs) listed in our Technical Specifications isolate penetration 61 from the RCS which has a normally open upstream motor operated valve (1(2)-SI-MOV-1890C). Valves 1(2)-SI-88, 91, 94, 238, 239, and 240 are downstream from penetrations 23, 60, 62 and 113 which are isolated by the normally closed upstream motor operated valves 1(2)-SI-MOV-1869A and B and 1(2)-SI-MOV-1890A and B. This configuration is not an Event V configuration as confirmed by the Safety Evaluation Report for Surry Power Station entitled "PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES (WASH-1400, EVENT V)", dated April 20, 1981 which concluded that, "the valve configurations of concern have been correctly identified." There is no need for further evaluation of Event V configurations as suggested by the discussion in Anomaly 14.

The third statement that needs clarification is as follows.

> "Attachment 2 of the licensee's response to GL 87-06, dated June 12, 1987, lists several valves as PIVs that are not categorized A or A/C in the IST program and that are not leak rate tested to assure their leak tight integrity. Only three of these valves (1-SI-109, -130, and -147) are exercised to the closed position by the IST program, the remainder are exercised and stroke timed (for the power-operated valves) to only the open position."

There are 24 valves for each Surry unit identified in our response to GL 87-06 that fit the definition of PIVs in GL 87-06 but are not identified as PIVs in the IST program. Of these 24 valves, 11 are tested to the closed position due to their safety function as identified in the IST program. Three of the 11 valves (SI accumulator discharge check valves 1(2)-SI-109, -130, and -147) were thought to have a safety function to close after the accumulators discharged. A subsequent evaluation revealed that these valves have no safety function to close. Therefore, the closure test requirement for valves 1(2)-SI-109, -130, and -147 was deleted from the IST program as indicated in our letter to the NRC dated March 7, 1995 (Serial No. 95-108). The remaining eight valves of the 11 valves are motor operated containment isolation valves.

Including the three accumulator discharge checks 1(2)-SI-109, -130, and -147, there are 16 valves of the 24 that receive only an open test. These 16 valves are the six accumulator discharge check valves, the four high head safety injection (HHSI) to the hot and cold legs check valves (1(2)-SI-224, 225, 226 and 227), the two low head safety injection (LHSI) to hot legs check valves (1(2)-SI-228 and 229), and four residual heat removal system motor operated isolation valves (1(2)-RH-MOV-1(2)700, 1(2)-RH-MOV-1(2)701, 1(2)-RH-MOV-1(2)720A and 1(2)-RH-MOV-1(2)720B).

The current IST program does not identify the 24 valves mentioned above as PIVs nor does it invoke ASME Section XI, OM-10 testing requirements due to a reactor coolant pressure boundary function. The only PIVs included in the IST program are those listed in the Technical Specifications. These six check valves are on the low head safety injection lines to the cold leg paths as shown in Figure 1 and were identified during the review for WASH-1400 Event V valve configurations which was conducted in the early 1980s. As stated above, Surry Power Station received a Safety Evaluation Report dated April 20, 1981 which agreed with the results of the review. Because Surry was licensed before 1979, only the Event V PIVs were required to be in the Technical Specifications.

The valves identified in our response to GL 87-06 fit the definition of a PIV provided in GL 87-06. The definition is as follows.

> "Pressure isolation valves (PIVs) are defined for each interface as any two valves in series within the reactor coolant pressure boundary which separate the high pressure reactor coolant system (RCS) from an attached low pressure system. These valves are normally closed during power operation. The reactor coolant pressure boundary (RCPB) is defined in 10 CFR 50.2 and generally includes all PIVs."

However, in our response to GL 87-06 we did not assign any testing requirements to 24 non-Event V valves based on a reactor coolant pressure boundary function. It was never our intention when we responded to GL 87-06 to perform additional testing for these non-Event V valves. We disagree with the conclusion in Anomaly 14 that,

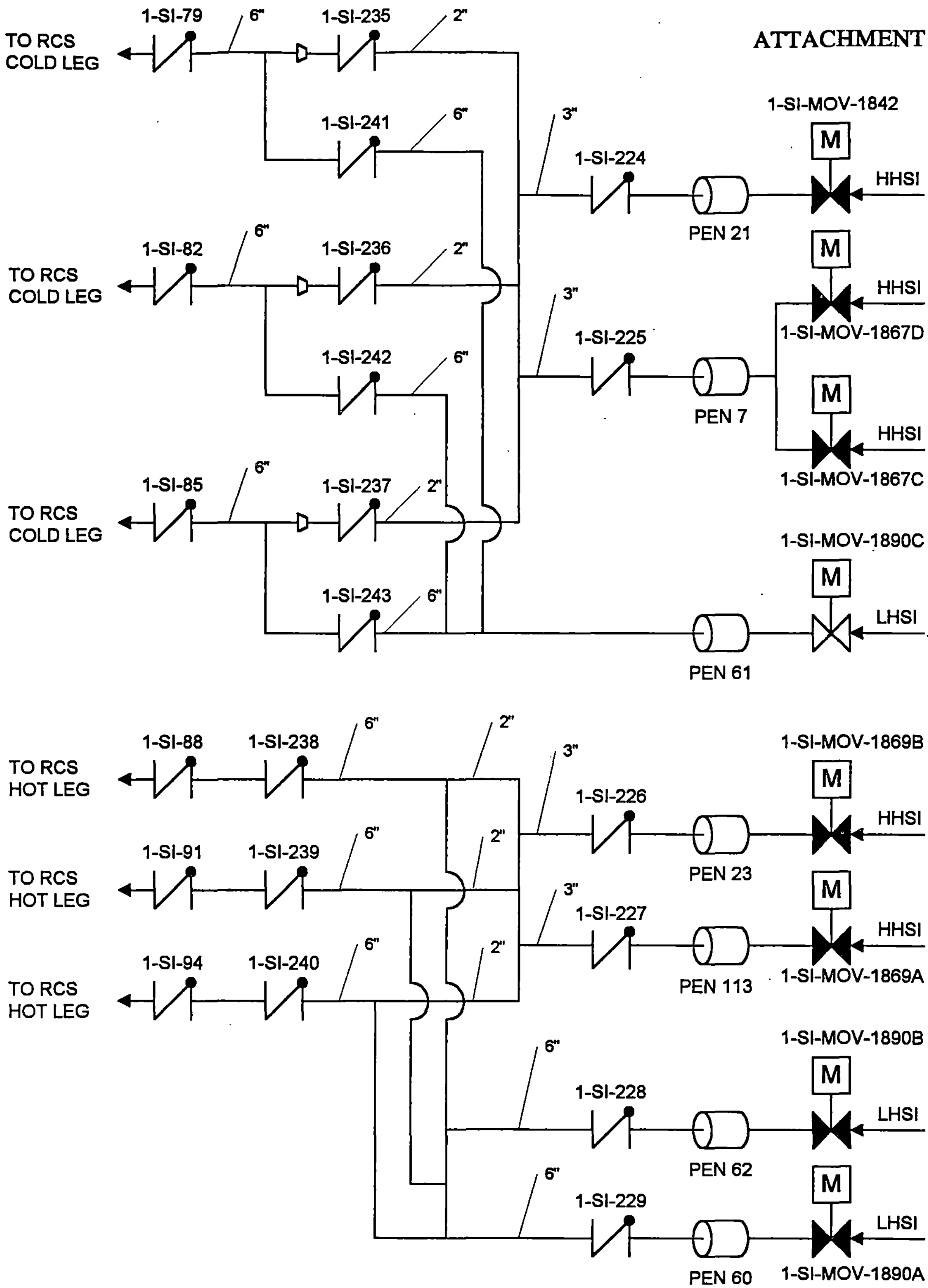
> "Since these valves perform a pressure isolation function in the closed

position (as identified in the licensee's response to GL 87-06), they should also be exercised to the closed position and the power-operated valves stroke timed to the closed position."

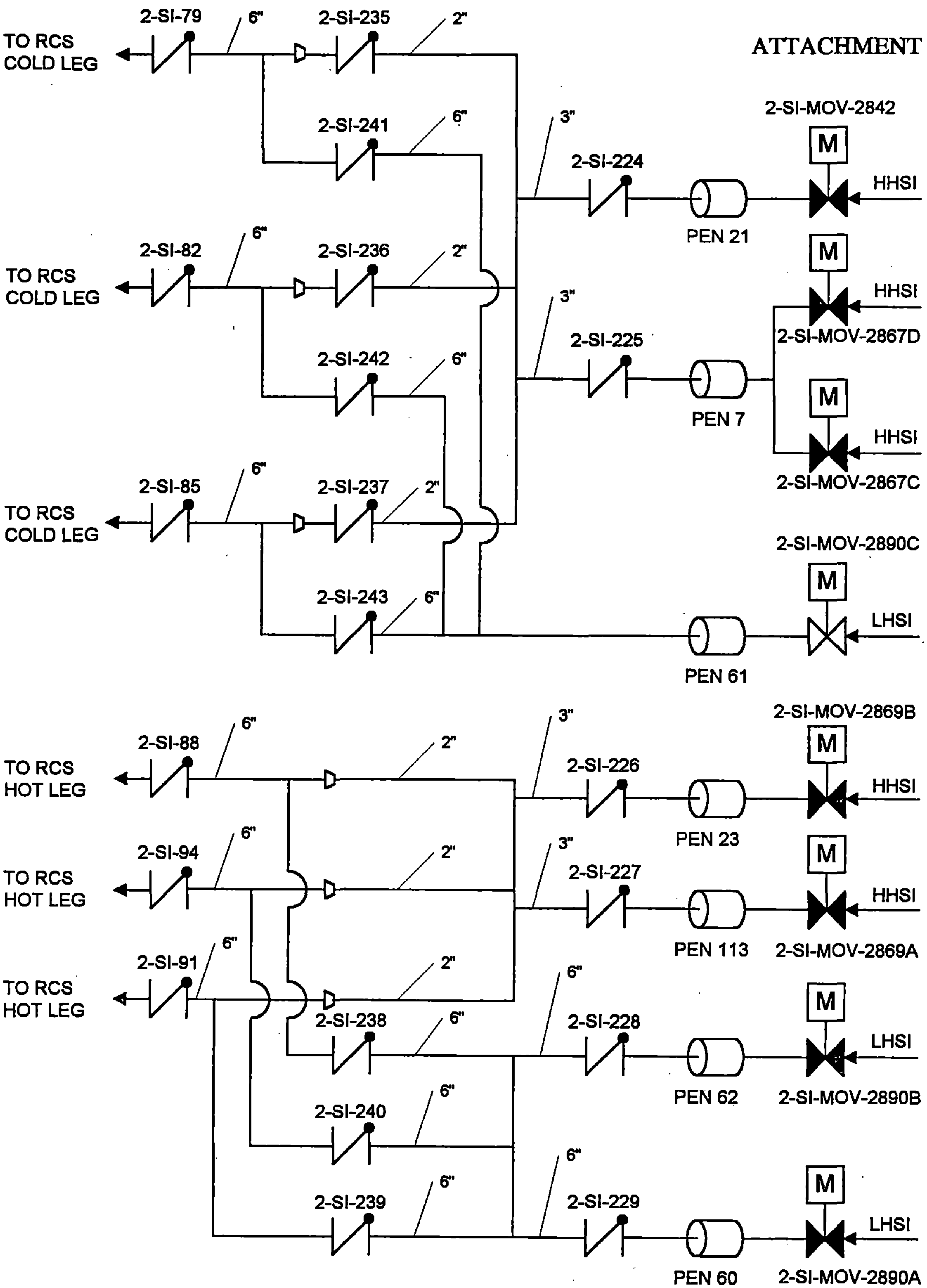
The licensing basis for Surry Power Station does not require testable, double isolation for the reactor coolant pressure boundary except for the Event V PIVs. The reactor coolant system is constantly monitored for leakage and the plant must shutdown if this leakage exceeds the limits given in the Technical Specifications. This constant monitoring ensures that the reactor coolant pressure boundary maintains adequate integrity without having to individually leak test or exercise to the closed position the boundary valves.

Also, we disagree with the philosophy stated above that passive valves need to be stroked. For example, the residual heat removal valves (1(2)-RH-MOV-1(2)700, 1(2)-RH-MOV-1(2)701, 1(2)-RH-MOV-1(2)720A and 1(2)-RH-MOV-1(2)720B) form the barrier between the high pressure RCS and the low pressure residual heat removal system. These valves are normally closed and passively fulfill their boundary function. As defined in Section 1.3 of the Code, OM-10, the closure function is passive because the valve obturator does not have to move. According to Table 1 in OM-10, there is no exercise requirement for testing a passive function.

As to the PIVs included in or excluded from the IST program, Surry Power Station complies with the guidance given in GL 89-04 and the minutes of public meetings on GL 89-04. Position 4 in GL 89-04 says that the Event V PIVs must be included in the IST Program. Response to question 27 in the minutes indicates that the Event V valves are a subset of the plant PIVs and that the "staff has recently undertaken a program to reevaluate various aspects of PIVs, including testing." Response 28 states that, "The responses to Generic Letter 87-06 are being used as input for the resolution of Generic Issue 105, 'Interfacing Systems LOCAs at Light Water Reactors,' under investigation by the NRC Office of Nuclear Regulatory Research." Based on these responses, we have concluded that the issue of testing the non-Event V PIVs would be addressed by the NRC as a generic industry wide issue. Therefore, for the reasons stated above, we disagree with the conclusion stated in Anomaly 14 that the non-Event V reactor coolant pressure boundary valves should be tested to the closed position.




SURRY UNIT 1
FIGURE 1




**SURRY UNIT 2**

**FIGURE 2**