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June 16, 1983

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Byron Generating Station Units 1 and 2
Braidwood Generating Station Units 1 and 2
Power Opearational Relief Valves
NRC Docket Nos. 50-454, 50-455, 50-456, and 50-457

Reference (a): October 7, 1982, letter from B. J. Youngblood to L. O. DelGeorge.

Dear Mr. Denton:

This is to provide additional information regarding the use of power operated relief valves (PORV's) for venting and depressurizing the reactor coolant systems at Byron and Braidwood stations. Enclosed with this letter is the response to FSAR question 212.159 which was transmitted in reference (a). This response will be incorporated into the Byron/Braidwood FSAR in a future amendment.

Please address further questions regarding this matter to this office.

One signed original and fifteen copies of this letter and the enclosures are provided for your review.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

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Enclosure

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Question 212.159 (a) Use of PORV's as Pressurizer Vents

In your response to TMI Action Item II.B.1 in the Byron/Braidwood FSAR Appendix E.19, you did not address venting of the pressurizer. In recent discussions with CECo, the staff understands that you plan to take credit for use of the PORV's as vents. The requirement for venting the pressurizer is addressed in 10 CFR 50.44(c)(3)(iii) which requires venting of high points in the RCS. The requirement states that the high point vents must conform to Appendices A and B of 10 CFR Part 50, which includes GDC 2 "Design Bases for Protection Against Natural Phenomena" and GDC 4 "Environmental and Missile Design Bases." Because use of the vents is anticipated only for very severe accidents, venting systems should be designed for the Safe Shutdown Earthquake and harsh environments.

Describe how the current pressurizer vent system of PORV's, block valves and their associated operators meets the requirements for environmental and seismic qualification, or provide a system which will meet these requirements.

Answer

The requirement to add high point vents is stated in 10CFR50.44(c)(3)(iii) as

"...each light-water nuclear power reactor shall be provided with high point vents for the reactor coolant system, for the reactor vessel head, and for other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems."

Byron has a reactor vessel head vent system which meets the requirements set forth in the question. The pressurizer vent system need not be designed to the requirements set forth in the question since non-condensible gas accumulation in the pressurizer will not cause a loss of the core cooling function. Two sets of events have been examined in this regard. The first of these are the design basis events.

Hydrogen generation due to the zirconium-water reaction is discussed in Section 6.2.5.3.1 for the design basis accident. (See also Table 6.2-63 and Figure 6.2-35). The quantity of hydrogen generated due to reaction of 1.5 percent zirconium cladding is approximately 5100 scf. Should this entire volume of gas enter the pressurizer (neglecting hydrogen loss through the break or entrapment in the reactor vessel head or steam generator tubes), pressurizer venting would not be required to maintain adequate core cooling.

Additionally ultra-conservative analyses for the second set, beyond design basis events have shown that the Byron/Braidwood reactor coolant system with the elevated steam generators maintained at their safety valve setpoint have sufficient capacity to maintain the primary system in the safe shutdown condition with significant hydrogen (i.e. the amount corresponding to as much as 20 percent of the zircaloy clad oxidized by a zirconium water reaction) trapped in the primary system by using only the reflux mode of heat transfer to the steam generator secondary side. (This corresponds to about 400 pounds of hydrogen in the primary system.) More realistically, by using the steam generator atmospheric steam dump valves to depressurize the secondary side of the steam generators, the plant could easily be maintained in the safe shutdown condition with hydrogen corresponding to up to 50 percent of the cladding oxidized (i.e., about 1000 lbs. of hydrogen) trapped in the primary system.

The general scenario considered in each case is one in which the ECCS is unavailable and core uncovery occurs for an extended time period. This allows core damage to occur and generates significant hydrogen through a zirconium water reaction. Some of this hydrogen would be bottled up in the primary system, and the remainder would leak out of the break or stuck open valve which initiated the event. The scenario further specifies that recovery has occurred, i.e., core cooling water is made available later in the event for injection into the primary system. Two criteria must be met for the system to be maintained in a safe condition. The first is that the makeup flow into the primary system must be greater than or equal to the rate of primary system inventory loss, and the second is that there must be a means available for decay heat removal.

The thermodynamic condition at which the primary system stabilizes is related to both these criteria. The means available for decay heat removal are energy removal by the break itself, or heat removal through the secondary side of the system generator. The rate of these energy removal mechanisms is directly proportional to the pressure and temperature of the primary system. The primary system will stabilize at a condition where decay heat input is equal to the energy removal.

In general the most limiting condition for recovery is one in which the break occurs by a valve sticking open, the ECCS system is unavailable, at that time core damage and hydrogen generation results, ECCS recovery and core reflooding occurs, and the valve reseats. This provides for loss of the breakflow energy removal mechanism, reduction in the effectiveness of steam generator heat removal, and trapping of some hydrogen in the primary system. (Scenarios involving a physical break or continuously stuck open valve are less severe since the "break" provides a hydrogen vent and a means of sweeping hydrogen and steam out of the primary system. For such cases, ECCS recovery is the only criteria for core cooling and the pressurizer vent system has no role to play.)

The condition of the primary system at this point in the transient is a recovered core (i.e., water covered) with hydrogen present in the system (thus decreasing the effective heat transfer by the steam generator.) The primary system will attempt to stabilize at a pressure where decay heat can be removed. If this pressure is below the safety valve setpoint no inventory loss will occur and no makeup flow is necessary. The mode for heat removal from the primary system would be a "reflux mode" of heat transfer in which boiling in the core generates steam which condenses on the steam generator tubes and the resulting condensate then flows back into the core. It is this situation for which the analyses was performed whose results were previously discussed. Clearly, in this case, the worst case assumption involves the postulation that all the hydrogen is located in the steam generators. Any hydrogen located in the reactor vessel could be removed by the reactor vessel head vent and any hydrogen located in the pressurizer would have no effect other than to enhance the heat removal capability of the steam generators.

These results were obtained under conservative analysis of the Byron/Braidwood plant. In order to bound the problem, two possible configurations of the hydrogen in the primary system were considered. The first assumed that the hydrogen formed a bubble in the top of the steam generator U-tubes causing a flow blockage and a reduction in the available heat transfer area. The second configuration considered was that the hydrogen formed a homogeneous mixture with the steam in the steam generator tubes.

The first situation is illustrated in Figure 1. Masses of hydrogen corresponding to various amounts of zirconium water reactions have been assumed to form a discrete bubble in the steam generator. These amounts of hydrogen obviously represent some fraction of the total hydrogen which could have been generated in some postulated severe accident. This fraction is probabally far less than one, and in fact hydrogen masses corresponding to 50 percent zirconium water reaction maybe actually physically unrealizable given the fact that as the accident occurs steam and hydrogen relief will occur thus limiting the amount of hydrogen present in the primary system at the recovery time. Certainly assuming that all of the hydrogen is present in the steam generator represents a significant conservatism. The bubble is at pressure equilibrium with the primary system, but in thermal equilibrium with the secondary side. It is also conservatively assumed that this bubble blocks the loop flow and thus the only mode of heat transfer is the reflux method illustrated in Figure 1, and thus only half of the tube surface is available for condensation.

The method of analysis for the case of a separate hydrogen bubble forming in the tops of the steam generator U-tubes was as follows. First the volume of the steam generator taken up by the hydrogen bubble was determined based on the ideal gas law and the assumption that the hydrogen bubble temperature was the same as the secondary side saturation temperature. (This methodology is consistent with assuming no heat transfer from the bubble.) The required heat transfer coefficient was then determined based on the heat transfer area available in the steam generator, and a conservative decay heat at 1000 seconds. (Note that calculations indicate that for this limiting case using a decay heat value corresponding to 1000 seconds is very conservative.) The analysis was performed for a range of primary system pressures from 2500 psia to 400 psia, for a range of percent cladding reaction, and for the case of a pressurized secondary side (steam generator at its safety valve setpoint) and a depressurized secondary side. The required heat transfer coefficient for the two cases, pressurized secondary side and depressurized secondary side, is plotted in Figures 2 and 3 respectively as a function of primary system pressure at which the system could stablize and the percent clad reacted. These graphs can be used to identify the pressure at which the primary system stabilizes at following an accident which generates a given amount of hydrogen, once the expected heat transfer coefficient has been quantified. If this pressure is below the primary system's safety valve setpoint (i.e., about 2500 psia) the plant will not continue to lose inventory and the plant is in a safe stable condition. Figure 2 shows that if the secondary side is at its safety valve setpoint (i.e., about 1200 PSIA) for 5, 10 and 20 percent clad reaction the primary system will stabilize at 2500 psia or less given that the heat transfer coefficient from the primary or secondary side is 28, 35, or 80 BTu/Hr/Ft^2-°F or greater respectively. Similarly Figure 3 shows that if the secondary side has been depressurized to about atmospheric conditions (i.e., by using the safety grade atmospheric dump system) then for Zirconium water reaction of 5, 10, 20, 30, and 50 percent, the primary system will stablize at 2500 psia or less given that the heat transfer coefficient from primary to secondary is 5.6, 6. , 7. , 8.0, and 12. BTu/Hr-Ft^2-°F or greater respectively. These required heat transfer coefficients are significantly below those expected for this situation.

The overall heat transfer coefficient is set by three separate heat transfer resistances. The boiling heat transfer coefficient on the secondary side, the conduction through the tube, and the condensation on the secondary side. The boiling and conduction coefficients can be calculated rather straightforwardly and yield values of approximately 2000 and 3000 BTu/Hr-Ft^2-°F respectively. The final component, the condensation heat transfer coefficient inside the steam generator tubes, can be calculated by performing a force balance at the inlet of the steam generator tubes to determine the water film thickness. This water film thickness can then used to calculate the film heat transfer coefficient by assuming that the heat transfer coefficient can be represented by the thermal conductivity of water divided by the film thickness. Using the film heat transfer coefficient, along with the steam generator tube conductivity and boiling heat transfer coefficient on the secondary side, an expected overall heat transfer coefficient was estimated. This expected heat transfer coefficient was then compared with the required heat transfer coefficients. This method is conservative since the force balance was performed at the steam generator tube inlet thus giving the thickest film layer and greatest resistance to heat transfer. The condensation coefficient is calculated to be approximately 420 BTu/Hr-Ft^2-°F and the overall heat transfer coefficient about 310 BTu/Hr-Ft^2-°F.

The results of this analysis indicate that 20 percent clad reaction (about 400 pounds hydrogen) could be tolerated assuming a pressurized secondary side (steam generator safety valve setpoint) and up to 50 percent clad reaction (about 1000 pounds hydrogen) could be tolerated assuming a depressurized secondary side. Final points worth noting are that the range of the required heat transfer (i.e., from 6 to 80 BTu/Hr-Ft^2-°F) are in the range of those expected for free and forced convection, and are significantly below those expected for condensation heat transfer. The calculation performed to quantify the condensation heat transfer coefficient was probably not even necessary since from a judgemental point of view it should be obvious that the expected heat transfer coefficient will be greater than the required. It should also be noted that the ability to maintain counter-current flow in the steam

generator U-tubes was verified using the Wallis flooding criteria. The conclusion being that counter current flow can be maintained for the primary system pressures of interest (i.e. 2500 - 400 psia), and the decay heat level used.

The second situation studied is illustrated in Figure 4. All of the hydrogen which has been assumed to be generated, is residing as a homogeneous mixture in thermal equilibrium with the steam in the steam generator and occupying the total tube volume.

The method of analysis for the case of a homogeneous mixture of steam-hydrogen in the steam generator was to first determine the steam-hydrogen mixture temperature based on the ideal gas law and Dalton's law of partial pressures. Based on the mixture temperature, the secondary side saturation temperature, and decay heat at 1000 seconds, a required heat transfer coefficient was determined for a range of primary system pressures (i.e. 2500 to 400 psia) and for a pressurized and unpressurized primary side. The required heat transfer coefficient for the two cases, pressurized and depressurized secondary sides, is plotted in Figures 5 and 6 respectively as a function of primary system pressure and percent clad reacted. From the plots, it can be seen that a smaller heat transfer coefficient is required to remove decay heat than those calculated for the case of a hydrogen bubble in the tops of the steam generator U-tubes. This is due to the larger heat transfer area available (essentially the entire steam generator) used in this portion of the analysis. The required heat transfer coefficients determined were compared to an adjusted stagnant Tagami correlation and to work performed by Sparrow, Minkowycz, and Saddy[1] for heat transfer in the presence of non-condensibles. This comparison should be conservative since the empirical data from Tagami and the analytical work of Sparrow, Minkowycz, and Saddy was developed for low pressure steam-air mixture. Steam diffusion through a hydrogen boundary layer is expected to be faster than diffusion through an air boundary layer and increases in system pressure will also increase the rate of mass transfer. The results of this portion of the analysis support the conclusions of the first portion of the analysis.

It should also be noted that one significant conservatism in this analysis is that no credit was taken for the safety grade reactor head vent which provides a noncondensible gas vent, and a means to remove decay heat. Another conservatism places all the hydrogen in the steam generator. This is equivalent to saying that the Byron design can accommodate for more than the 50% Zirconium-water reaction without a loss of core cooling. Clearly, the reactor vessel head vent extends this margin and is the vent associated with the flow configuration of interest for Byron.

In summary for design basis events the PORV's are not required as non-condensible vents. For beyond design basis events the Byron Braidwood plant specific primary system design provides a significant capability for core cooling even given significant quantities of noncondensible gases are trapped in the primary system. The PORV's would not be required for removing non-condensibles given a wide spectrum of beyond design basis events.

REFERENCES

1. Int. J Heat and Mass Transfer Vol. 10 pp. 1829-1845.

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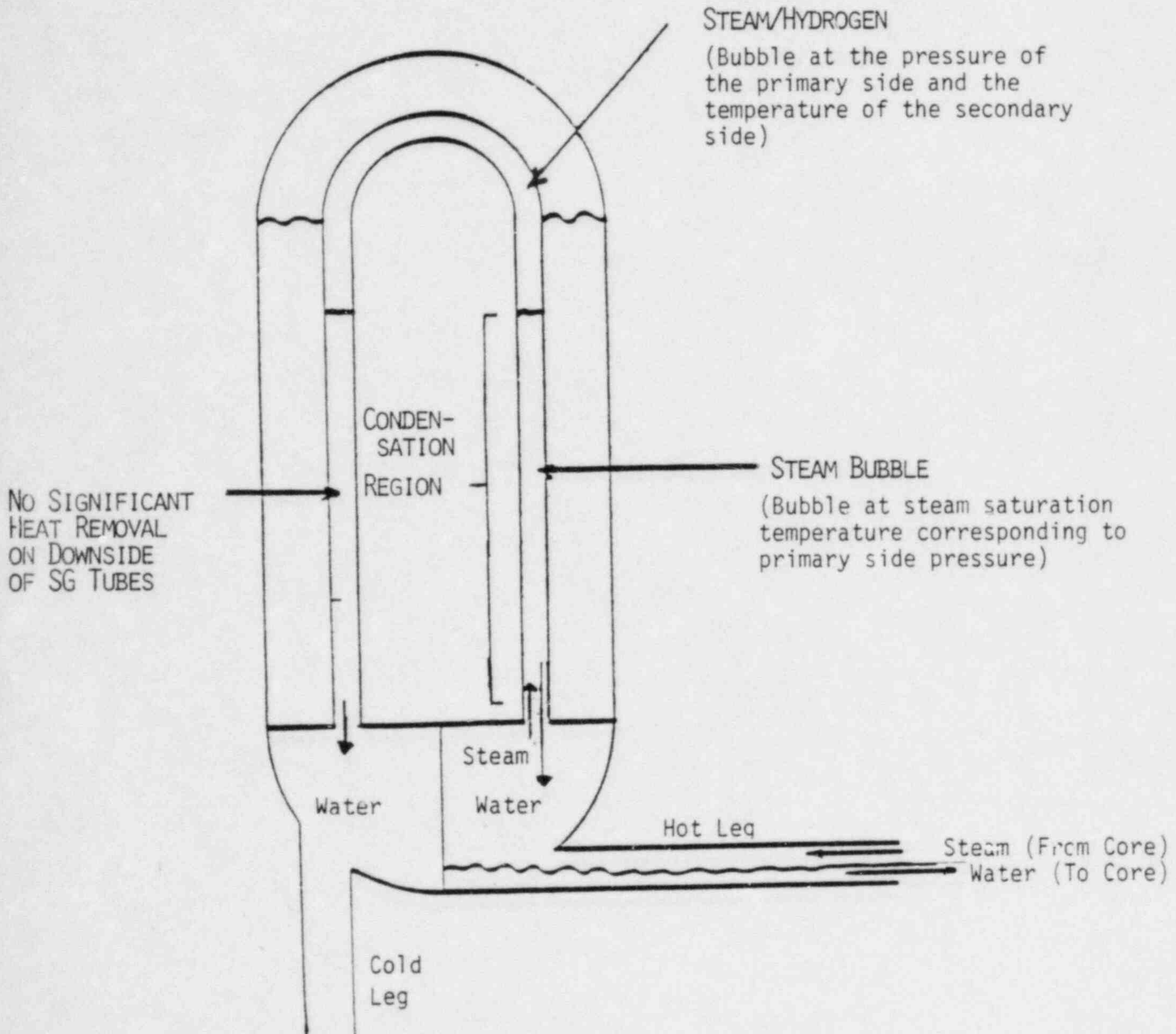

Figure 1
Loop Reflux Illustration

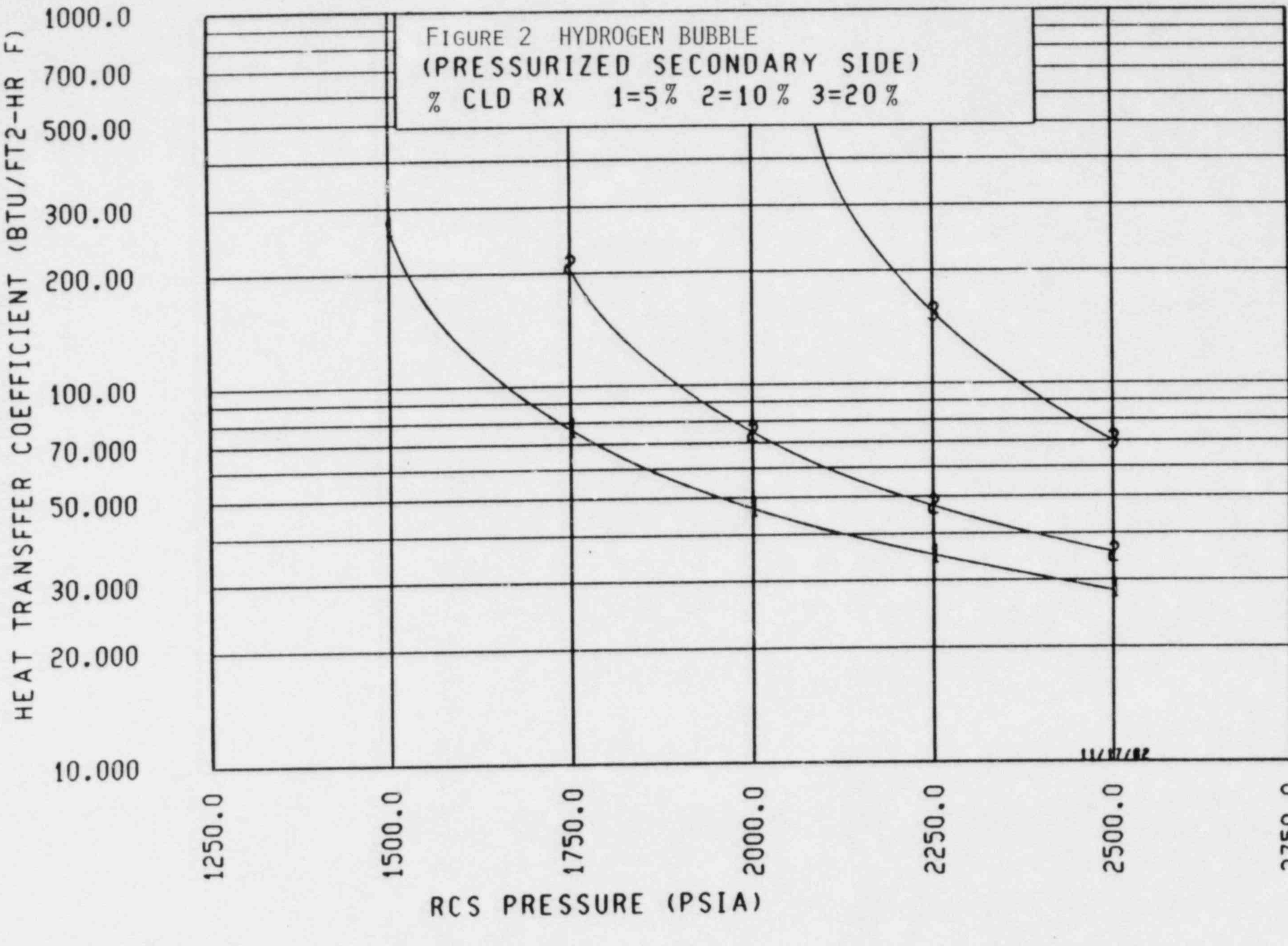

FIGURE 2 HYDROGEN BUBBLE
(PRESSURIZED SECONDARY SIDE)
% CLD RX 1=5% 2=10% 3=20%
HEAT TRANSFFER COEFFICIENT (BTU/FT2-HR F)
1000.0
700.00
500.00
300.00
200.00
100.00
70.000
50.000
30.000
20.000
10.000
1250.0
1500.0
1750.0
2000.0
2250.0
2500.0
2750.0
RCS PRESSURE (PSIA)

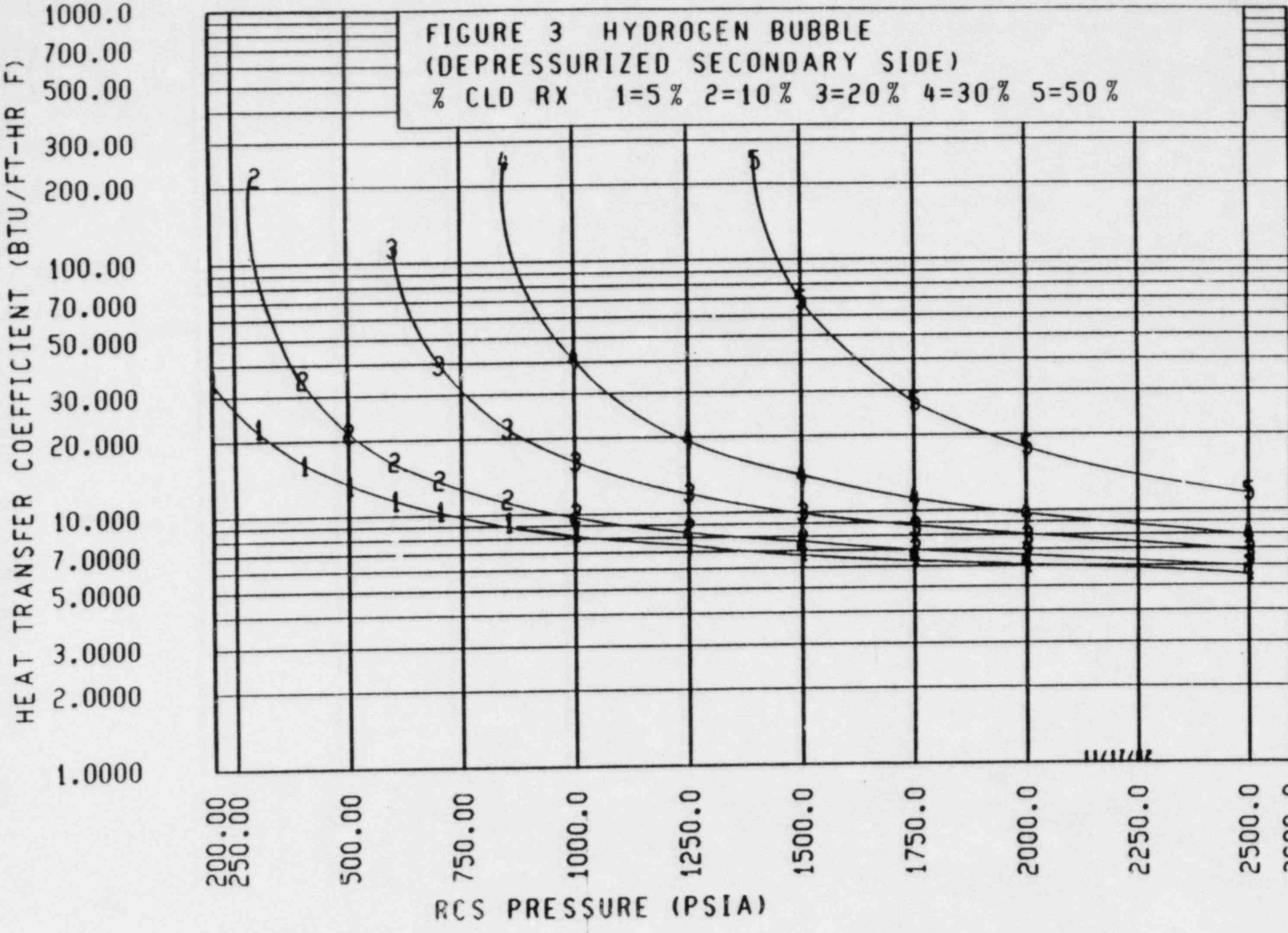
FIGURE 3 HYDROGEN BUBBLE
(DEPRESSURIZED SECONDARY SIDE)
% CLD RX 1=5% 2=10% 3=20% 4=30% 5=50%
HEAT TRANSFER COEFFICIENT (BTU/FT-HR F)
1000.0
700.00
500.00
300.00
200.00
100.00
70.000
50.000
30.000
20.000
10.000
7.0000
5.0000
3.0000
2.0000
1.0000
200.00
250.00
500.00
750.00
1000.0
1250.0
1500.0
1750.0
2000.0
2250.0
2500.0
2600.0
RCS PRESSURE (PSIA)

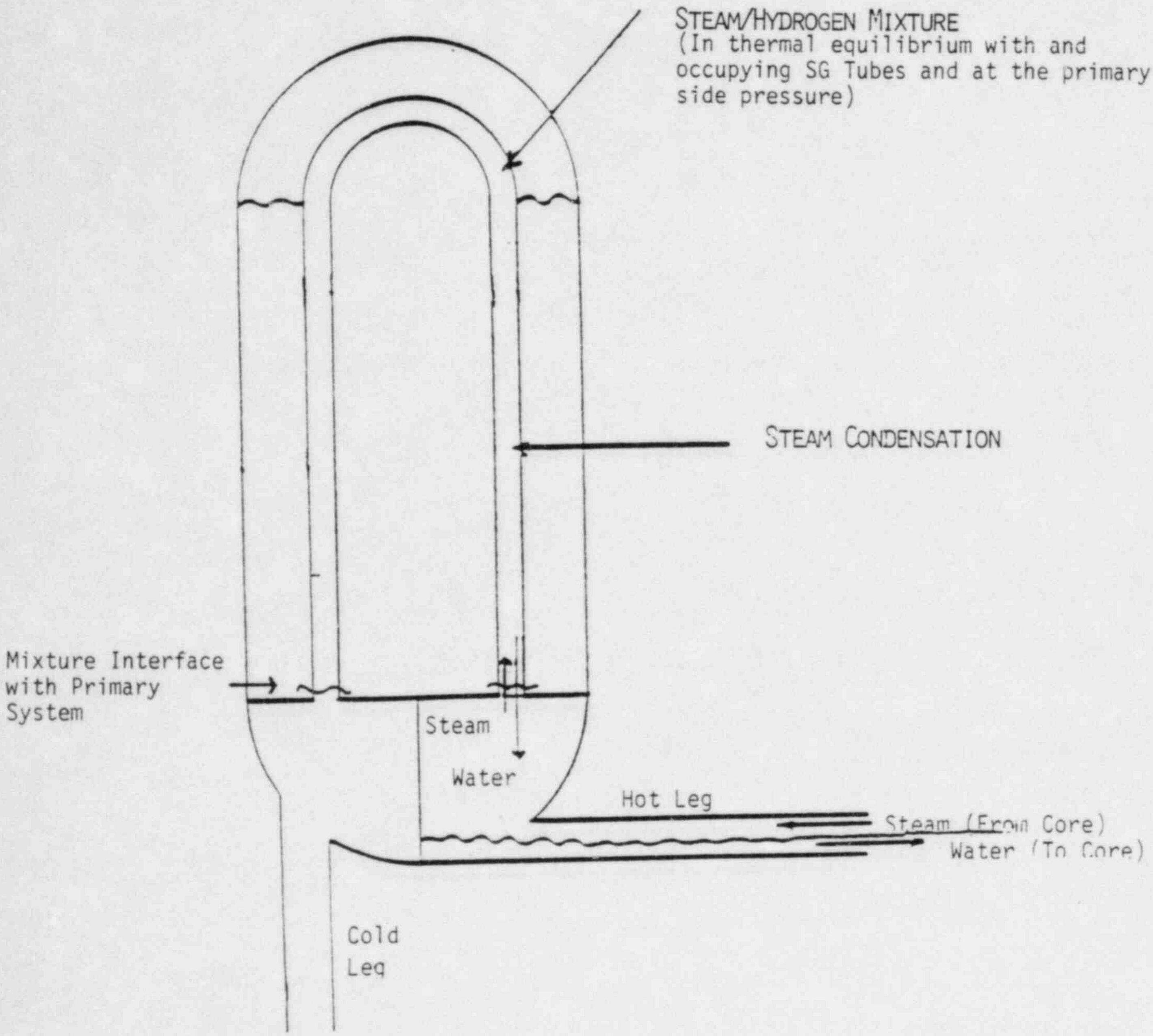

Figure 4

Loop Reflux Illustration

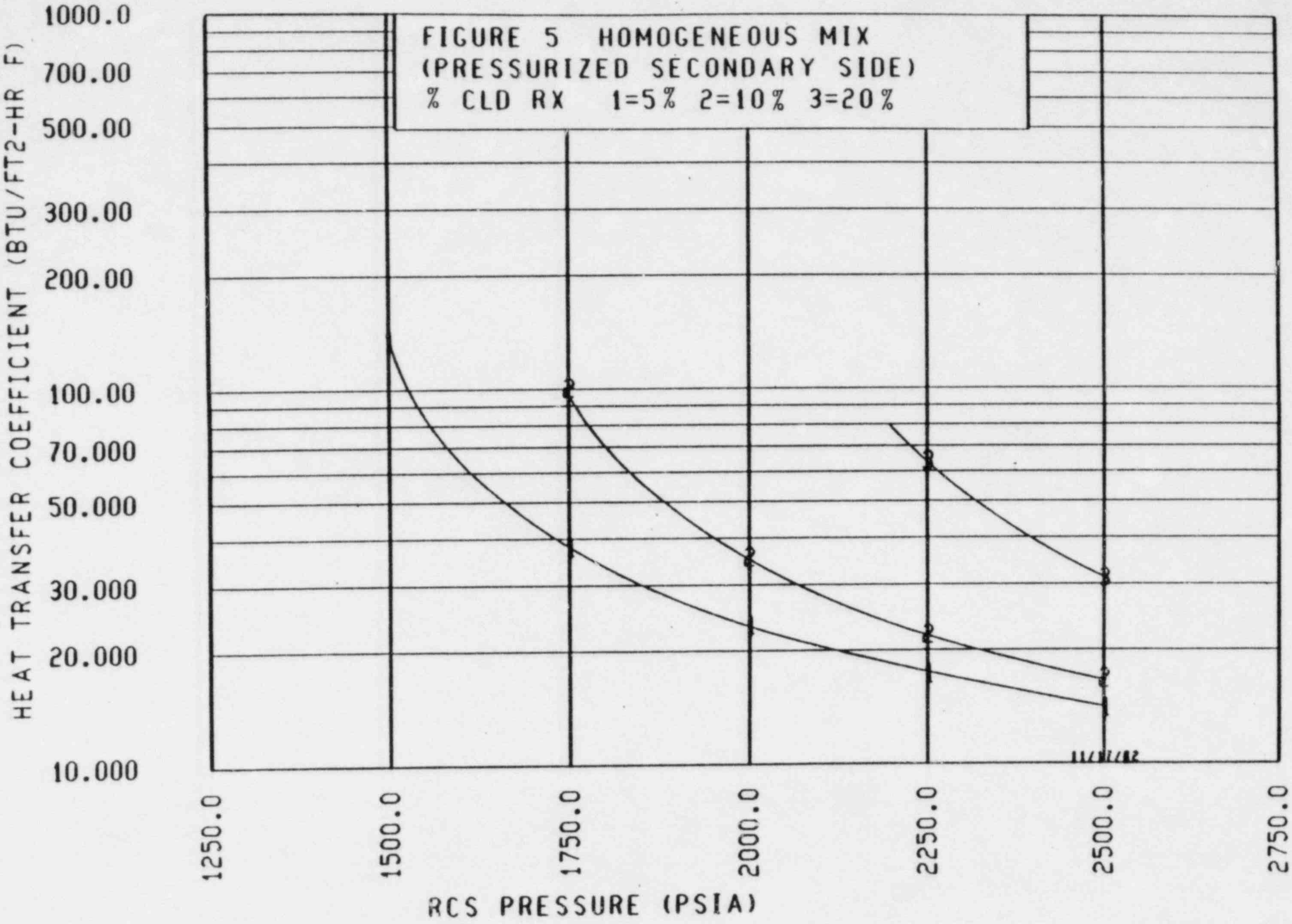
FIGURE 5 HOMOGENEOUS MIX
(PRESSURIZED SECONDARY SIDE)
% CLD RX 1=5% 2=10% 3=20%
HEAT TRANSFER COEFFICIENT (BTU/FT2-HR F)
1000.0
700.00
500.00
300.00
200.00
100.00
70.000
50.000
30.000
20.000
10.000
1250.0
1500.0
1750.0
2000.0
2250.0
2500.0
2750.0
RCS PRESSURE (PSIA)

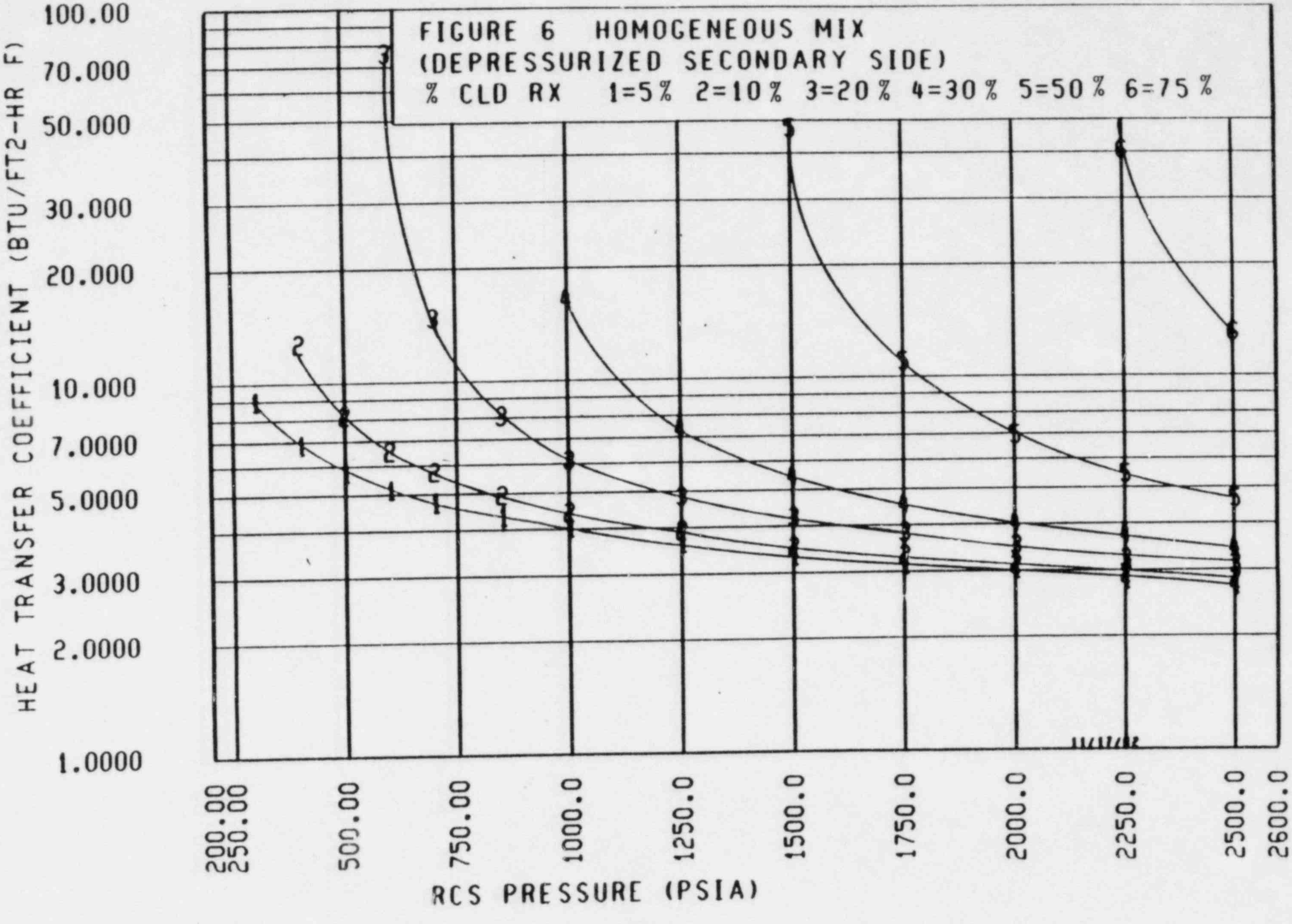
FIGURE 6 HOMOGENEOUS MIX
(DEPRESSURIZED SECONDARY SIDE)
% CLD RX 1=5% 2=10% 3=20% 4=30% 5=50% 6=75%
HEAT TRANSFER COEFFICIENT (BTU/FT2-HR F)
100.00
70.000
50.000
30.000
20.000
10.000
7.0000
5.0000
3.0000
2.0000
1.0000
200.00
250.00
500.00
750.00
1000.0
1250.0
1500.0
1750.0
2000.0
2250.0
2500.0
2600.0
RCS PRESSURE (PSIA)

b) Method of Depressurization to Cold Shutdown - RSB Branch Technical Position 5-1 (see SRP Section 5.4.7)

RSB BTP 5-1 requires that the reactor be designed to be taken from normal operating conditions to cold shutdown using only safety-grade systems and that thses systems satisfy GDC 1 through 5. It is the staff's position that to satisfy this functional requirements, the PORV and its operator should be designed to withstand the SSE.

Describe how your PORVs and operators meet this requirement or provide an alternative method for depressurization to cold shutdown, using only safety-grade components.

Answer

Previous discussion of compliance with RSB 5-1 has been provided in response to questions 212.6, 212.47, and 212.154. The response to this last question contained detailed information related to the use of the CVCS auxiliary spray and the PORVs in performing the depressurization function. Byron is a class 2 plant and can take credit for repairs and manual actions as outlined in the above question reponses. Based on the foregoing, we take the position that the PORV and operator need not be qualified to the SSE. However, we have conducted analyses that show that the PORV and it's operator will be capable of operation subsequent to an SSE with only minor modifications. Edison will make these modifications. We believe that this extra margin, while not required for a class 2 plant, should satisfy the staff's interest.

c. Depressurization following Steam Generator Tube Rupture Events. In the Byron/Braidwood FSAR, Section 15.6.3, you take credit for operation of the PORVs to mitigate the consequences of a steam generator tube rupture event. It is the staff's position that, to take this credit, the PORV;s should be classified as safety related equipment and should meet the applicable General Design Criteria, specifically, to be seismically and environmentally qualified. Describe how you system meets the staff requirements or provide an alternative method, using safety-related components, to mitigate this event, or demonstrate by analysis that the PORV's are not needed to mitigate the consequence of this event.

Your resolution for (b) and (c) above may include the development of procedures with manual backups and verification that an adverse environment will not be created by use of the PORVs in these situations.

<u>Answer</u>

As discussed in Section 15.6.3, depressurization will be accomplished via any of several methods depending upon the availability of the components and power supplies (i.e. offsite power). Depressurization following a SGTR will be accomplished in three stages. The first stage involves depressurizing the reactor coolant system to a pressure slightly less than that of the faulted steam generator secondary side. The second and third stages bring RCS pressure down to RHRS initiation conditions (approximately 415 psia) and, finally, to atmospheric pressure. Adequate time exists for both the second and third depressurization stages to permit manual actions to recover previously unavailable components (such as auxiliary spray).

An adverse environment is not expected within the containment during the initial recovery period of the SGTR and prior to opening of a PORV for depressurization. Following opening, the containment pressure has been calculated to increase approximately 7 psig while the containment temperature would rise to approximately 160^{o}F. Analyses have been performed that demonstrate the ability of the PORVs to withstand this environment.

In any event, the single opening of a PORV is adequate to depressurize the RCS. Should the PORV fail in the open position, the PORV block valve provides a fully qualified safety related means of isolating the stuck open PORV.

As noted in our response to (b) above, the PORV and it's operator will withstand the SSE and will function subsequent to such an event.

GTK/sb/2378b*