Duquesne Light Company Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077-0004

JOHN D. SIEBER
Vice President - Nuclear Group

(412) 393-5255

March 23, 1990

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Response to R. G. 1.97

Gentlemen:

This letter is in response to the request for additional information (RAI) on the Regulatory Guide 1.97 (R.G. 1.97) variable for steam generator wide range (SGWR) level instrumentation. The RAI was by telecon with the Region 1 and NRR Staff on March 8, 1990.

During the March 8th telecon, our response of January 31, 1990 to the R. G. 1.97 SGWR level instrumentation deviation was discussed. It was determined that it would be helpful if Duquesne Light Company provided a detailed discussion of the alternative parameters and the combination of parameters as they relate back to steam generator level (i.e. heat sink availability). Additional information on the associated flowpaths through the Emergency Operating Procedures (EOPs) was also requested.

Please contact my office if there are any questions regarding this submittal.

Very truly yours,

J D Sieber

J. D. Sieber
Vice President
Nuclear Group

Attachment

cc: Mr. J. Beall, Sr. Resident Inspector
Mr. T. T. Martin, NRC Region I Administrator
Mr. P. Tam, Sr. Project Manager
Mr. C. Anderson (Region 1)

A003
1/1

Problem Description and NRC Request

The three wide range steam generator level channels are indicated and recorded on a single recorder. A failure of the power supply to the recorder would result in a loss of three wide range level indications.

The three wide range level channels are powered from separate vital busses but are routed together in neutral cable trays from the sensors to the indicating and recording device. No isolation devices are installed. Regulatory Guide 1.97 describes steam generator level as a D1 variable.

The Duquesne Light Company (DLC) submittal of December 18, 1989 discusses alternative and diverse instrumentation. A more detailed discussion of these parameters and the combination of parameters as they relate back to steam generator level (i.e. heat sink availability) is requested for NRC review. Additional information on the associated flowpaths through the Emergency Operating Procedure (EOPs) is also requested.

Response to NRC Request for Additional Information

Our response to the NRC RAI is arranged according to the following outline:

1. Background - This item summarizes the previously submitted information by DLC.

2. BVPS-1 Functional Design - Specific BVPS-1 information is provided from the UFSAR and the background documents of the EOPs.

3. EOPs

   A. This item provides a discussion of the three BVPS-1 procedures beyond the design basis which utilize the SGWR level instrumentation.

   B. This item provides a discussion of the other pathways in the EOPS which do not reference the SGWR level-instrumentation but addresses conditions of the secondary heat sink.

4. Summary and Conclusions

1. Background

As discussed in our letter to the NRC dated January 31, 1990, we performed the review of and provided a resolution to the SGWR level instrumentation deviation to R. G. 1.97. Our proposed resolution is based on the following information.

- In our letter of December 18, 1989, we provided the safety implications based on discussions in the UFSAR and with respect to the design basis events. We concluded that there are adequate diverse and independent instrumentation channels in accordance with R. G. 1.97 for accident diagnostics and mitigation of design basis events.

- In the same letter, we also addressed the issue of all three wide range indications appearing on a common recorder. We indicated that the wide range level indication is also provided to three separate computer systems which, although they are not 1E, are battery backed systems. We included this information on Table 1 of the letter. The Table also listed the alternate indications on the emergency shutdown panel, locally in the auxiliary feedwater pump rooms, and the control room and provided the power supply vital bus designations.

- In the same letter, we also addressed the issue of the operators knowing how to respond to a loss of wide range steam generator level. We noted that the procedures require maintaining auxiliary feedwater flow to the steam generators until level is in the narrow range, thereby providing adequate guidance. This operator action provides assurance of an adequate heat sink.

- In our letter of January 31, 1990, we provided the logic model which we developed to show that the procedures are supported by the BVPS-1 design. The model shows that additional systems are available to mitigate conditions leading to steam generator dryout. The logic model shows several success paths for supplying 350 gpm of auxiliary feedwater to any steam generator. The system is supplemented by an additional auxiliary feedwater pump, FW-P-4, which was included to satisfy Appendix R. The 4160V electrical supply for the auxiliary feedwater system will have another power source added for Station Blackout. The model also shows the redundancy of the river water supplies. In reviewing the SGWR level instrumentation, we noted that the configuration is similar to other three-loop plants.

- Finally, in our telecon with the NRC on March 8, 1990, we included a discussion of the alternate instrumentation for accident diagnostics. This instrumentation and discussion of its use is provided below.

2. BVPS-1 Instrumentation and Functional Design Instrumentation

Figure 1 of this attachment is a graphical view of the instrumentation provided for each steam generator. The figure can act as a reference for the discussions which follow. As can be seen on the figure, there are the following instrumentation channels which have indication in the control room. Please refer to Table 1 of our submittal of December 18, 1989 for additional information on the auxiliary feedwater flow and the WR and NR level instrumentation.

- Auxiliary Feedwater (AFW) - There is one (1) flow transmitter for each steam generator in the line supplied by the two motor driven and one steam driven AFW pumps. The flow from the fourth AFW pump, FWP-4, can be detected by two flow transmitters in the normal feedwater supply line.

- Steam Generator Steam Flow and Pressure - There are two flow transmitters and three pressure transmitters for each steam generator. The flow instrumentation is located upstream of the safety and relief valves as shown on the figure and the pressure transmitters are upstream of the main steam isolation valves.

- Level Transmitters - The approximate arrangement of the one WR level and three NR level transmitters are shown on the figure.

- Also, not shown on the figure, radiation monitoring is provided for the air ejector and blowdown lines.

UFSAR Major Secondary System Pipe Rupture

Included below are the functions which provide the necessary protection against a steam pipe rupture

1. Safety injection system actuation from any of the following:

   a. Two-out-of-three low pressurizer pressure

   b. Two-out-of-three low steamline pressure in any one loop

   c. Two-out-of-three high containment pressure

2. The overpower reactor trips (neutron flux and T) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

3. Redundant isolation of the main feedwater lines: Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to normal control action which will close the main feedwater valves, a safety injection signal will rapidly close all feedwater control valves, trip the main feedwater pumps and close the feedwater pump discharge valves.

4. Trip of the fast acting steam line stop valves (designed to close in less than 5 seconds after receipt of the signal) on:

   a. Two-out-of-three low steam line pressure in any loop (above Permissive P-11)

   b. High-high containment pressure

   c. Two-out-of-three high steam line pressure rate in any loop (below Permissive P-11).

Fast-acting isolation valves are provided in each steam line that will fully close within 5 seconds of a large break in the steam line. For breaks downstream of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would blowdown under single failure criteria.

Steam flow is measured by monitoring dynamic head in nozzles inside the steam pipes. The nozzles which are of considerably smaller diameter than the main steam pipe are located inside the containment near the steam generators and also serve to limit the maximum steam flow for any break further downstream.

It should be noted that following a steam line break only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In the case of loss of offsite power, this heat is removed to the atmosphere via the steam line safety valves which have been sized to cover this condition.

UFSAR Major Rupture of a Main Feedwater Pipe

Included below are the functions that provide the necessary protection against a main feedwater rupture.

1. A reactor trip on any of the following conditions:

   a. High pressurizer pressure

   b. Overtemperature delta-T

   c. Low-low steam generator water level in any steam generator

   d. Low steam generator level plus steam/feed flow mismatch in any steam generator

   e. Safety injection signals from either of the following:

      1) Low steam line pressure

      2) High containment pressure

2. An auxiliary feedwater system to provide an assured source of feedwater to the steam generators for decay heat removal. (Refer to Section 10 for description of the auxiliary feedwater system.)

A major feedwater line rupture is defined as a break in a feedwater pipe large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedline between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break. Further, a break in this location could preclude the subsequent addition of auxiliary feedwater to the affected steam generator.

It is noted that for the main feedwater pipe rupture that the worst possible break area is assumed; i.e., one that empties the affected steam generator and causes a reactor trip on low-low steam generator water level at the same time as the fluid inventory in the unaffected steam generator drops to the trip point for low level coincident with steam/feed flow mismatch. It is also noted that the steam generator level referred to above as causing the reactor trip is the narrow range instrumentation.

With respect to the above accidents, we note that the minimum requirements for the main control board indications for S. G. level are one level channel per steam generator (either wide or narrow range). (Reference Table 7.5-2, UFSAR.)

EOPs

The following information is taken from the background documents of the BVPS-1 EOPs. The purpose of providing this information is to show: 1) what secondary information is most important to the operator for identifying a faulted steam generator and, 2) the alternate parameters useful to the operator in identifying the approach to steam generator dryout when all steam generators are faulted. It is not intended that the parameters presented below are all inclusive of those available. Complete listings of available instrumentation are provided in Item 3 below.

1. Procedure E-2, "Faulted Steam Generator Isolation," provides actions to identify and isolate a faulted steam generator. The procedure is entered from E-0, "Reactor Trip Or Safety Injection," or E-1, "Loss of Reactor or Secondary Coolant," when any SG pressure decreases in an uncontrolled manner or any SG completely depressurizes. Other procedures have a transition to E-2 whenever a faulted SG is identified and faulted SG isolation is not verified. After taking the required actions in this procedure, the operator is directed to either E-1, "Loss of Reactor or Secondary Coolant" or E-3, "Steam Generator Tube Rupture," depending on whether a SGTR is identified. If all SGs are determined to be faulted, the operator is directed to ECA-2.1, "Uncontrolled Depressurization of All Steam Generators." Procedure E-2, "Faulted Steam Generator Isolation," is intended to identify and isolate a loss of secondary coolant resulting from a fault in a main steamline, main feedwater line or in any piping system that interconnects with the secondary side pressure boundary (e.g., auxiliary feedwater system, blowdown piping).

   Small Secondary Break

   For this size break, it is not expected that an automatic reactor trip or safety injection would occur and, therefore, procedure E-2 would not be implemented.

Intermediate Size Secondary Break

An intermediate size break is lower bounded by those sizes in which the normal plant control systems are unable to maintain approximate nominal plant operating conditions and upper bounded by those sizes in which the protective functions do not occur within approximately five minutes following initiation of the event. The intermediate steamline break is categorized by a slowly decreasing steamline pressure in at least one loop depending upon the location of the break. If the break occurs in the steam header, all loops will experience decreasing pressure. Due to the increased steam load for which the control systems are unable to compensate, a slowly decreasing steam generator water level will result and also a slowly decreasing primary average temperature.

If the break occurs upstream of the main steamline isolation valves (MSIVs), the steam generator associated with the faulted loop will blow down to atmospheric pressure. If the break occurs downstream of the steamline isolation valves, the transient is terminated following MSIV closure. The system process parameter trends that are used to identify a faulted SG are an uncontrolled pressure decrease in at least one steamline or a SG that is completely depressurized. Other symptoms include increased main feed flow to at least one steam generator, slowly decreasing primary average temperature and slowly decreasing steam generator water level in at least one steam generator.

For an intermediate feedline break in which the control systems are incapable of compensating for the loss of flow, the secondary side would experience a slowly decreasing steam generator water level in at least one steam generator. The transient is eventually terminated by manual reactor trip or when the low or low-low level trip setpoint is reached in any one steam generator. This results in a reactor trip and auxiliary feedwater initiation. A subsequent turbine trip occurs due to reactor trip. If the break occurs downstream of the main feedline non-return valves, all steam generators continue to experience a reverse blow down through the steam generator associated with the faulted loop until a low steamline pressure setpoint is attained resulting in a safety injection initiation and steamline and feedline isolation. The faulted steam generator will then blow down until atmospheric pressure is reached. If the break occurs upstream of the feedline non-return valves, the feedwater spillage is terminated and the auxiliary feedwater system is sufficient to mitigate the consequences of the resultant loss of normal feedwater transient. The system parameter trends that are used to identify a faulted SG are an uncontrolled pressure decrease in at least one steamline or a SG that is completely depressurized. Other symptoms include decreasing water level in at least one steam generator and slowly rising primary system average temperature prior to reactor trip.

For either of the above transients, if the break occurs inside containment, an increasing containment temperature and/or pressure indication could be observed. If the break occurs outside containment, audible or visual indications may assist the operator in diagnosing the transient.

Large Secondary Break

For the double-ended main steamline break, an immediate decrease in pressure in at least one steamline occurs depending upon the location of the break. The low steamline pressure setpoint is reached (5-10 seconds) which results in safety injection initiation and steamline isolation. This yields a reactor trip, turbine trip, main feed isolation and auxiliary feedwater initiation. If the break is downstream of the MSIVs, closure of the MSIVs may terminate the blowdown to atmospheric pressure. The important system parameter trends for this break are an uncontrolled pressure decrease in at least one steamline or a SG that is completely depressurized, other symptoms include decreasing steam generator water level in at least one steam generator and initially decreasing primary pressure and temperature.

The double-ended main feedline break exhibits characteristics quite similar to the double-ended steamline break. The system response is characterized by a rapid decrease in steam generator water level in at least one steam generator. Following reactor trip, all steam generators exhibit reverse blowdown through the faulted feedline until the low steamline pressure setpoint is reached in any steamline resulting in steamline and feedline isolation and safety injection initiation. Following MSIV closure, one steam generator blows down to atmospheric pressure.

For both the double-ended steamline break and feedline break, steamline pressure in at least one steamline would be rapidly decreasing. Containment pressure and temperature increases would be observed if the break occurred inside containment. Audible and visual confirmation of the break may be possible if the break occurs outside containment. Also, for all secondary double-ended high energy line breaks, a distinct characteristic is the closure of all main steamline isolation valves.

2. Loss of Secondary Heat Sink

The most serious challenge to the Heat Sink Critical Safety Function is an indication of loss of secondary heat sink. A loss of secondary heat sink occurs if decay heat removal is needed through the SGs and all feed flow capability is lost.

Feed flow must be re-established or an alternative heat removal mode (e.g., bleed and feed) must be established to prevent core uncovery and eventually an inadequate core cooling condition. It is noted that the parameters monitored on the Heat Sink Status Tree are SGNR Level, Feedwater Flow, and SG Pressure.

Procedure FR-H.1, "Response To Loss Of Secondary Heat Sink," provides guidance to address an extreme challenge (i.e., RED priority) to the Heat Sink Critical Safety Function that results if total feed flow is below a minimum value and level is below the narrow range in all SGs at any time.

Procedure FR-H.5, "Response to Steam Generator Low Level," provides guidance to address a not satisfied condition (i.e., YELLOW priority) from low secondary system inventory that affects any SG. Procedure FR-H.5 has been developed and structured to complement procedure FR-H.1, "Response to Loss of Secondary Heat Sink," in maintaining secondary heat sink. Whereas Procedure FR-H.1 provides guidance to maintain adequate secondary heat sink (i.e., adequate inventory in at least one SG), procedure FR-H.5 provides guidance to restore and maintain secondary inventory in the normal range (i.e., narrow range level) in all nonfaulted SGs.

The objective of Procedure FR-H.1 is to maintain reactor coolant system (RCS) heat removal capability by establishing feed flow to a SG or through establishing RCS bleed and feed heat removal. Procedure FR H.1 is entered at the first indication that secondary heat removal capability may be challenged. This permits maximum time for operator action to restore feedwater flow to at least one steam generator before secondary inventory is depleted and secondary heat removal capability is lost. Once secondary heat removal capability is lost, RCS bleed and feed must be established to minimize core uncovery and prevent an inadequate core cooling condition.

A loss of secondary heat sink can occur as a result of several different initiating events. It is not the intent here to investigate these events but to get to the bottom line, operator information necessary to initiate RCS Bleed and Feed Heat Removal.

Analyses for a high pressure plant were used to determine the conditions necessary to initiate successful bleed and feed for different PORV flow to power ratios. High pressure plants are those plants which have charging/HHSI pumps and, thus, can inject at pressures above the pressurizer PORV setpoint pressure. Based on the analyses, these plants can successfully initiate bleed and feed heat removal before or even shortly after the time of steam generator dryout if the PORV flow to power ratio is greater than 140 (lbm/hr)/Mwt.

Steam generator dryout is defined as the loss of sufficient secondary inventory such that the core heat load shifts to heating the primary RCS inventory. The best symptom to use for loss of heat sink would then be a direct indication of loss of secondary inventory. Wide range steam generator level provides a direct indication of secondary inventory and, therefore, represents the best indication for initiating bleed and feed for high pressure plants.

As steam generator dryout occurs, an increase in RCS pressure and temperature results in the opening of the pressurizer PORVs. Thus, for high pressure plants with a PORV flow to power ratio greater than 140 (lbm/hr)Mwt, an alternative symptom for successful initiation of bleed and feed is increasing RCS pressure and temperature or pressure greater than or equal to 2335 PSIG (PRZR PORV set pressure).

Finally, the following provides concerns in using the identified symptoms above and identifies the alternatives.

For a plant with only a single wide range level channel per SG, the symptom is modified to ensure that a failure of one channel will not prevent timely initiation of bleed and feed heat removal. Therefore, for an "N" loop plant, the symptom is "N-1" wide range SG levels to be less than the level indicated for initiation of bleed and feed heat removal.

If the wide range SG level channels are not qualified, the indications cannot be used in an adverse containment environment. Therefore, an alternative symptom is required for the wide range SG level indication. However, BVPS Unit 1 SG wide range level transmitters are environmentally qualified.

An alternative symptom for plants with a PORV flow to power ratio greater than 140 (lbm/hr) Mwt is increasing RCS pressure and temperature or PRZR pressure greater than or equal to the PRZR PORV set pressure (e.g., 2335 PSIG).

3. EOPs

This section provides the pathways through the EOPs and Function Restoration Procedures for those steps concerned with the condition of the secondary heat sink. The first pathways (3A) are of most interest in this submittal because they address the use of the S.G. WR instrumentation. A listing of instrumentation as it appears in the background documents is provided for selected steps.

3A. Three Procedures Utilizing the SG WR instrumentation.

The three procedures utilizing the SGWR instrumentation are FR-H.1, "Response To Loss of Secondary Heat Sink", FR-H.5 "Response to SG Low Level", and ECA-0.0, "Loss of All Emergency 4KV AC Power." FR-H.1 and FR-H.5 were discussed above in Section 2. They are primarily entered when required from the "Heat Sink" Status Tree F-0.3. The entry conditions are shown on Figure 2 for FR-H.1, FR-H.5 and ECA-0.0.

Procedure ECA-0.0, "Loss of All 4KV Emergency AC Power," provides procedural guidance for loss of all emergency AC power as an initiating event or as a coincident occurrence in combination with a loss of reactor coolant, loss of secondary coolant or steam generator tube rupture. The procedure and supporting analysis are primarily structured to address the loss of all emergency AC power with or without a loss of all normal AC power as an initiating event that occurs when the plant is in the startup or power operational mode. However, the procedure has been augmented to provide appropriate guidance should a concurrent loss of reactor coolant, loss of secondary coolant or steam generator tube rupture exist.

The next most important element to the restoration of AC power in this procedure, and one that is appropriate for our purposes here, is the maintenance of plant conditions for optional recovery. This element consists of actions to mitigate deterioration of RCS conditions and establish plant conditions amenable to optimal recovery following AC power restoration. The operator is limited in actions available to mitigate deteriorating RCS conditions. By minimizing RCS inventory loss and maintaining a secondary heat sink the operator can extend the time to core uncovery.

RCS inventory loss is minimized by depressurizing the secondary system, thereby (1) reducing RCS temperature to minimize RCP seal degradation and (2) reducing RCS pressure to reduce RCP seal leakage and to permit injection of SI accumulator water to partially replace the RCS inventory lost through the RCP seals.

Secondary heat sink is maintained by controlling the turbine-driven AFW pump and the rate of steam generator steam release to maintain narrow range level in at least one intact steam generator. Actions are included to isolate a ruptured or faulted steam generator and to switch the AFW suction to an alternate water supply, if necessary. Therefore, much of the discussion about steam generator depressurization is also applicable here.

In ECA-0.0 the SGWR level instrumentation is used in Steps 17 and 18. In FR-H.1 and FR-H.5 the level instrumentation is used in Steps 3, 10, and 21, and in Step 4, respectively. Copies of these steps and the Step descriptions are attached.

3B. Other procedures addressing the conditions of the secondary heat sink.

The primary procedure of concern here is E-2, "Faulted Steam Generator Isolation," which was discussed above in Section 2.

E-2 is entered from the following:

1. E-0, "Reactor Trip or Safety Injection," Step 22, with the following symptoms:

    a. Any SG pressure dropping in an uncontrolled manner.

    b. Any SG completely depressurized.

2. E-1, "Loss of Reactor or Secondary Coolant," Step 4;
   E-3, "Steam Generator Tube Rupture," Step 8;
   ECA-3.1, "SGTR With Loss of Reactor Coolant - Subcooled Recovery Desired," Step 10;
   ECA-3.2, "SGTR With Loss of Reactor Coolant - Saturated Recovery Desired," Step 3, with the following symptoms and/or conditions:

    a. Any SG pressure dropping in an uncontrolled manner.

    b. Any SG completely depressurized.

    c. Faulted SG isolation not verified.

3. FR-H.5, "Response to Steam Generator Low Level," Step 3, when the affected SG is identified as faulted.

4. Other procedures whenever a faulted SG is identified.

As can be seen from the above, E-2 is entered for the most part wherever required by S.G. pressure reduction.

4. Summary and Conclusions

What we intended to show in this submittal is that, for the majority of loss of heat sink accidents, as addressed in the BVPS-1 existing EOPs, the S.G. NR level and the SG (steam) pressure are important system parameters and that the SGWR level is typically used in conjunction with S.G. NR level. It is not until entry into the function restoration procedures that SGWR level indication is not used in conjunction with SGNR level. Then, it is only after the loss of all AFW capability. We have shown in our previous submittal by a logic model that the BVPS-1 AFW System is a highly reliable system and includes additional design features atypical of other plants.

Also, in our previous submittal, we provided the following:

As discussed in FSAR Section 7.5.2, Tables 7.5-1 and 7.5-2, the wide range level channels, in conjunction with the narrow range level channels, are necessary for the assurance of an adequate heat sink. In addition, they are used as diagnostic instrumentation for steam generator tube ruptures and high energy breaks in the main steam and feedwater systems. The side range steam generator level indications are referenced in the emergency operating procedures for station blackout, low steam generator level and loss of heat sink. These three procedures, however, are beyond the design bases for Beaver Valley 1 and would not be entered unless there were multiple failures of the emergency diesel generators and/or auxiliary feedwater system in addition to a loss of both offsite power sources. Therefore, an additional failure of the wide range level system is highly improbable.

With respect to the design bases events (steam generator tube ruptures and secondary piping failures), redundant and diverse instrumentation are available for diagnostic purposes as follows:

Steam Generator Tube Ruptures:

Three narrow range level channels per steam generator, two feedwater flow channels per steam generator, two steam flow transmitters per steam generator, three steam pressure channels per steam generator, air ejector radiation monitoring, blowdown radiation monitoring, one auxiliary feed flow channel per steam generator, containment sump level indication and alarm, containment temperatures, pressures, radiation and humidity indicators.

Secondary Systems' Piping Failures:

One auxiliary feedwater flow channel per steam generator, three steam pressure channels per steam generator, two steam and feed flow channels per steam generator, three narrow range level channels per steam generator, containment sump, pressure, temperature, humidity and radiation indication.

We also note again, that in Table 1 of our December 18, 1989 submittal, the SGWR level indication is available in the control room, at the shutdown panel, locally in the AFW Pump Room, and on three separate computer systems. In addition, we have provided by letter to all BVPS-1 licensed personnel, specific information on these sources of SGWR level information and instructions on actions to be performed in accordance with the EOPs in the absence of all sources of SGWR level information. Information on redundant and diverse instrumentation available for diagnostic purposes for steam generator tube ruptures and secondary system piping failures was also provided in the letter.

Therefore, based on the above, we conclude that:

- The existing primary and secondary systems instrumentation is sufficiently qualified, redundant, and diverse to enable the operators to determine the condition of the secondary heat sink.

- The existing EOPs correctly instruct the operators on determining loss of heat sink (i.e. faulted steam generator(s)) and in coping with these conditions.

- The operators are aware of the sources for, and alteratives to, SGWR Level information if they are confronted with a loss of secondary heat sink.

- The AFW System is a highly reliable system such that multiple failures need to be postulated to lose its function. As a corollary to this, we believe that SG dry out at BVPS-1 is an extremely low probability event. This low probability is assured through an additional non-safety related diesel backed feedwater pump, diverse sources of demineralized water, diverse safety related sources of river water, and multiple (5) indications available for wide range steam generator level indication beyond the installed 3 pen indicating recorder.

FIGURE 1

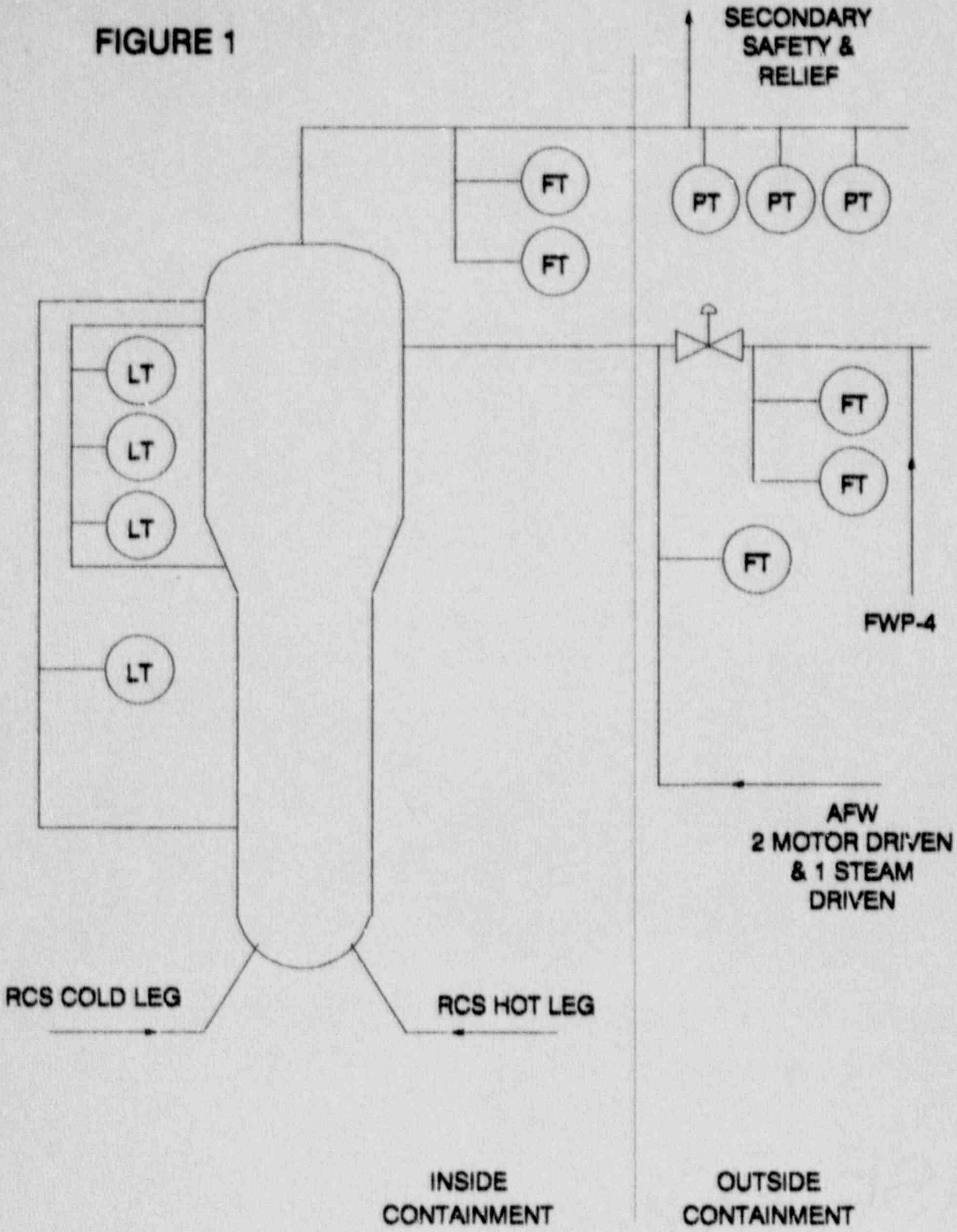
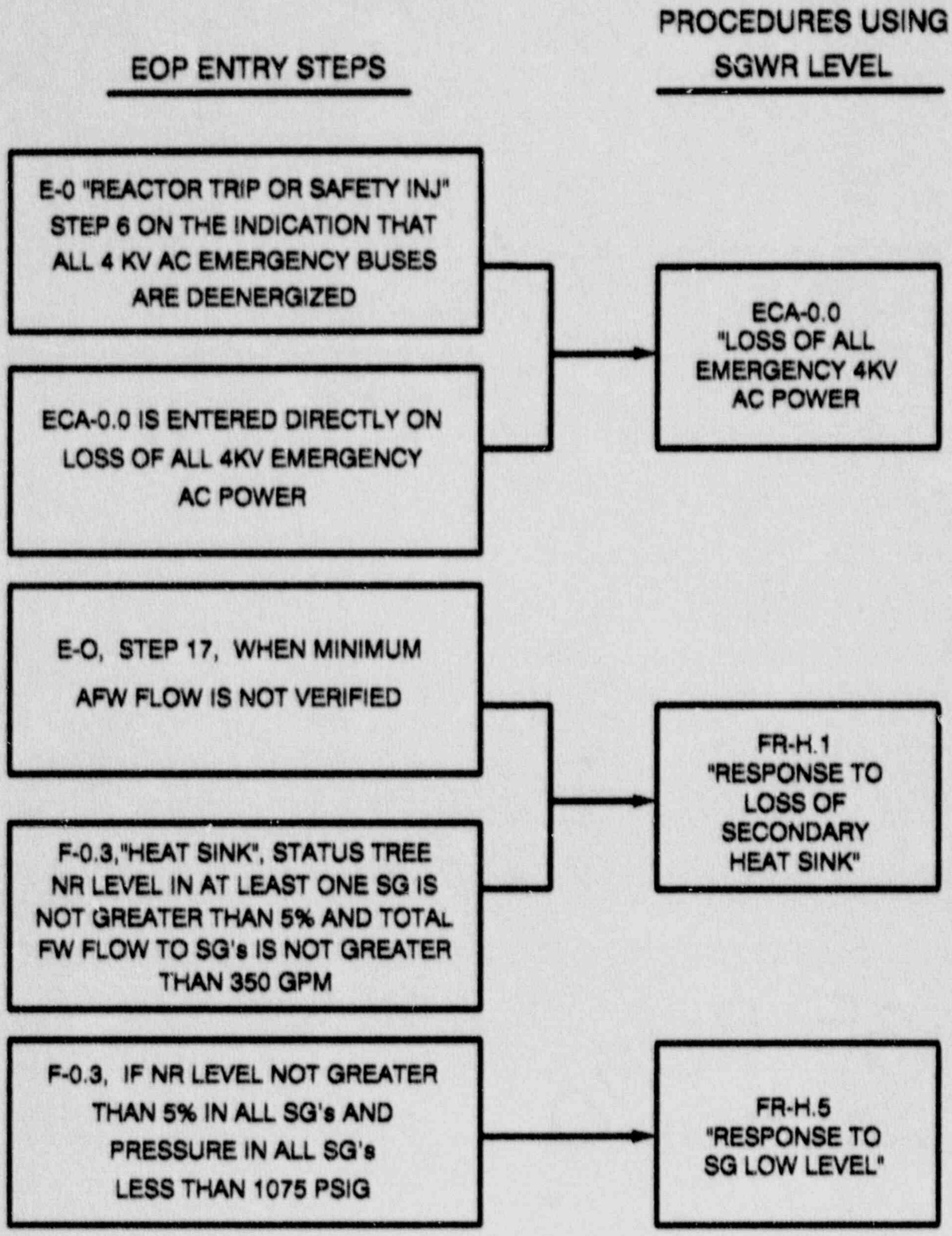

SECONDARY
SAFETY &
RELIEF
FT
FT
PT
PT
PT
LT
LT
LT
LT
FT
FT
FT
FWP-4
AFW
2 MOTOR DRIVEN
& 1 STEAM
DRIVEN
RCS COLD LEG
RCS HOT LEG
INSIDE
CONTAINMENT
OUTSIDE
CONTAINMENT

FIGURE 2


EOP ENTRY STEPS
PROCEDURES USING SGWR LEVEL
E-0 "REACTOR TRIP OR SAFETY INJ" STEP 6 ON THE INDICATION THAT ALL 4 KV AC EMERGENCY BUSES ARE DEENERGIZED
ECA-0.0 IS ENTERED DIRECTLY ON LOSS OF ALL 4KV EMERGENCY AC POWER
ECA-0.0 "LOSS OF ALL EMERGENCY 4KV AC POWER
E-O, STEP 17, WHEN MINIMUM AFW FLOW IS NOT VERIFIED
F-0.3,"HEAT SINK", STATUS TREE NR LEVEL IN AT LEAST ONE SG IS NOT GREATER THAN 5% AND TOTAL FW FLOW TO SG's IS NOT GREATER THAN 350 GPM
FR-H.1 "RESPONSE TO LOSS OF SECONDARY HEAT SINK"
F-0.3, IF NR LEVEL NOT GREATER THAN 5% IN ALL SG's AND PRESSURE IN ALL SG's LESS THAN 1075 PSIG
FR-H.5 "RESPONSE TO SG LOW LEVEL"