October 24, 1995

APPLICANT: Westinghouse Electric Corporation

PROJECT: AP600

SUBJECT: SUMMARY OF MEETING TO DISCUSS AP600 REACTOR SYSTEM DESIGN

The subject meeting was held in Rockville, Maryland, on April 25 and 26, 1995, between representatives of Westinghouse Electric Corporation and the Nuclear Regulatory Commission staff. The purpose of the meeting was to address items relating to the AP600 reactor system design. The discussions were based on issues raised in a meeting on March 27, 1995, and formally documented in a letter to Westinghouse from the NRC dated April 19, 1995.

The staff and Westinghouse went over each item in detail and agreed to document resolution or additional action commitments in the Open Items Tracking System (OITS) database. No additional issues were raised during the meeting.

Westinghouse also provided a short presentation opened to all the technical review staff on the recent design changes to the AP600.

Attachment 1 is the list of meeting attendees. Attachment 2 includes handouts provided by Westinghouse during the meeting to clarify various discussions items.

original signed by:

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Office Of Nuclear Reactor Regulation

Docket No. 52-003

Attachments: As stated

cc w/attachments: See next page

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Westinghouse Electric Corporation

Docket No. 52-003

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MEETING ATTENDEES
APRIL 25 AND 26, 1995

NAME	ORGANIZATION
Andrea Sterdis	W AP600 Licensing
John Butler	W Safety Analysis
Tim Rowell	W Safety Analysis
Mike Corletti	W Nuclear Safety
Terry Schulz	W AP600 Systems
Bill Huffman	NRC/ADAR/PDST
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Y. Gene Hsii	NRC/DSSA/SRXB
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Attachment 1

REACTOR SYSTEMS BRANCH

T. L. SCHULZ
SYSTEMS ENGINEERING
APRIL 25, 1995

Attachment 2

Question 440.89

Section 6.3.1.2 of the SSAR states that the frequency of automatic depressurization system actuation is limited to a low probability to reduce safety risks and to minimize plant outage. Define the term "low probability." and explain how this goal is achieved.

Response:

The response to RAI 471.20 provides an estimate of the frequency of inadvertent ADS for the AP600. The results indicate that the frequency of inadvertent ADS actuation is approximately 2E-3/yr. This frequency is similar to the frequency of very small and small LOCAs used in the AP600 PRA (1.07E-3/yr).

Achieving a "low probability" of inadvertent ADS actuation is achieved through the use of certain design features and valve opening characteristics. These features and characteristics include the following (also included in the response to RAI 471.20):

1. The SSAR chapter 15 analysis shows that ADS operation does not occur during non-LOCA events or steam generator tube ruptures. It is only expected to occur during an inadvertent ADS event, LOCA's, or long term safe shutdown when all ac power is lost for more than 24 hours.

2. The ADS is actuated with 2 out of 4 logic. As a result, multiple instrumentation failures are required to inadvertently actuate the ADS.

3. Each ADS stage 1, 2, and 3 line has two normally closed valves in series. The ADS stage four valves are interlocked to prevent their opening at normal RCS pressures. As a result, multiple failures are necessary to inadvertantly actuate the ADS.

4. If ADS is actuated inadvertently or for a small RCS LOCA, the fourth stage will not be actuated assuming that the operators start the normal residual heat removal pumps. These pumps provide the RCS injection function and cause the CMT injection to stop with the CMT level above the fourth stage actuation setpoint.

5. If the normal residual heat removal system is unavailable in an inadvertent ADS event and the ADS fourth stage valves open, the containment will slowly flood to its maximum level over several days. Recovery of the normal residual heat removal system during this time allows the floodup to be terminated.

SSAR Revision: NONE

Question 440.200

It appears that given a LOCA of any size, even a very small-break LOCA, part of the expected plant response is to use full ADS actuation. Is this correct? If yes, the frequency of full ADS actuation is higher than the frequency of LOCAs that is approximately 2E-03 per year. Is this accurate?

Response:

Automatic depressurization system actuation is not expected for reactor coolant system leaks and very small LOCAs (<3/4"). As discussed in Reference 440.200-1, the chemical and volume control system has the capability to cope with these events. For small LOCAs up to one inch, the chemical and volume control system, in conjunction with core makeup tank injection provides sufficient makeup to bring the plant to shutdown prior to automatic depressurization system actuation. Therefore, the frequency of full automatic depressurization system actuation is not higher than the frequency of LOCAs.

Not every automatic depressurization system actuation leads to opening of the fourth stage automatic depressurization system valves (which causes high containment pressure and temperature and containment floodup). If the normal residual heat removal system is available, its injection stops the core makeup tank draindown and prevents the fourth stage automatic depressurization actuation. As discussed in Reference 440.200-2, the results of the initiating event frequency evaluation included in the RTNSS implementation, do not identify any RTNSS significant nonsafety-related SSCs with respect to these intitiating events.

References:

440.200-1 WCAP-13793, "AP600 System / Event Matrix," June 1994.

440.200-2 WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Summary Report," September 1993.

SSAR Revision: NONE

MOV INTERLOCKS (15)

- **Current Plant Approach**
  - Accumulator MOVs
    - Power lock out at motor control center
    - 2-way redundant valve position indication and alarm
    - Tech Spec monitoring of position
    - Confirmatory open signal

- **AP600 Approach**
  - Accumulator / IRWST MOVs
    - Identical to current plants
  - PRHR HX / CMT inlet MOVs
    - Redundant series controllers
      - Both must actuate to close valve
    - 3-way redundant valve position indication and alarm
    - Tech Spec monitoring of position
    - Confirmatory open signal

Question 440.104

Section 6.3.2.5.1 of the SSAR states that the PXS has been "specifically designed to treat check valves failures to reposition as active failures." The core makeup tank discharge line contains two tilt-disc check valves in series. The FMEA in Table 6.3-6 does not consider the failure modes for these check valves because they are not considered active failures as they are normally open and remain in the same position on demand. However, for an accident where the accumulators discharge into the RCS, these check valves will close to prevent backflow into the CMT, and will have to reopen to inject borated water into the RCS.

a. The arrangement of two check valves in series does not meet the single failure consideration. Either modify the CMT discharge line check valve arrangement, or provide justification for treating these check valves as passive components. Also, provide the results of the FMEA analysis for these CMT discharge line check valves.

b. Describe the CMT discharge line check valve design to discuss how they are normally maintained open.

c. Technical Specification 3.5.2 specifies the LCO and Surveillance Requirements (SR) for the CMT. Why are there no SRs to verify that the CMT outlet check valves are open, and no action requirement when the check valves are not open?

Response:

Since these check valves sit in an open position for up to two years between inservice tests that exercise them, the main concern is their failure to close. As a result, series check valves have been installed to account for the possibility of a single normally open check valve failing to close. The differential pressure available to close them is limited to the differential pressure generated in the direct vessel injection line during accumulator injection. This differential pressure is sufficient to overcome the counterbalance weight that normally keeps the valves open. Because of the counterbalance weight, no flow/differential pressure is required to re-open the valve. Since more flow/differential pressure is required to close the valve than open it, it is unlikely that the check valves would not re-open as soon as the accumulator flow slows down. This position is consistent with the PRA which treats standby check valve failures as 2.0 E-7 fail/hr. The chance of a standby core makeup tank check valve failing between refueling outages is then 1.75 E-3. The chance of a core makeup tank check valve failing to re-open 5 minutes after it closes is therefore only 1.66 E-8, or 5 orders of magnitude less. As a result, parallel check valves have not been incorporated. Adding parallel check valves would increase the complexity of the piping design (thermal stresses), increase the chance of a reactor coolant system leak or a LOCA, reduce the chance that the accumulator bypass flow will be isolated, and increase testing and maintenance which increases radiation exposure.

These check valves are tilt disk check valve designs which are counter balanced such that they normally hang in a fully open position. Because the disk sits completely in the flow stream it closes quickly with reverse flow.

The technical specifications for the core makeup tanks do not address the normal position of these check valve because they will normally be in their proper position and during power operation there is no way to change their position. This approach is consistent with how the accumulator check valves are handled in current plants.

SSAR Revision:

Revise the fourth paragraph of Subsection 6.3.2.5.1 to include re-opening of the core makeup tank check valves as an exception to the general treatment of check valves as single failures as follows:

There are two exceptions to this treatment of check valve failures in the passive core cooling system. One ~~The only~~ exception to this treatment ~~in the passive core cooling system~~ is made for the accumulator check valves, which is consistent with the treatment of these specific check valves in currently licensed plant designs. The other exception is made for the core makeup tank check valves failure to re-open after they have just closed during an accident. This exception is based on the low probability of these check valves not re-opening within a few minutes after they have cycled closed during accumulator operation.

RCS LEAK VS ADS ACTUATION (25)

- **Normal Makeup System (CVS) Can Makeup for 3/8" Break**
  - One of 2 CVS pumps can accommodate 3/8" instrument line break
  - Can support normal plant shutdown
  - Safety-related systems would not be actuated

- **CMT Can Support Shutdowns for Smaller RCS Leaks**
  - No credit for CVS makeup
  - 10 gpm leak will result in ADS after 30 hours

- **For Larger RCS Leaks ADS Will Actuate**
  - No credit for CVS makeup
  - 100 gpm leak (3/8 ") will actuate ADS after 3 hours

3.5 PASSIVE CORE COOLING SYSTEMS

3.5.8 In-containment Refueling Water Storage Tank (IRWST) - Shutdown, RCS Inventory Low

LCO 3.5.8 The In-containment Refueling Water Storage Tank, with two injection flow paths and two containment sump recirculation flow paths, shall be OPERABLE.

APPLICABILITY: MODE 5 with RCS open, level not visible in pressurizer; MODE 6 with upper internals in place and cavity level less than full.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or both motor-operated containment sump isolation valve(s) in one flow path inoperable.	A.1 Restore motor-operated containment sump isolation valve(s) to OPERABLE status.	72 hours
B. One motor-operated containment sump isolation valve not fully closed.	B.1 Close motor-operated containment sump isolation valve.	72 hours
C. One motor-operated IRWST isolation valve inoperable.	C.1 Restore motor-operated IRWST isolation valve to OPERABLE status.	72 hours
D. IRWST water volume, boron concentration, or water temperature not within limits.	D.1 Restore IRWST to OPERABLE status.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 Automatic Depressurization System (ADS) - Shutdown, RCS Open

LCO 3.4.14 ADS stage 1, 2, and 3, flow paths shall be open.

APPLICABILITY: MODE 5 with RCS pressure boundary open;
MODE 6 with upper internals in place and the refueling cavity water level < 23 ft above the reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required flow path closed. AND One or more ADS stage 4 flow path(s) OPERABLE.	A.1 Open the affected flow path. OR A.2 Open an alternative flow path with an equivalent area.	72 hours 72 hours
B. One required flow path closed. AND All ADS stage 4 flow paths closed and inoperable.	B.1 Open the affected flow path. OR B.2 Open an alternative flow path with an equivalent area. OR B.3 Restore at least one stage 4 flow path to OPERABLE status.	12 hours 12 hours 12 hours
C. Required Action and Associated Completion Time not met while in MODE 5. OR LCO not met for reasons other than Conditions A, B, or D while in MODE 5.	C.1 Initiate action to fill the RCS to establish a visible level in the pressurizer. AND C.2 Maintain RCS temperature as low as practical. AND C.3 Suspend positive reactivity additions.	Immediately Immediately Immediately

3.6 CONTAINMENT

3.6.10 Containment Penetrations

LCO 3.6.10 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by [four] bolts;

b. The containment airlocks shall be clear of obstructions such that they can be closed quickly;

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE containment isolation (or other safeguards actuation)signal, or

APPLICABILITY: MODE 5, RCS open;
MODE 6, with upper internals in place and cavity less than full.

Question

A passive PRHR HX tube rupture event was not analyzed in Chapter 15.6 of the SSAR. Provide this analysis

Response

In assessing the small break LOCA performance of AP600, the most relevant parameter is the minimum RCS inventory observed during events. The most limiting case in the small break LOCA spectrum is the one which exhibits the smallest minimum RCS inventory (i.e. possesses the least margin to core uncovery). Among the breaks analyzed for the SSAR, it is the larger breaks, such as the double-ended direct vessel injection (DEDVI) pipe break, which exhibit the smallest minimum core inventory. Given this, and given the fact that the equivalent diameter of (both ends) a double-ended PRHR tube rupture is 0.88 inches, the analysis of a one-inch cold leg break in the SSAR is adequate to address the small end of the break spectrum.

Question 440.56

a. Provide a description of plant instrumentation designed to operate properly during shutdown and mid-loop operations. The instrument accuracy, availability, appropriateness of key parameters (RCS level, RCS temperature, and RHR system performance), and the intended monitoring ranges should be addressed for shutdown operations.

b. Identify any deviations, and provide the technical bases and justification for these deviations, from the guidance of NUREG-1449 (Page xiii, Sections 6.6.1.1 and 7.3.3 of the report) that requests each plant to provide an independent and diverse means of accurately monitoring RCS water level, the capability to continuously monitor decay heat removal system (DHR) when a DHR system is being used for cooling the RCS, and visible and audible indications of abnormal conditions in temperature, level, and DHR system performance (see also Q440.53, Q440.55, Q440.58, Q440.71, and Q440.72).

c. Identify safety- and non-safety-related instruments used during shutdown operations. For the safety-related instruments, confirm that the instruments will be within the scope of environmental qualifications and quality assurance criteria. For non-safety-related instruments, provide a description of the quality assurance program that will be used to provide instruments with accurate information in the expected ranges of shutdown measurement that will enhance operator confidence in the instruments, and the training program for operators to understand and interpret data provided by the instruments.

Response:

a. The following table includes the RCS and related instrumentation that operate during shutdown. Instrument channel accuracy will be dependent on final intrument selection.

Instrumentation Description	Safety Class	Number of Channels	Approximate Range	Comments
RCS Hot Leg Wide Range Temperature	C	1 per hot leg	0 - 700°F	Available during shutdown operations including mid-loop
RCS Cold Leg Wide Range Temperature	C	1 per cold leg	0 - 700°F	Available during shutdown operations except for mid-loop
Core Exit Thermocouples	C	2	200 - 2200°F	Available during shutdown operations including mid-loop
RCS Wide Range Pressure	C	4	0 to 3300 psig	Available during shutdown operations including mid-loop

Instrumentation Description	Safety Class	Number of Channels	Approximate Range	Comments
RCS Hot Leg Level	C	1 per hot leg	Bottom of hot leg to top of hot leg bend into SG	Available during shutdown operations including mid-loop
Pressurizer Wide Range Level	E	1	Top of pressurizer to bottom of hot leg	Available during shutdown operations including mid-loop
RCS Flow	C	4 per cold leg	0 - 120%	Available during RCP operation (RCS temperature > 160°F)
Reactor Coolant Pump Speed	C	1 per pump	0 - 120%	Available during RCP operation (RCS temperature > 160°F)
Reactor Coolant Pump Bearing Temperature	C	4 per pump	70°F - 450°F	Available during RCP operation (RCS temperature > 160°F)
Reactor Coolant Pump Vibration	E	2 per pump		Available during RCP operation (RCS temperature > 160°F)
Reactor Coolant Pump Stator Temperature	E	1 per pump		Available during RCP operation (RCS temperature > 160°F)
RHR Pump Suction and Dicharge Pressure	D	4	0 - 900 psig	Available during shutdown after the RNS is aligned
RHR Heat Exchanger Inlet and Outlet Temperature	D	4	0 - 450 °F	Available during shutdown after the RNS is aligned
RHR Pump Discharge Flow	D	2	0 - 3000 gpm	Available during shutdown after the RNS is aligned

b. The AP600 conforms to the guidance of NUREG-1449 as follows:

NUREG Guidance:

"...provide an <u>independent</u> and <u>diverse</u> means of accurately monitoring RCS water level...."

AP600 Conformance:

Independent hot leg level channels are available to measure the level in each hot leg. In addition, independent and diverse instrumentation are available to provide an indication of core cooling during mid-loop operations. As shown in the above table, the AP600 provides diverse indication of core cooling via the core exit thermocouples, the hot leg wide range temperature, hot leg level, and RNS flow and temperature.

The AP600 provides diverse means of core cooling during mid-loop. The normal RHR system provides decay heat removal during shutdown including mid-loop operation. If the normal RHR system fails the passive core cooling system provides safety-related core cooling. The passive core cooling system is operable during mid-loop operations. Since the AP600 provides diverse means of core cooling, and provides diverse indication of core cooling during shutdown (including mid-loop), diverse hot leg level indication is not required for the AP600.

As discussed in SSAR section 5.4.7.2.1, the AP600 normal RHR system has numerous design features that significantly improve mid-loop operations. The probability of losing the normal RHR system due to errors ocurring during mid-loop operations is reduced for the AP600.

NUREG Guidance:

"provide...the capability to continuously monitor decay heat removal system (DHR) when a DHR system is being used for cooling the RCS"

AP600 Conformance:

As shown in the table above, instrumentation is provided to continuously monitor the normal RHR system as well as the RCS to monitor decay heat removal.

NUREG Guidance:

"provide...visible and audible indications of abnormal conditions in temperature, level, and DHR system performance."

AP600 Conformance:

RCS temperature and level, and normal RHR temperatures and flow indication and alarms are provided in the main control room.

c. Please see the response to (a.) regarding the safety classification of the instrumentation available during shutdown. The environmental qualification of the safety-related instrumentation accounts for the most severe environmental conditions applicable as provided in SSAR Appendix 3D. The nonsafety-related equipment class D instrumentation is designed in accordance with the requirements of ASME NQA-1-1989 Edition through NQA-1b-1991 Addenda.

SSAR Revision: NONE

Question

For the various modes of plant operation, provide the analysis and the following information: (1) the time of initiation of automatic termination of the boron dilution (if not already provided), (2) the signal that results in the initiation of the protective actions and how the corresponding time of reaching the actuation signal is determined, including the time delay for instrumentation, process parameter, and valve closure, (3) the times it would take from the start of the dilution to lose shutdown margin and reach criticality, (4) appropriate Technical Specification for the instrumentation, alarms, and valve closure time.

Response

Mode 1

As a general introduction to the Mode 1 response, it is noted that reactor operation under both manual and automatic rod control conditions are considered.

(1) In manual operation, the SSAR reports that for the limiting case, (that with the shortest time from alarm to criticality), a reactor trip occurs 203.7 seconds after start of dilution. This reactor trip signal effectively initiates the automatic termination of the boron dilution. The AP600 protection system design includes a safety related function that upon any reactor trip signal, source range flux-doubling signal, loss of offsite power, or a safety injection signal, automatically isolates the potentially unborated water from the demineralized water storage (DWS) system, thereby terminating the dilution. Additionally, the suction lines for the CVS pumps are automatically realigned to draw borated water from the CVS boric acid tank.

However, since the realignment of the suction for the CVS pumps to the boric acid tank is a non-safety related operation, the only consideration given to the reboration phase of the event in the analysis is with respect to the unborated CVS purge volume. That is, no credit is taken for any increase in the RCS boron concentration due to this realignment and instead it is assumed that the boron dilution continues until the potentially unborated piping volume is purged.

For the case with automatic rod control, a boron dilution results in a power and temperature increase in such a way that the rod controller attempts to compensate by slow insertion of the control rods. Should the dilution continue, the response of the controller would produce at least three alarms to the operator:

a) Rod insertion limit low level alarm

b) Rod insertion limit low-low level alarm (if insertion continues beyond item a)

c) Axial flux difference alarm (ΔI outside of the target band)

If the operator takes no action, despite the many alarms, the reactor may eventually trip which terminates the dilution, as described in the Mode 1 manual control discussion.

(2) During manual operation of the plant, the signal that initiates a reactor trip during manual operation and the subsequent automatic termination of the dilution is the overtemperature ΔT function. Before reaching the overtemperature ΔT reactor trip, the operator will have received an alarm on overtemperature ΔT and an overtemperature ΔT turbine runback. For the Mode 1 analysis, no formally documented response time has been defined for the post-trip dilution source isolation function. However, assuming some reasonable values makes it rather obvious that sufficient time exists for the isolation function to actuate prior to criticality. As will be discussed below (for Modes 3, 4, and 5), the maximum analysis reponse time for the boron dilution protection system (BDPS) is 3.579 minutes (214 seconds) with a 200 gpm dilution flow rate. This response includes the time required to borate the purge volume in the CVS system and the delays associated with the data-sampling and calculational logic. The post-trip isolation function, itself, has no such data processing delays associated with it. Therefore, it would be conservative to assume a total delay time approximately equal to that used for the BDPS. Applying this logic, 210 seconds has been used as the response time of the post-trip dilution source isolation function.

(3) In manual operation, the SSAR reports that for the case with the shortest time from alarm to criticality, the reactor trip occurs 203.7 seconds (3.4 minutes) after start of dilution. If the dilution were to continue, return to criticality would occur 57.73 minutes after the trip which is 61.13 minutes after the start of the dilution. However, the post-trip dilution source isolation function will terminate the dilution long before recriticality occurs.

<u>NOTE</u>: The 57.73 and 61.13 minute figures are revised values relative to those associated with Revision 0 of the SSAR. The values that correspond to Revision 0 of the SSAR are 78.33 and 81.73 minutes, respectively.

(4) Technical Specification is attached.

<u>Mode 2</u>

(1) For the Mode 2 analysis, the reactor is assumed to trip once the flux level reaches the source range high flux trip setpoint. Since this is an inadvertent dilution, no credit is taken for the the operator blocking the source range trip (permissive P-6). This trip also serves as an alarm for the operator that a reactivity transient (boron dilution, since the rods are not moving) is under way. As indicated in the Mode 1 discussion, above, a reactor trip will automatically terminate the dilution by isolating the potentially unborated water from the DWS system.

The analysis assumes that the control rods are initially at the insertion limits, thereby minimizing the available shutdown margin. The RCS boron concentration at the time of the trip is assumed to correspond to the predicted critical boron concentration for the rods at the insertion limits. With this scenario and choice of boron concentrations, the time of alarm (reactor trip) will coincide with the analysis initial condition assumption. This calculational approach therefore means that the time of the start of the dilution is not explicitly defined.

Using the conservative total delay time discussed above for the DWS isolation function, the termination of the dilution occurs 210 seconds after the time of alarm (reactor trip).

(2) Per the previous discussion, the signal assumed to terminate the boron dilution in Mode 2 is the source range high flux trip function. The basis for the 210 second delay time modeled for Modes 1 and 2 is discussed in some detail in the sections that address Modes 3, 4, and 5.

(3) Since the boron concentrations defined in the Mode 2 analysis do not define a start time for the dilution, all that can be stated is that the termination of the dilution should take place within 210 seconds after the reactor trip occurs. If, for purposes of understanding the nature of the dilution being modeled, one ignores the intervention of the safety grade DWS isolation function, the core would conservatively be predicted to return critical 76.1 minutes after the time of rector trip. Another way to look at this result is to say that, even with a conservative response time, the DWS system isolation function should automatically terminate the boron dilution 72.6 minutes before the reactor would reach criticality.

(4) Technical Specification is attached.

<u>Mode 3</u>

(1) For the limiting case under Mode 3 conditions, the time from the start of the dilution until the automatic mitigation system generates an isolation signal is 70.17 minutes. This figure includes the predicted time after the dilution starts to reach the BDPS setpoint involved (68.67 minutes) plus a 90 second algorithm delay (discussed below).

(2) In Mode 3, the function that initiates the automatic mitigation of the boron dilution is the source range neutron flux doubling signal This same signal also provides protection from a boron dilution event in Modes 4 and 5. The time delays associated with this signal for all modes are as follows:

a. Conservative fixed delay time from reaching flux-multiplication setpoint until boron dilution protection system (BDPS) microprocessor responds (algorithm delay) - 90 seconds

b. Delay from microprocessor to mitigation actuation (signal delay) - 10 seconds

c. Typical value for closure of the isolation valves from the demineralized water storage (DWS) system - 10 seconds

d. Typical value for the opening of the valves from the CVS boric acid tank to the CVS pumps - 15 seconds

e. Purge time for the CVS piping from the DWS to the RCS:

i) For 150 gpm nominal dilution flow - 119.7 seconds
ii) For 200 gpm nominal dilution flow - 89.76 seconds

f. The total delays for 150 and 200 gpm dilution flows are as follows:

i) For 150 gpm: 90 + 10 + 10 + 15 + 119.7 = 244.7 seconds (4.08 minutes)
ii) For 200 gpm: 90 + 10 + 10 + 15 + 89.8 = 214.8 seconds (3.58 minutes)

Therefore, the predicted time to reach criticality after the flux doubling setpoint has been reached must be greater than these total time delays indicated above.

(3) If the intervention of the BDPS is ignored, the core is predicted to reach criticality 73.45 minutes after the boron dilution begins.

(4) Technical Specification is attached.

<u>Mode 4</u>

(1) For the limiting case under Mode 4 conditions, the time from the start of the dilution until the automatic mitigation system generates an isolation signal is 7.35 minutes. This figure includes the predicted time (5.85 minutes) after the dilution starts to reach the BDPS setpoint involved plus the 90 second algorithm delay (discussed above).

<u>NOTE</u>: The 7.35 minute figure is revised from the figure of 10 minutes reported in Revision 0 of the SSAR.

(2) In Mode 4, the function that initiates the automatic mitigation of the boron dilution is the source range neutron flux doubling signal. The associated delay times are presented in the discussion for Mode 3.

(3) If the intervention of the BDPS is ignored, the core is predicted to reach criticality 9.67 minutes after the boron dilution begins.

(4) Technical Specification is attached.

<u>Mode 5</u>

(1) For the limiting case under Mode 5 conditions, the time from the start of the dilution until the automatic mitigation system generates an isolation signal is 8.15 minutes. This figure includes the predicted time (6.65 minutes) after the dilution starts to reach the BDPS setpoint involved plus the 90 second algorithm delay (discussed above).

(2) In Mode 5, the function that initiates the automatic mitigation of the boron dilution is the source range neutron flux doubling signal. The associated delay times are presented in the discussion for Mode 3.

(3) If the intervention of the BDPS is ignored, the core is predicted to reach criticality 11.00

minutes after the boron dilution begins.

(4) Technical Specification is attached.

Table 3.3.2-1 (page 7 of 8)
Engineered Safeguards Actuation System Instrumentation

FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS/ DIVISIONS	CONDITIONS	SURVEILLANCE REQUIREMENTS	NOMINAL TRIP SETPOINT	DAL(a)
e. Steam Generator (SG) Wide Range Water Level--Low	1,2,3,4	4/SG	F,J	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5	≥67%	±0.25%
f. RCS Depressurization	Refer to Function 7.a, 7.b, 7.c, and 7.d (RCS Depressurization) for all initiating functions and requirements					
g. Pressurizer Water Level--High	1,2,3(c)	4	F,I	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5	≤92%	±0.25%
11. Demineralized Water System Makeup Isolation						
a. ESFAC Logic	2(b),3,4,5	2 ESFAC with 2 redundant logic groups	C,J	SR 3.3.2.2 SR 3.3.2.5	N/A	N/A
b. PLC	2(b),3,4,5	2	G,J	SR 3.3.2.2 SR 3.3.2.5	N/A	N/A
c. Source Range Neutron Flux Doubling	2(b),3,4,5	4	F,J	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5	Source Range Flux Doubling in 10 minutes	±3.0%
12. Chemical Volume and Control System Makeup Isolation						
a. ESFAC Logic	1,2,3	2 ESFAC with 2 redundant logic groups	C,I	SR 3.3.2.2 SR 3.3.2.5	N/A	N/A
b. PLC	1,2,3	2	G,I	SR 3.3.2.2 SR 3.3.2.5	N/A	N/A
c. Steam Generator (SG) Narrow Range Water Level--High-2	1,2,3	4 per SG	F,I	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5	≤79%	±0.25%
d. Pressurizer Water Level--High	1,2,3(c)	4	F,I	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.5	≤92%	±0.25%

(continued)

(a) Deviation from "as left" (DAL) is the maximum acceptable deviation in percent of channel span where deviation equals the absolute value of the "as found" setpoint from the "as left" setpoint.

(b) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(c) Above the P-11 (Pressurizer Pressure) interlock.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Both logic groups on one or two divisions inoperable.	D.1 Place inoperable divisions in bypass.	6 hours
	<u>AND</u>	
	D.2 Restore 1 logic group on 3 out of 4 divisions to OPERABLE status.	24 hours
	<u>AND</u>	
	D.3 Restore 1 logic group on all 4 divisions to OPERABLE status.	48 hours
E. One channel inoperable.	E.1 Place inoperable channel in bypass.	6 hours
	<u>AND</u>	
	E.2 Restore the inoperable channel to OPERABLE status.	12 hours
F. One or tw channel(s) inoperable.	F.1 Place inoperable channel(s) in bypass.	6 hours
	<u>AND</u>	
	F.2 Restore 3 out of 4 channels to OPERABLE status.	24 hours
G. One Protection Logic Cabinet (PLC) processor inoperable.	G.1 Restore PLC to OPERABLE status.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. Required Action and associated Completion Time of Condition B, C, F, or G not met.	H.1 Be in MODE 3.	6 hours
I. Required Action and associated Completion Time of Condition B, C, F, or G not met.	I.1 Be in MODE 3. <u>AND</u> I.2 Be in MODE 4.	6 hours 36 hours
J. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	J.1 Enter LCO 3.0.3.	Immediately

BASES

ACTIONS (continued)

I.1 and I.2

Condition I is applicable when the Required Actions of Conditions B, F, or G cannot be met within the required Completion Time. The plant must be placed in a MODE where the LCO does not apply. For those LCOs that are applicable in MODES 1, 2, and 3, this is accomplished by placing the plant in MODE 3 within 6 hours and in MODE 4 within the following 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 4, these functions are no longer required OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or even something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the match criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Surveillance Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.2

SR 3.3.2.2 is the performance of an ACTUATION LOGIC TEST. Each IPC, ESFAC, and PLC are tested every 92 days using the semiautomatic tester. The division being tested is placed in the bypass condition, thus preventing inadvertent actuation. Through the semiautomatic tester, all possible logic combinations, with and without applicable permissive, are tested for each protection function. The Frequency of every 92 days is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.2.3

SR 3.3.2.3 is the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. This test is a check of the Actuation functions and is performed every 24 months. Each Actuation function is tested up to, and including, the PLC relay coils. In some instances, the test includes actuation of the end device (such as pump starts and valve cycles). The Frequency is justified in Reference 6.

SR 3.3.2.4

SR 3.3.2.4 is the performance of a CHANNEL CALIBRATION. CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor.

The Frequency is based on operating experience and consistency with the refueling cycle.

This Surveillance Requirement is modified by two Notes. The first Note states that this test should include verification that the time constants are adjusted to the prescribed values where applicable. The second Note excludes Resistance Temperature Detectors from the CHANNEL CALIBRATION. RTD channels are to be calibrated in place using cross calibration techniques or in a test bath after removal from piping.

SR 3.3.2.5

This SR ensures the individual channel ESF RESPONSE TIMES are less than or equal to the maximum values assumed in the accident analysis. Response Time testing criteria are included in Reference 9. Response time tests are conducted on a 24-month STAGGERED TEST BASIS. The 24-month Frequency

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.5 (continued)

is consistent with the typical refueling cycle and is based on plant operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

REFERENCES

1. AP600 SSAR, Chapter 6, "Engineered Safety Features."
2. AP600 SSAR, Chapter 7, "Instrumentation and Controls."
3. AP600 SSAR, Chapter 15, "Accident Analysis."
4. Institute of Electrical and Electronic Engineers, IEEE-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," April 5, 1972.
5. 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."
6. WCAP-10271-P-A, Supplement 2, Rev. 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," dated June 1990.
7. 10 CFR 50, Appendix A, "General Design Criteria for Nuclear Power Plants."
8. NUREG-1218, "Regulatory Analysis for Resolution of USI A-47," 4/88.
9. TO BE PROVIDED

W / NRC AP600 Meetings
Reactor Systems Branch Discussion Items

April 25-26, 1995

INTERFACING SYSTEMS LOCA

SECY-90-016

"Design to the extent practicable all systems and subsystems connected to the RCS to an ultimate rupture strength $\geq$ full RCS pressure."

"...systems that have not been designed to withstand full RCS pressure...
(1) the capability for leak testing of the pressure isolation valve
(2) valve position indication in the control room that is available when valve operators are de-energized
(3) high pressure alarms to warn control room operators when RCS pressure approaches design pressure of low-pressure systems and both isolation valves are not closed"

"...for some low pressure systems attached to the RCS, it may not be practical or necessary to provide a higher system ultimate rupture pressure capability for the low pressure connected systems. The staff will evaluate these exceptions on a case by case basis during specific design certification reviews."

INTERFACING SYSTEMS LOCA

AP600 Position: Full Compliance with SECY-90-016

SSAR
- Section 1.9.5
- Section 5.4.7.2.2

Responses to RAIs
- 440.132

INTERFACING SYSTEMS LOCA

Primary Sampling System

Response to 440.132 Identified Low Pressure Portions of the System

Compliance Justification
- Lines that Carry RC Design for Full Pressure
- Interfacing Lines $\leq$ 3/8"
- Automatic Isolation on Safeguards Actuation

Modifications to PSS Subsequent to 440.132 Response
- PSS Lines (and Valves, Fittings, etc.) Design P/T = RCS Design Pressure
- Low Pressure Portion of PSS Limited to Lines Vented to the Atmoshphere

INTERFACING SYSTEMS LOCA

Waste Liquid Processing System

Connects to the CVS Purification Loop

Letdown Orifice Reduces Pressure in the WLS During Letdown Operations

Leakage Limited to Capabilities of Normal Makeup System

Four Safety-Related Isolation Valves Automatically Close on ESF Signals Indicative of a LOCA

Relief Valve Provided to Protect Low Pressure Portions of the WLS

High Pressure Alarms in Low Pressure Portion of the WLS

Not Practical or <u>Necessary</u> to Design WLS to Full RCS Pressure
- Analagous to the VCT / Waste Systems in Current Plants
- Not an Issue in Current Plants
- Not Risk Important Based on PRA Results

INTERFACING SYSTEMS LOCA

Normal RHR Pump Seal

Design Pressure / Temperature
900 psig / 400 °F

Functional Requirement: Limit Leakage to < 100 gpm at RCS Operating Pressure

Normal RHR System Design Features Prevent Pump Seal Overpressurization

- RCS Isolation Valves Designed in Accordance with Provisions of SECY-90-016
- Power Removed Fom These Valves (at the MCC) During Power Operation
- Additional Isolation Valve Outside Containment Designed for Full RCS Pressure
- Normal RHR Relief Valve Inside Containment will Limit Seal Pressurization (~600 psig; ~550 gpm)

Not Practicable to Design Pump Seal to Withstand Full RCS Pressure

- High Pressure Seal would be Major Development; Less Reliable at Normal Pressure
- CVS Provides Redundant Makeup at 100 gpm
- Not Risk Important Based on PRA Results

REACTOR VESSEL LEVEL INSTRUMENTATION

TMI Action Item II.F.2

AP600 Complies with the Intent of TMI Action Item II.F.2

Important Design Features that Enable AP600 to Show Compliance to Requirements

- Adequate Core Cooling Indication Provided by Instrumentation
  - Core Exit T/Cs; HL Temperature; RCS Pressure; Pressurizer Level; HL Level
- RCPs Are Tripped on Safeguards Actuation Signal
  - Obviates the Need for Operators to Decide on RCP Trip Following an Accident
- PRHR Provides Safety-Related Heat Removal
  - Can Function with Saturated RCS Conditions
  - Obviates the Need to Maintain RCS Subcooling Following an Accident
- Automatic Depressurization System
  - RCS Level is Maintained Within the Range of the Hot Leg Level Instrumentation

REACTOR VESSEL LEVEL INSTRUMENTATION

Operator Responses Based on RVLIS for Current PWRs

- Trip RCPs Post-'S'
- Establish / Re-establish SI Flow
- Manually Depressurize RCS

AP600 Equivalent

- Automatic RCP Trip
- No Operator Actions Necessary to Initate SI Flow; Parameters Monitored Include:
  - PXS valve position indication (CMT, accumulator and IRWST discharge valves)
  - CMT water level and temperature
  - IRWST level
  - RCS pressure, temperature
  - Reactor vessel (hot leg) water level
  - RCS core exit thermocouples
  - ADS valve position indication
- Automatic Depressurization

AP600 HOT LEG LEVEL INSTRUMENTATION

Mid-loop Operation - Current Plants

- Mid-loop Operations Required For SG Maintenance
  - Level Must Be Reduced to Mid-Plane
- Vortex-Induced Air Entrainment Into RHR Pump Suction
  - Can Cause a Loss of Decay Heat Removal at Shutdown

  Suction Pipe Routing Prevents Pump Restart w/o Local Venting
- High Risk Due to Reliance on Operators
  - Low Level Margin in Hot Leg
  - Poor Level Instrumentation
  - Safety Significance Not Recognized

AP600 HOT LEG LEVEL INSTRUMENTATION

AP600 Mid-L. ., Operations

- Safety-Related Protection Provided by Passive Safety Systems
  - IRWST Automatically Injects
  - Loss of Decay Heat Removal / Loss of Inventory

- Shutdown PRA Demonstrates AP600 Reduced Risk
  - Significant Improvement Over Current Plants

AP600 HOT LEG LEVEL INSTRUMENTATION

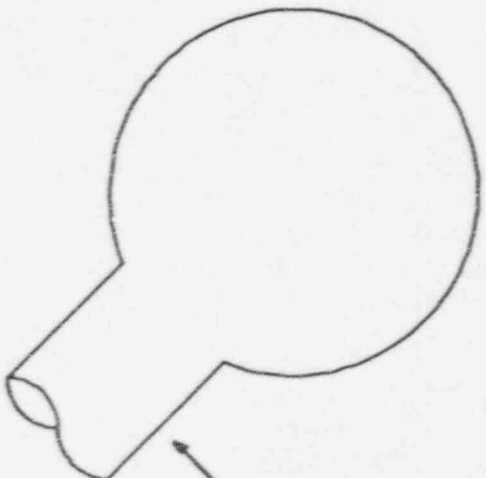
AP600 Mid-loop Operation Enhancements

- Step-Nozzle Connection to Hot Leg
  - Short Section of 20-inch Pipe Connected to Hot Leg
  - Allows Pump Operation with Lower Water Levels
  - Prevents Air Entrainment Above 5%
  - 1/4 Scale Test

- Raised Steam Generator
  - Higher Normal Mid-Loop Level (80%)
  - Increases Margin to Vortexing

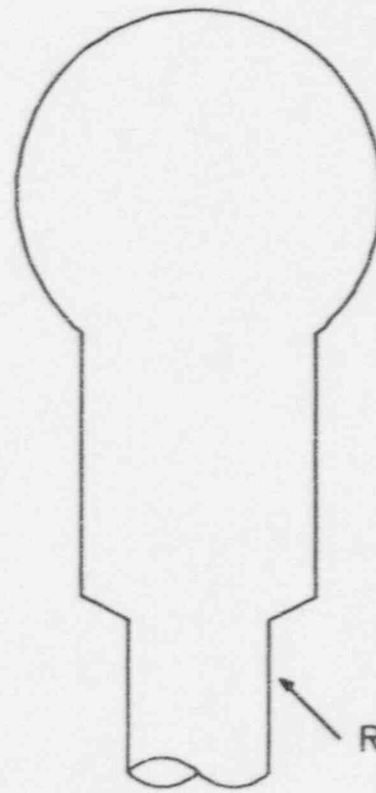
AP600 HOT LEG LEVEL INSTRUMENTATION

CURRENT PLANTS



RHR SUCTION LINE
EXITS HOT LEG AT
A 45 DEG ANGLE

AP600 STEP NOZZLE



STEP NOZZLE
20" PIPE

RHR SUCTION LINE

AP600 HOT LEG LEVEL INSTRUMENTATION

AP600 Mid-loop Operation Enhancements (cont.)

- Improved Hot Leg Level Instrumentation
  - Hot Leg Level Provided in MCR
  - Pressurizer Level Measurement Extends to Bottom of Hot Leg
  - All Operations Controlled from MCR

- Pump Suction Line Routed with No Local High Points
  - Prevents Air From Collecting in Line

- No Throttling of RHR Flow Required for Mid-Loop Operation
  - RHR Pumps Can Operate with RCS Saturated
  - Improves Reliability

AP600 HOT LEG LEVEL INSTRUMENTATION

Hot Leg Level Margins During Mid-Loop Operations

Nominal Level	Elevation (in.)	Time** (min.)
Top of Level Tap	54.37	-
Top of Hot Leg	31.00	0
Water Level for Refueling Draindown	27.74	~30
Water Level Assuming I&C Errors*	26.10	
Approximate Low Level Alarm Setpoint	22	~80
Approximate Auto-Isolation of Letdown	18	~100
"Mid-Loop Level"	15.50	
Onset of Incipient Vortex Formation	10.64	~130
5% Air Entrainment	8.60	~145
RHR Pump Operation	< 8.60	

* - Instrumentation Errors - 3% of Span

** Times Assuming Nominal Drain Rate of 20 gpm

AP600 Low-Temperature Overpressure Protection

Design Basis

Systematic Evaluation was Performed to Identify Limiting Transients

- Heat Input Events
  - Startup of an RCP with 50°F ΔT Between SG Secondary Side and RCS

- Mass Input Events
  - Two Makeup Pumps Operating at Maximum Flow Conditions

AP600 Low-Temperature Overpressure Protection

Mass Input Event

- Two Makeup Pumps Running at Maximum Flow Conditions
  - Safety-related Isolation of Makeup Pumps on High Pressurizer Level Neglected
    - Inadvertent CVS Operation
  - Assumes Water Solid Operation
  - Inadvertent Closure of Letdown PCV
  - Analysis Neglects NPSH Limitations on Makeup Pumps (Very Conservative)

Limiting LTOP Event

AP600 Low-Temperature Overpressure Protection

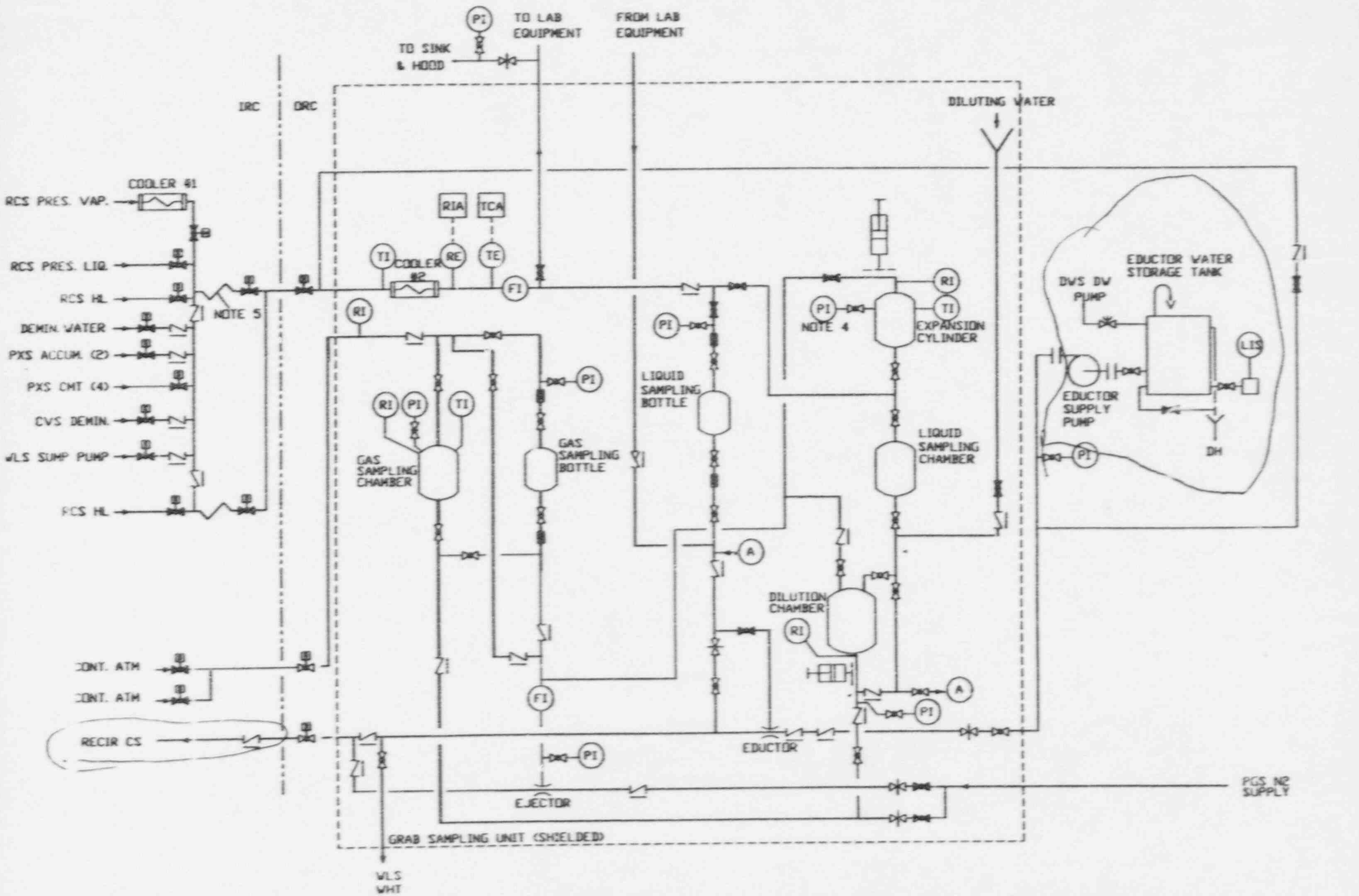
Heat Input Event

Inadvertent Startup of an RCP

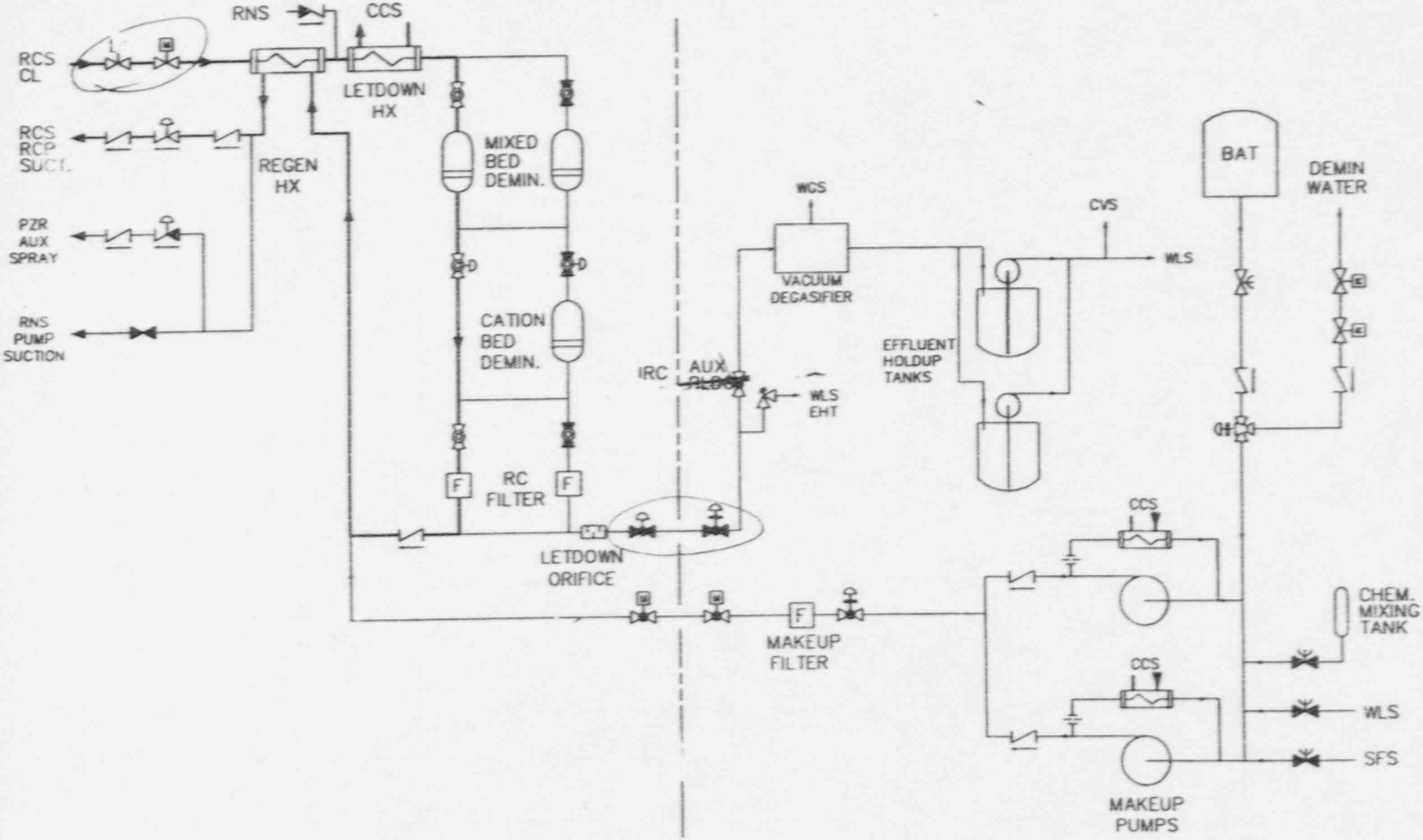
- RCS Water Solid
- SG Secondary Side 50°F Hotter than Primary System
  - Based on RCS Operations
    - Trip RCPs @ 160°F
    - Continue Cooling RCS via Normal RHR System
    - Operators Neglect to Continue Cooling SG
    - Operators Fill Pzr Water Solid
    - Operators Inadvertently Start an Idle RCP after RCS is Cooled to 110°F
- AP600 Has Applied This Transient to RCS Conditions $\leq$ 200°F
  - AP600 Design Minimizes Water Solid Operation
  - No Water Solid Operation Anticipated Above 120°F (Probably Never)
  - Administrative Controls Require A Steam Bubble Above 200°F

Evaluation of the Heat Input Event at RCS Temperatures Above 200°F Indicate Appendix G Limits are Not Exceeded

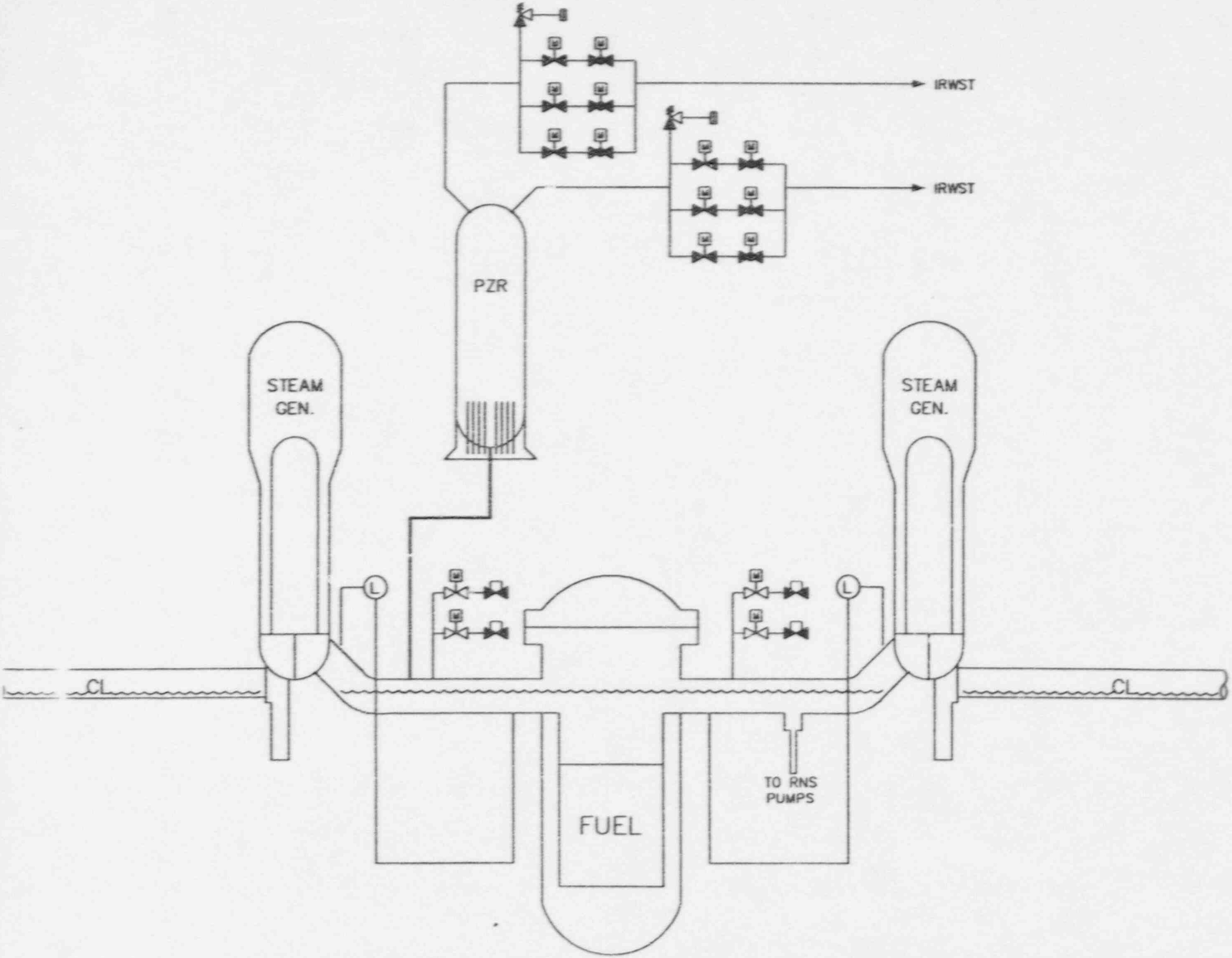
AP600 PRIMARY SAMPLING SYSTEM



AP600 - CHEMICAL AND VOLUME CONTROL SYSTEM
WITH LIQUID WASTE PROCESSING SYSTEM INTERFACES



AP600 REACTOR COOLANT SYSTEM



AP600 DRAINED SHUTDOWN CAPABILITY

- **Non-Safety Related Systems**
  - Heat removal; RNS
  - RCS makeup; CVS, RNS
  - Containment; fan coolers

- **Safety Related Systems**
  - Heat removal; passive feed/bleed
    - 23 min heatup to boiling, 2.3 hr to core uncovery
  - RCS makeup; IRWST (ADS)
  - Containment cooling; PCS
  - Tech Spec require
    - IRWST operable, ADS stage 1-3 open, PCS
    - Containment closure
      - Equipment hatches closed, air locks open but operable
      - Maintenance cables/pipes use permanent or temporary (5x12") penetrations

AP600 SHUTDOWN PERFORMANCE

- **Hand Calc of Mid-Loop Performance Shows Substantial Margin**
  - Based on conservative assumptions without ADS stage 4

	Mid-Loop (hand)	Spur ADS (hand)	Spur ADS (NOTRUMP)
- Initial RCS temp / pres	140 F 15 psia	560 F 2250 psia	564 F 2250 psia
- Time of IRWST cut-in	219000 sec	2170 sec	2176 sec
- Heat Removal, decay	0.43 %	1.69 %	2.06 %
sensible	0.00 %	0.12 %	0.39 %
total	0.43 %	1.81 %	2.45 %
- ADS Flows, stage 1-3	9.3 #/sec	12.2 #/s	9 #/sec
stage 4	0.0 #/sec	26.8 #/s	38 #/sec
total	9.3 #/sec	39.0 #/s	47 #/sec

Background

- **November 1994 Meeting**
  - Indicated that allowable times were too short for safe, controlled repair
  - Should use entire plant (safety and non-safety related) to implement Technical Specifications
  - Indicated that some mechanism would be allowed to extend allowable times to account for capability of safety systems
- **Open Item 16.1-4**
  - "Westinghouse is proposing a significant revision to LCO 3.0.3 which does not require the reactor coolant system (RCS) temperature to be reduced below 200 F."

LCO 3.0.3

Proposed Bases

AP600

Add to Bases Section as follows:

Upon entry into LCO 3.0.3, 1 hour is allowed to prepare for an orderly shutdown before initiating a change in unit operation. This includes time to permit the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach a lower MODE of operation permit the shutdown to proceed in a controlled and orderly manner. **The specified COMPLETION TIMES assume the availability of nonsafety-related systems and components that are normally used during a plant cooldown. If these systems and components are not available, the passive safety-related systems would be used to establish and maintain safe shutdown conditions. The time to reach specified conditions is expected to vary, depending upon the specific systems and components used and the initial and transitory plant conditions during the MODE change. However, the MODE change will be completed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit using the available plant systems and components. For example, the unit can be brought to MODE 4 in approximately 37 hours using the passive residual heat removal subsystem under design basis plant conditions** This reduces thermal stresses on components of the Reactor Coolant Systyem and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. The use and interpretation of specified times to complete the actions of LCO 3.0.3 shall be consistent with the discussion of specification 1.3 "Completion Times".

AP600 CHANGES

T. L. SCHULZ
SYSTEMS ENGINEERING
APRIL 26, 1995

RECENT AP600 CHANGES

- **Changes**
  - Increased Pressurizer volume
  - Elimination of one PRHR HX
  - Use of squib valves in IRWST injection lines
  - Use of squib valves in ADS stage 4 lines
  - Use of autocatalytic H2 recombiners
  - Startup Feedwater changes (flow, connection)

INCREASED PRESSURIZER SIZE

AP600

- **Change**
  - Total volume increased from 1300 ft3 to 1600 ft3
  - Full power liquid volume increased from 830 to 1010 ft3
  - Inside dia. increased from 84 in to 90 in

- **Basis**
  - Increase margins (similar to RCP / SG)

- **Impact**
  - Testing; none, computer codes can handle difference
  - Analysis; affects many analysis, will be included in 5/95 submittal
    increases time for operator action to prevent Pzr overfill
  - PRA; none, no affect on success criteria

ELIMINATE ONE PRHR HX

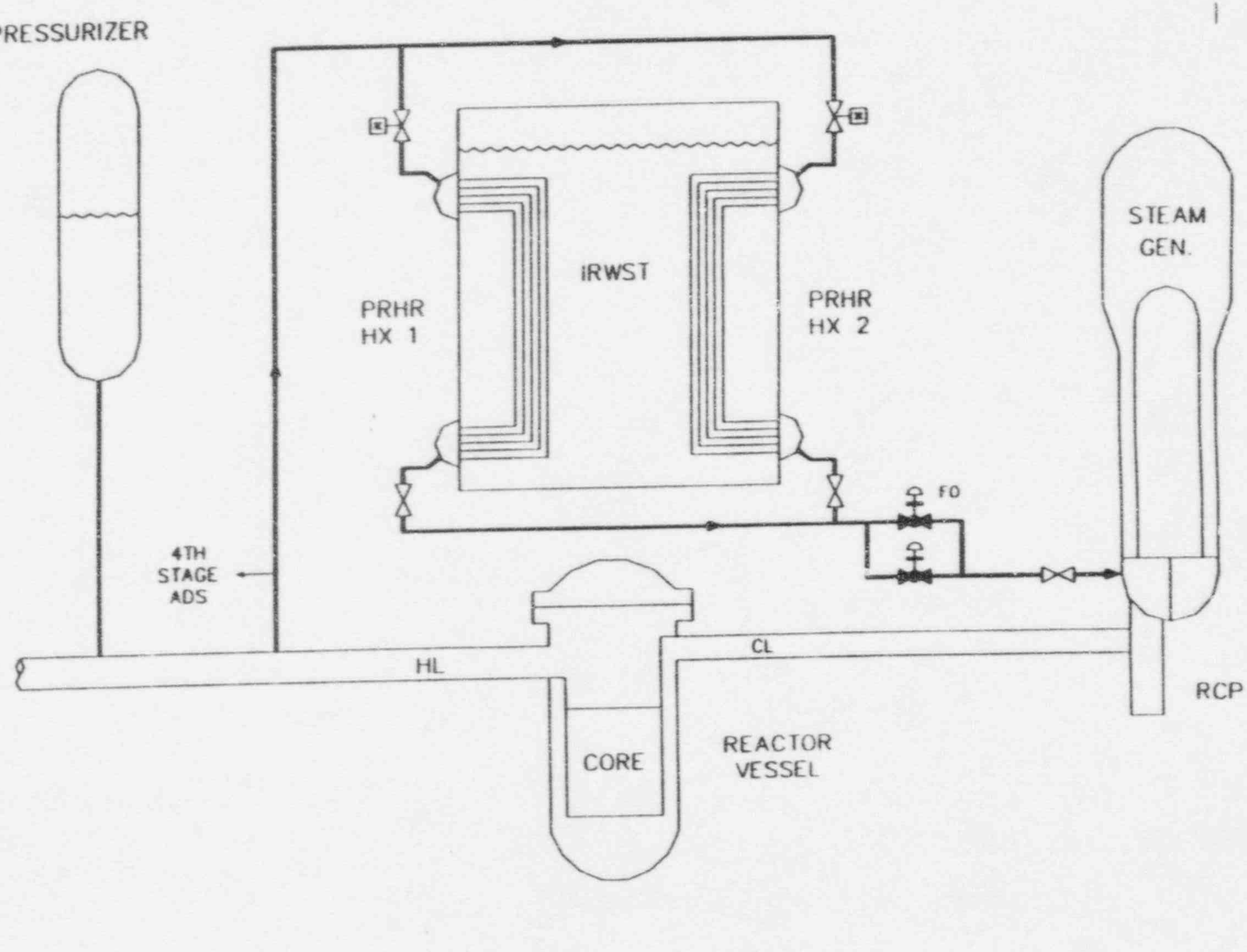
AP600

- **Change**
  - Eliminate one PRHR HX
  - Eliminate associated inlet / outlet valves
  - Add Tech Spec to allow limited leakage (ala SG)

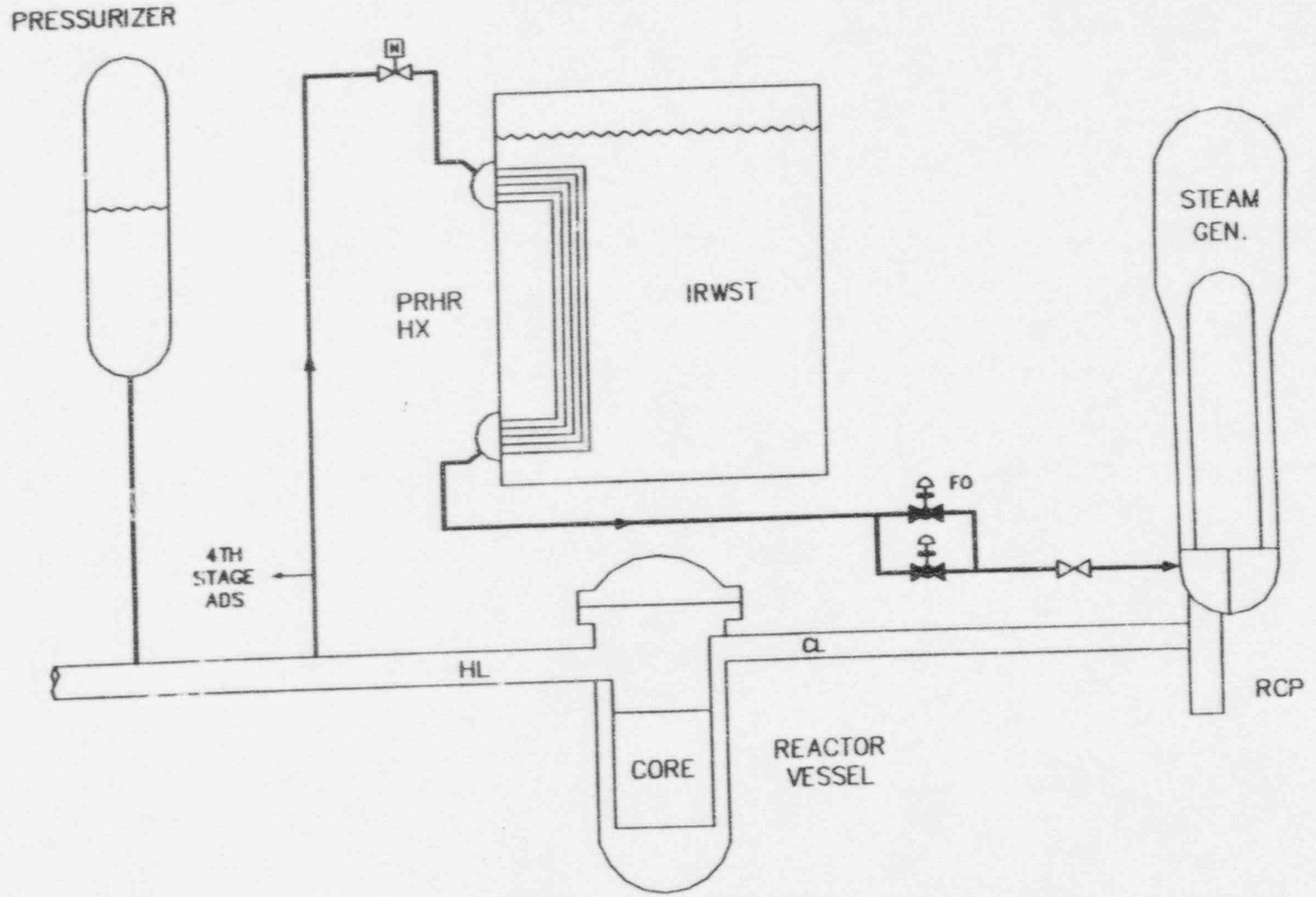
- **Basis**
  - No affect on limiting heatup accidents
    - AP600 Tech Spec only require one HX
  - Reduces chance of accidents (PRHR HX tube rupture)
  - Reduces excessive RCS cooldown / thermal stratification

- **Impact**
  - Testing; none, most done with one HX
  - Analysis; none on heatup events
    improves cooldown events
  - PRA; none, no affect on success criteria

AP600 PASSIVE RHR SYSTEM (OLD)



AP600 PRHR HX SYSTEM (NEW)



USE SQUIB VALVES IN IRWST

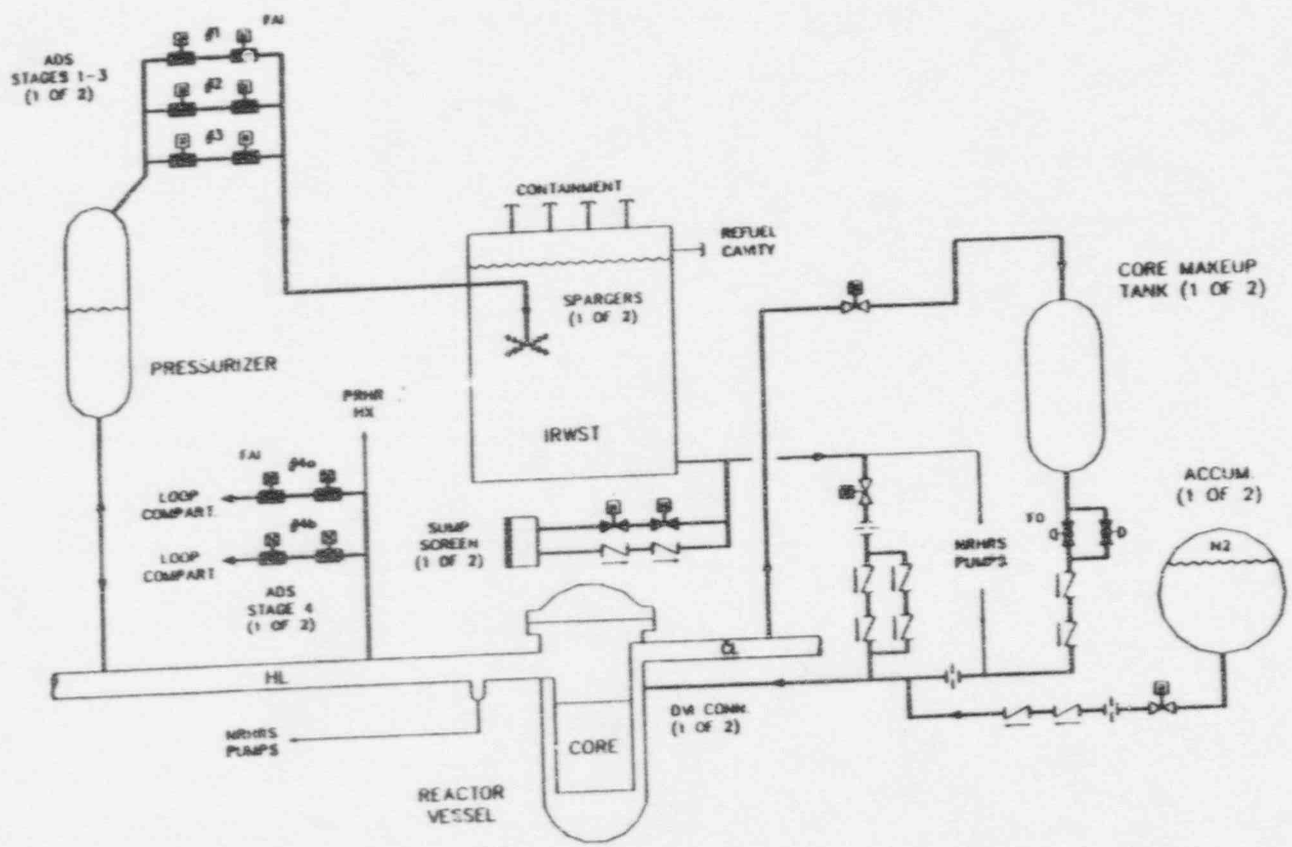
AP600

- **Change**
  - Replace 4 check valves in IRWST injection lines
    - Actuate on ADS stage 4 signal
  - Replace 2 check valves & 2 MOVs in containment recirc lines
    - Actuate on low IRWST level (same as current MOVs)

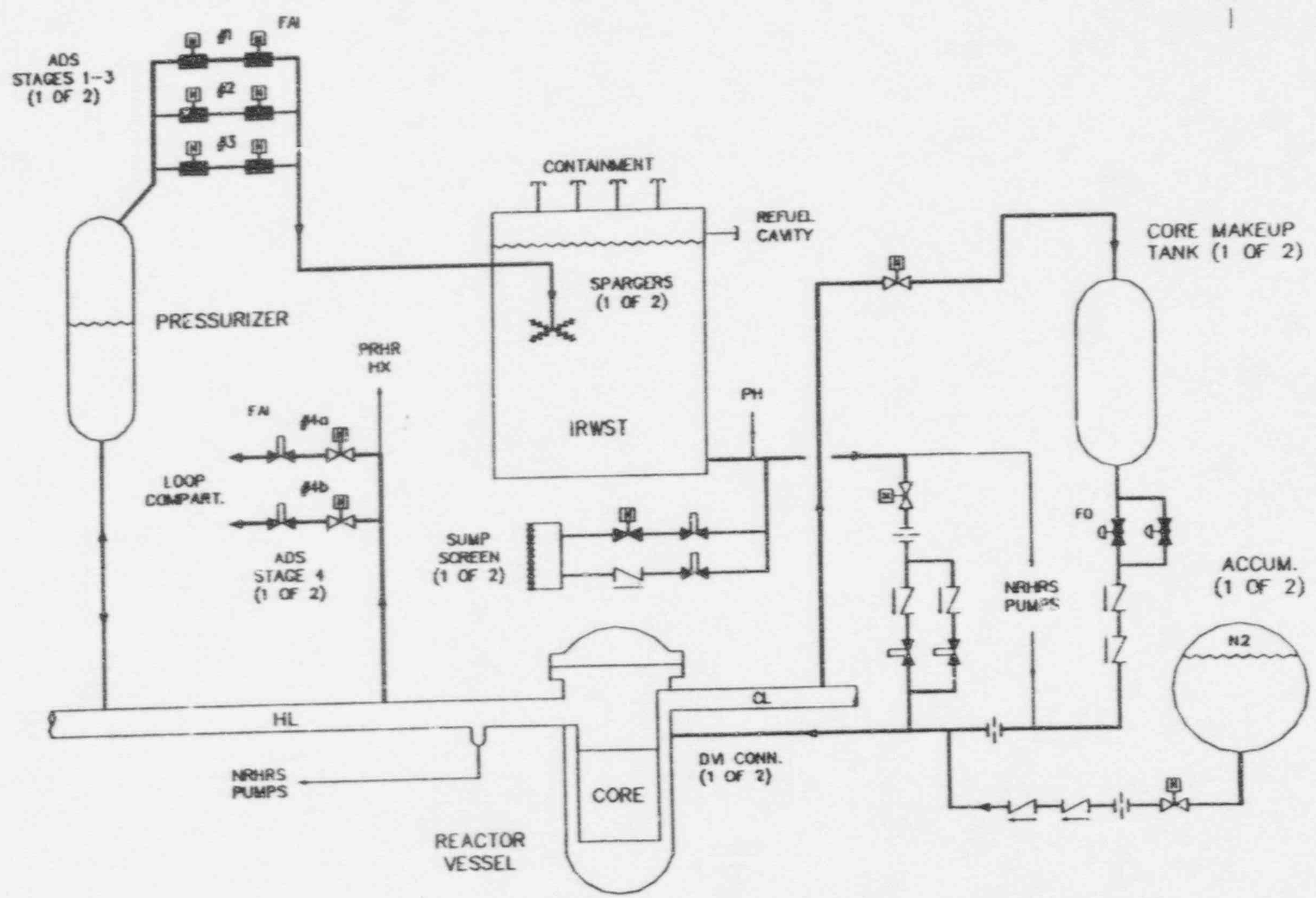
- **Basis**
  - Reduce chance of leakage during normal operation
  - Reduce uncertainty in IRWST check valve opening
    - IRWST check valves now only see low DP

- **Impact**
  - Testing; none, valves have save L/D, lines open at same time
  - Analysis; none, valves have same L/D, squibs opened at same time or before check valve open
  - PRA; minor, actuation / dc power already required for ADS

AP600 PASSIVE SI SYSTEM (OLD)



AP600 PASSIVE SI SYSTEM (NEW)



USE SQUIB VALVES FOR ADS STAGE 4

AP600

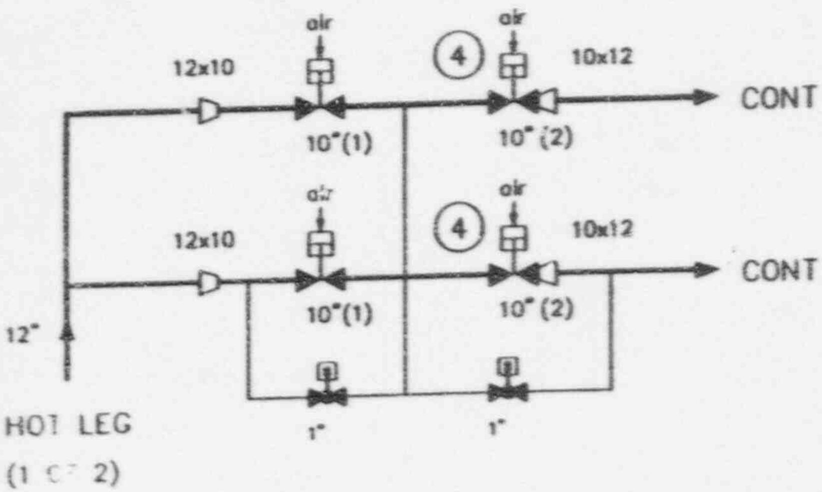
- **Change**
  - One of approaches identified in 2/15/94 change report

- **Basis**
  - Reduced chance of leakage during normal operation
  - Reduced ISI / IST efforts
  - Reduced maintenance
  - Very reliable

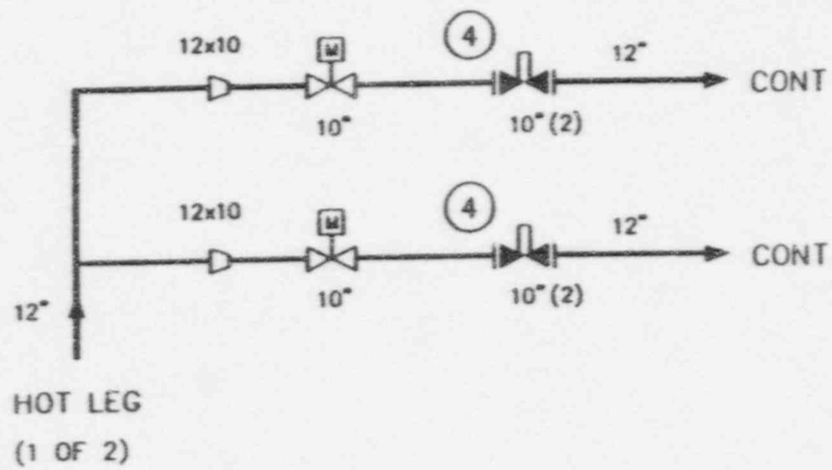
- **Impact**
  - Testing; none, already incorporated in tests
  - Analysis; none, evaluated in 2/15/94 change package
  - PRA; none, already incorporated in PRA rev 2

ADS STAGE 4 EXAMPLES

EXAMPLE 1



~~EXAMPLE 2~~ NEW



NOTE: (1) Upstream isolation valve sequenced open before opening down stream control valve.

(2) Stage 4 ADS valves blocked from opening when RCS > 1200 psig.

RECOMBINER CHANGE

- **Change**
  - Use 2 autocatalytic H2 recombiners
    - Replaces 2 electric recombiners
    - Required for DBA LOCAs
- **Basis**
  - Simplifies design
  - No electric power required
- **Impact**
  - Same or better H2 recombination
  - Reduced IST on electrical / I&C
  - Simplified post 72 hour design / operation

SFW CHANGES

AP600

- **Change**
  - Increased flow from 500 to 540 gpm to both SGs
  - Changed connection from MFW line to SG

- **Basis**
  - Increase margin (same as RCP / SG)
  - Reduce thermal transients on MFW line / SG nozzle

- **Impact**
  - Testing; minor, computer codes can handle
  - Analysis; minor impact on cooldown events
  - PRA; minor improvement for separate connection

AP600 – STARTUP FEEDWATER SYSTEM (NEW)