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KNIGHTON,G. Licensing Branch 3

SUBJECT: Forwards revised responses to NRC Questions 210.1 & 440.3 re inter-sys LOCA surveillance requirements,NRC concerns not applicable to plants.Proposed amended FSAR pages provide justification for including operating limits in Tech Specs.

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Arizona Public Service Company

P.O. BOX 21666 • PHOENIX, ARIZONA 85036

February 24, 1983
ANPP-23084 - WFQ/KEJ

Director of Nuclear Reactor Regulation
Attention: Mr. G. Knighton, Chief
Licensing Branch No. 3
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Palo Verde Nuclear Generating Station (PVNGS) Docket Nos. STN-50-528/529/530
File: 83-056-026; G.1.01.10

References:
1. Letter from E. E. Van Brunt, Jr., APS, to R. L. Tedesco, NRC, dated September 2, 1981
2. Letter from R. L. Tedesco, NRC, to E. E. Van Brunt, Jr., APS, dated July 14, 1981.
3. Letter form E. E. Van Brunt, Jr., APS to R. L. Tedesco, NRC, dated September 28, 1981.

Dear Mr. Knighton:

Attached are revised responses to NRC question Nos. 210.1 and 440.3 (PVNGS FSAR question Nos. 3A.15 and 5A.14) concerning Inter-System LOCA Surveillance Requirements. APS has re-evaluated its commitments concerning these requirements and believes that the NRC's concerns are not applicable to PVNGS.

The attached proposed FSAR amended pages provide technical justification for not including the requested operating limits and surveillance in the PVNGS Technical Specifications. It should be noted that the commitment to perform ASME IWV-2000 category AC testing is not affected by this change.

The following lists how APS originally responded to the question concerning Inter-System LOCA Surveillance Requirements.

Reference (1) submitted the APS responses to the draft Mechanical Engineering Branch input to the PVNGS SER. Items 53, 54, 55 and 56 of that submittal responded to the subject surveillance requirements.

Reference (2) forwarded NRC question No. 210.1 to APS which restated the question from the draft MEB input concerning the subject surveillance requirements. NRC question No. 210.1 was considered to be responded to per Reference (1). Reference (3) submitted APS' response to NRC question's 440.3.

8001

If you have any questions concerning these changes, please contact me as soon as possible.

Very truly yours,

E. E. Van Brunt

E. E. Van Brunt, Jr.
APS Vice President,
Nuclear Projects
ANPP Project Director

EEVBJr/KEJ/sp

cc: E. Licitra (w/a)
D. Terao (NRC) "
K. Berlin "
A. C. Gehr "

February 24, 1983
ANPP-23084 - WFQ/KEJ

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, Edwin E. Van Brunt, Jr., represent that I am Vice President Nuclear Projects of Arizona Public Service Company, that the foregoing document has been signed by me on behalf of Arizona Public Service Company with full authority so to do, that I have read such document and know its contents, and that to the best of my knowledge and belief, the statements made therein are true.

Edwin E. Van Brunt
Edwin E. Van Brunt, Jr.

Sworn to before me this 23rd day of February, 1983.

Notary Public

My Commission expires:
My Commission Expires May 19, 1986

4. The space within the opening is sufficiently occupied by piping and pipe supports to preclude missile penetration.

The roof of the Main Steam Support Structure is elevated from the top of the walls to allow the escape of steam in the event of a major pipe break. The roof is cantilevered beyond the wall to provide the necessary missile protection.

<u>Question 3A.14</u> (NRC Question 450.2) (3.5.1.4)

Describe the protection of the control room air intakes and diesel generator exhaust pipes from tornado-generated missiles.

RESPONSE: 4

- The control room air intakes are enclosed within a box structure located within the Control Building (See figure 3A-7). The wall sections exposed to tornado-generated missiles are designed to withstand such impact without adverse effect upon the system.
- The diesel generator exhaust pipes are enclosed within a 1'9" thick vertical, concrete chimney which is designed to withstand tornado-generated missile impact. A thick, steel pipe sleeve, also capable of withstanding tornado-generated missile impact, provides protection for the exhaust piping at the vent opening at the top of the chimney.

<u>Question 3A.15</u> (NRC Question 210.1) (3.9.6)

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also 6
some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two

or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an inter-system LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action i.e., shutdown or system isolation when the final approved leakage limits are not met. Also surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

6 Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance and etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting valve would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

RESPONSE: PVNGS Operations plans to conduct periodic testing of the pressure isolation valves indicated below: 6

1. Loop 1A RC/SI Check SIV237
2. Loop 1B RC/SI Check SIV247
3. Loop 2A RC/SI Check SIV217
4. Loop 2B RC/SI Check SIV227
5. Loop 1A SIT Check SIV235
6. Loop 1B SIT Check SIV245
7. Loop 2A SIT Check SIV215
8. Loop 2B SIT Check SIV225
9. Loop 1A SI Header Check SIV542
10. Loop 1B SI Header Check SIV543
11. Loop 2A SI Header Check SIV540
12. Loop 2B SI Header Check SIV541
13. Loop 1 HP Long Term Recirculation Check SIV522

INSERT A

INSERT A TO PAGE 3A-17

The Reactor Safety Study, NUREG 75/014 (WASH-1400), identified that plants with pressurized water reactors have a probability of sustaining an intersystem loss of coolant accident (LOCA), which is a significant contributor to the risk of core melt accidents. The design examined in WASH-1400 contained in-series check values that isolate the high pressure piping (Reactor Coolant System) from low pressure piping. The sequence of events leading to the potential intersystem LOCA (Event V) was initiated by two in-series check valves failing to function as a pressure isolation barrier. The Reactor Safety Study concluded that this accident caused an overpressurization and rupture of the low pressure piping resulting in a LOCA that bypasses containment.

NUREG 0677, The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes, indicates that the probability of failure of the in-series pressure isolation check valves can be significantly reduced if these valves receive periodic leak testing.

The Franklin Research Center (Ref. 1) after reviewing the Event V valve configurations, reported that for a piping system to have a valve configuration of concern, the following five items must be fulfilled:

1) The high pressure system must be connected to the Primary Coolant System;
2) there must be a high pressure/low pressure interface present in the line;
3) this same piping must eventually lead outside containment;
4) the line must have one of the valve configurations shown in Figure 1; and
5) the pipe line must have a diameter greater than 1 inch.

The PVNGS Low Pressure Safety Injection (LPSI) system and the High Pressure (HP) long term recirculation system were initially identified as Event V configurations. However, closer investigation proves they are not. The valve configurations for these systems, as shown in Figures 2 and 3, were evaluated to determine the probability of intersystem LOCA using the guidance contained in NUREG-0677.

The LPSI system (see Figure 2* for valve configuration) has three check valves and a motor operated valve (MOV) in series, which forms the interface between the high pressure RCS and the low pressure portion of these systems.

*Discussion only includes loop 1B, however, the same logic is applicable to all loops.

The check valves are assumed to fail in either of two modes - leak or rupture. In the leak failure mode, it is postulated that after the valve has been exercised it does not reseat and establish a pressure boundary. WASH-1400 assigns a probability of 2.6×10^{-3} per year per valve to this type of check valve failure. In addition to failure by the leak mode, a check valve can fail to perform its isolation function due to rupture of its disk. This instantaneous rupture was estimated by WASH-1400 to occur with a probability of 8.8×10^{-5} per year per valve.

The MOV does not exhibit the leak mode failure because the valve is under positive control by its motor operator and position indication is provided. Any small seat leakage that would occur is not expected to be large enough to cause the event. Therefore, its failure mode is rupture of the valve internals, with the probability being the same as for check valve rupture (8.8×10^{-5} per year per valve).

In addition to rupture, the MOV's could be opened by operator error during plant operations. The failure rate for the operator inadvertently opening the valve and not correcting his error has been suggested by the NRC Probabilistic Analysis Staff in NUREG-0677 to be 1×10^{-4} per year.

The probability for each of these failure modes is evaluated and the results are summed for all failure mode combinations to determine the total intersystem LOCA probability for the isolation configuration shown in Figure 2. Evaluating for the 40 year life of the plant and assuming no periodic leak testing, the failure rate is 1.75×10^{-9} per reactor year, which is well below the NUREG-0677 target value of 10^{-7} per reactor year.

The HP long term recirculation system (Figure 3) has two check valves and two normally closed MOV's in series forming the interface between the high pressure RCS and the low pressure portion of the system. There are two operational constraints that must be considered. The configuration provides two normally closed MOV's which eliminates the impact of periodic valve stroking. Also, the double operator error failure mode (inadvertent opening of both MOV's) is eliminated by key-locking the MOV's in the closed position.

The intersystem LOCA probability for the HP long term recirculation system was evaluated for the 40 year life of the plant and assumed no periodic leak testing. The resulting failure rate is 8.79×10^{-11} per reactor year, which is substantially below the NUREG-0677 target value of 10^{-7} per reactor year.

In addition to both isolation configuration failure rates being well below the target value, PVNGS design provides a leak detection system between the first and second check valves. Any leakage past the first check valve can be continuously monitored using the pressure indication and associated control room alarms. In the event that leakage is identified past the first check valve in excess of the Limiting Condition for Operation (LCO) Identified Leakage rate included in

the PVNGS Technical Specification for Reactor Coolant System Operational Leakage, appropriate action will be taken. In conjunction with the leak detection system, a bleed-off line is provided between the first and second check valve. This bleed-off line is capable of venting up to 35 gpm of leakage past the first check valve, which is more than sufficient to vent the maximum LCO Identified Leakage rate.

In conclusion, PVNGS systems are designed toward preventing an intersystem LOCA which bypasses containment. Due to the low probability of failure, additional seat leakage testing imposed by Technical Specifications is not required.

REFERENCES

1. FRANKLIN RESEARCH CENTER, (P.N. Noel, T.C. Stilwell), Technical Evaluation Report -"Primary Coolant System Pressure Isolation Valves", NRC TAC No. 12890, NRC Contract No. NRC-03-79-118.

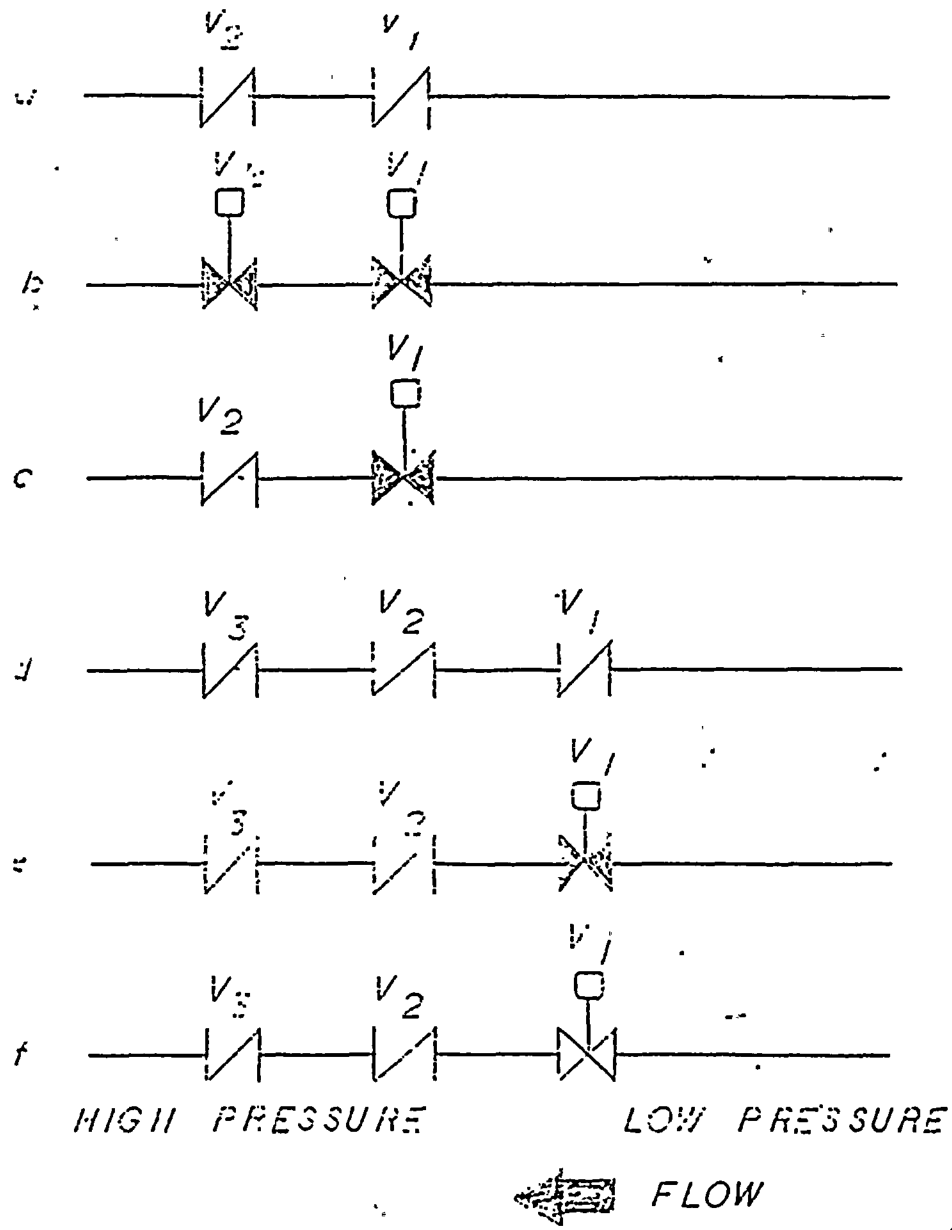

FIGURE 1

ISOLATION CONFIGURATIONS

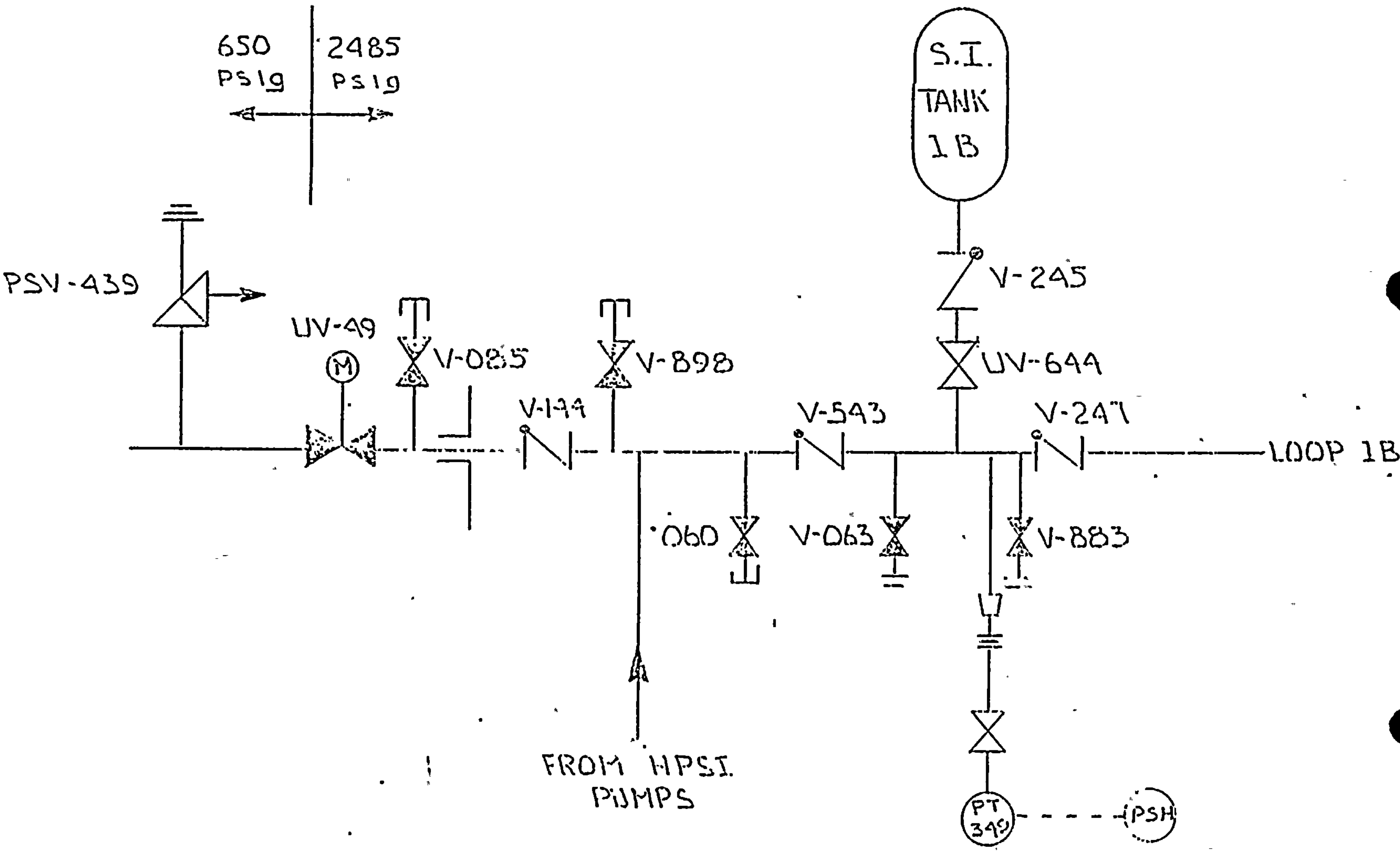

FIGURE 2
TYPICAL FOR LOOPS
1A, 2A, 1B, 2B
(SEE P&ID 13-M-SIP-002)

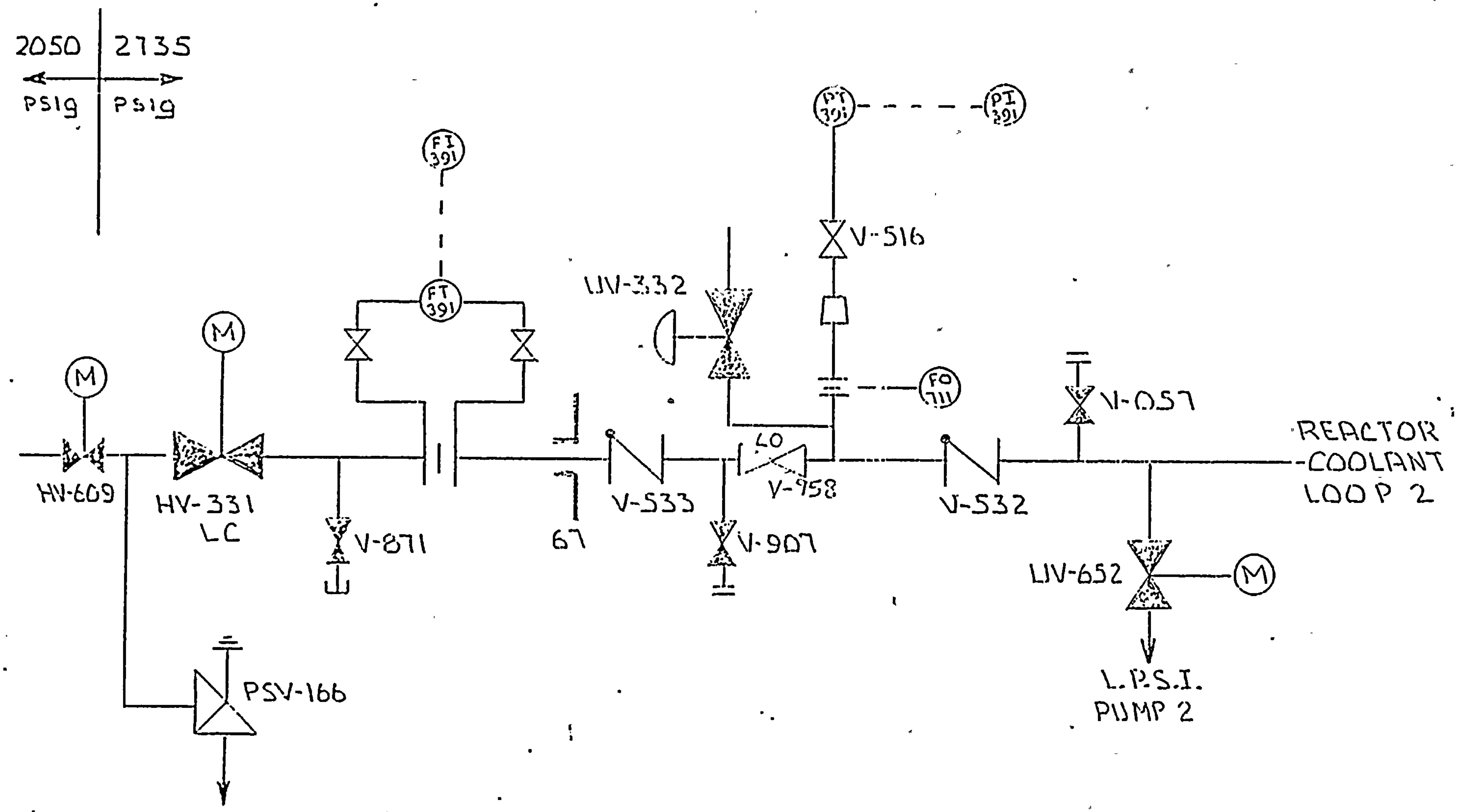

FIGURE 3
TYPICAL FOR LOOPS: 1 & 2
(SEE P&ID 13-M-SIP-002)

14. Loop 1 HP Long Term Recirculation Check SIV523
15. Loop 2 HP Long Term Recirculation Check SIV532
16. Loop 2 HP Long Term Recirculation Check SIV533

Adequate test connections have been provided to facilitate testing of the above listed valves.

Surveillance requirements will be included in the Technical Specifications to verify leakage is within limits:

- prior to reaching power operation following a refueling outage
- prior to returning the valve to service following maintenance, repair or replacement work on the valve
- following valve actuation due to system response to an engineered safety feature actuation signal.

6 The Technical Specifications will include a limiting condition for operation to address the NRC's present position to limit leakage from any reactor coolant system pressure isolation valve to one gal/min. Leak rates higher than one gal/min will be considered acceptable if the leak rate changes are below one gal/min above the previous test leak rate or system design precludes measuring one gal/min with sufficient accuracy.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

Where pressure isolation is provided by two valves, both will be leak tested. When three or more valves provide isolation, only two of the valves will be leak tested.

The four sets of P&ID's have previously been given to the NRC during the working meeting. Procedures for leak testing of these valves will be available onsite for NRC Review 60 days prior to fuel load of Unit 1.

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.

RESPONSE: The response will be provided on the CESSAR docket.

QUESTION 5A.13 (NRC Question 440.2) (5.2.2)

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

RESPONSE: The response will be provided on the CESSAR docket for those tests in the CESSAR scope. For remaining tests not in CESSAR scope, the response is provided in sections 1.8 and 14.B.11.

QUESTION 5A.14 (NRC Question 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.

RESPONSE: The response will be provided on the CESSAR docket for check valves classified Category AC, which are leak tested. The PVNGS design differences from the CESSAR design modifies the list as follows:

Safety Injection (SI) Valves

SI V-215	SI V-522	
SI V-217	SI V-523	
SI V-225	SI V-532	7
SI V-227	SI V-533	
SI V-235	SI V-540	
SI V-237	SI V-541	
SI V-245	SI V-542	
SI V-247	SI V-543	

Adequate test connections and lines, as shown in figure 6.3-1, have been provided to facilitate testing of the above listed valves to ASME IWV-2000 Category AC requirements. ~~The leak detection criteria of 1 gal/min will be included in the~~ Technical Specifications will not include a leak detection criteria for these valves (See our Response to 3A.15).

QUESTION 5A.15 (NRC Question 440.4) (5A)

On page 5A-2, it is indicated that a negative Doppler coefficient of -0.8×10^{-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