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NSD-NRC-97-5113
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Docket No.: STN-52-003

May 8, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: AP600 RESPONSE TO REQUESTS FOR ADDITIONAL INFORMATION

Dear Mr. Quay:

Enclosed are the Westinghouse responses to NRC requests for additional information and meeting action items pertaining to the AP600 PRA modeling of steam generator tube rupture (SGTR) events. Specifically, responses are provided for RAIs 720.371and 720.372, and meeting action items 1 & 3 from a meeting held on January 21 and 22, 1997 to discuss AP600 PRA.

These responses close, from the Westinghouse perspective, the RAIs. The NRC should review these responses and inform Westinghouse of the status to be designated in the "NRC Status" column of the OITS. The OITS numbers associated with the RAIs are 5119 and 5120, and the numbers associated with the meeting action items are 5014 and 5016.

Please contact Cynthia L. Haag on (412) 374-4277 if you have any questions concerning this transmittal.

Brian A. McIntyre

Brian A. McIntyre, Manager
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jml

Enclosure

cc: J. Sebrosky, NRC (Enclosure)
J. Flack, NRC (Enclosure)
A. Levin, NRC (Enclosure)
N. J. Liparulo, Westinghouse (w/o Enclosure)

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Enclosure to Westinghouse
Letter NSD-NRC-97-5113

May 8, 1997

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Question: 720.371 (OITS #5119)

Please explain the events and assumptions of cut set #2 (IEV-SGTR * ADF-MAN01 * RPX-CB-GO * AND-MAN01C). In particular, the staff requests the following:

a. Are there any T-H analyses to support this sequence? Can the leak be stopped before uncovering the core or passing water through the secondary side safety valves?

b. How fast must the operator act to open ADS stage #1 valves (event ADF-MAN01), given that this is not the preferred means? As is documented in the PRA, the operator will try to align CVCS in auxiliary spray mode first. According to the HRA, p. 30-29 of PRA, this action requires a long procedure and about 10 minutes of actual implementation time. Are there any procedures the operator must follow? Does event ADF-MAN01 correspond to a system level actuation or to actuation of individual stage 1 valves using PLS?

c. This scenario assumes that even when the operator action ADF-MAN01 fails, the accident can be mitigated by manually depressurizing the RCS using ADS (event AND-MAN01C). On what event(s) is the probability of event ADN-MAN01C "conditional?" How much time does the operator have to perform this action to avoid uncovering the core or overfilling the SG, given the other event(s) will have to be diagnosed and potential actions completed first? Is the modeling of this scenario in agreement with the procedure that operators must follow? Please explain by referring to HRA and other analyses documented in the PRA.

d. Cutset #34 (IEV-SGTR * RPX-CB-GO * ADN-MAN01) and cutset #65 (IEV-SGTR * RPX-CB-GO * LPM-MAN01) imply a different emergency response procedure than cutset #2 for same scenario. What do the emergency response procedures instruct the operator to do when a SGTR event is followed by failure to trip of one or more RCPs? If the operator is instructed to depressurize the RCS, what are the times available for diagnosis and action? Please provide the basis for the assumed success criteria for the systems used to mitigate the accident and for the time windows used in HRA.

Response:

Note: This RAI is directed at the sensitivity study performed by Westinghouse, at the request of the NRC, and provided in Westinghouse letter NSD-NRC-96-4913, dated 12/13/96.

a&b This cutset from the sensitivity study specified and agreed to by the staff represents the failure of the operator to actuate the ADS combined with the failure of the RCP breakers to trip. Note this cutset is not among the top 200 cutsets in the focused PRA. The success criteria represented by this cutset is not in the focused PRA, and there can be no relevant discussion of it for the purpose of understanding the focused PRA. The only relevant discussion of this question is for the purpose of clarifying the plant response to an SGTR, since that is a fundamental aspect of the AP600 design.

Since the staff requested that the sensitivity be done without credit for the CVS, there is no relevance to the CVS (or the pressurizer spray) for this cutset. (The words noted on the ADF-MAN01 basic event indicate a failure of the pressurizer spray, because the basic event originates from the baseline PRA, as was agreed to by the staff for the method of doing the sensitivity.) The description of the ADF-MAN01 action used in this

cutset is on page 30-18 of the PRA, where the timing of the operator action is mentioned. That description also notes that a time for the operator action similar to what would be seen for a medium LOCA is assumed, although a tube rupture is more like a small LOCA (relative to the time required to perform an action) than a medium LOCA. Thus, the probability of failure for the operator action is larger than would be expected for this event in the plant, and the failure probability for this cutset is larger than would be expected.
There are T-H analyses to support this sequence and were discussed at the meeting of January 22, 1997. These analyses show that there is more time for the operators to act than is assumed in the PRA analysis, and the leak from the primary side to the secondary side is stopped before uncovering the core. The analyses also show, as was discussed on January 22, 1997, that the safety valves are not challenged. With the reactor coolant pumps (RCPs) running, the passive residual heat exchanger (PRHR HX) removes the decay heat exceptionally well (forced convection through the heat exchanger). The removal of heat from the RCS depressurizes the primary side and stops the leak flow through the broken tube in less than 10 minutes. This is shown in Figure 720.371-1. Although a computer simulation is not essential to understand this principle of thermal-hydraulic analysis, the figure (and the other figures that follow) is from a computer run done specifically to satisfy this question. At no time in this transient does the core uncover. This is shown in Figure 720.371-2. That figure shows the RCS temperature compared to the saturation temperature (T_{sat}). Note that the RCS temperature drops rapidly and does not increase. Figure 720.371-3 shows the heat removal from the RCS through the PRHR HX and the heat input to the RCS, the decay heat. The heat removal from the RCS is significantly larger than the decay heat during and after the transient. The RCS level is shown in Figure 720.371-4. This figure shows the level remains above the core until the ADS opens. The level then drops to the hot leg level (where the ADS valves are). This is above the level of the core.

As shown in Figures 720.371-1 and 720.371-4 the operators do not need to open the ADS stage 1 valves to depressurize the RCS. The leak flow is stopped by the heat removal through the PRHR HX and the PRHR also depressurizes the RCS. The PRA model conservatively assumes that the RCS must be depressurized to the containment atmosphere pressure to terminate the transient. The PRA also assumes that the depressurization through the ADS further requires the injection of the IRWST water and the success of the recirculation function. The analysis shows the PRA is a very conservative model, because the depressurization through the ADS is not needed. Thus, the operators have longer than is modeled in the PRA to act to open the ADS stage 1 valves. The preferred means of doing something is to follow the Emergency Response Guidelines (ERGs) as was discussed in the response to RAI 720.326. Those guidelines will direct the operators to depressurize the RCS, and use the ADS if that is what is needed.

In the focused PRA, the PLS is not available, so the operators cannot be using the PLS in the focused PRA models. ADF-MAN01 models the actuation of ADS. The operator action is essentially the same for a full RCS depressurization or a partial depressurization. In either case, the PMS is expected to be used for manual actuation of the ADS, and this is the model for the focused PRA.

c. Any operator action that is conditionalized has a failure probability based upon the failure of a previous operator action. This situation is found throughout the focused PRA and the baseline PRA.

This sequence is not in the PRA, but there is merit in clarifying the fundamentals of the timing and operator actions in the models for the plant response to a SGTR. As discussed in the previous pages, there is a large

amount of time for the operators to respond to a SGTR and avoid core uncovery or overfill of the steam generators. The PRA models conservatively assume only 30 minutes are available, and the failure probabilities are based upon this assumption. The operator actions modeled in the response to a SGTR in the PRA analyses are based upon procedures.

d. The success criteria used for the baseline PRA are discussed in Chapter 6 of the PRA. The success criteria for the focused PRA are those in the baseline PRA without the non-safety systems as discussed in SECY-95-132 and Chapter 52 of the PRA. The success criteria for the sensitivity analysis are likewise those in the baseline PRA without those systems specified by the staff.

As discussed in Chapter 30 of the PRA, the ADN-MAN01 action refers to the actuation of the ADS. The LPM-MAN01 action refers to the diagnosis of the need for RCS depressurization. These are not different emergency response procedures, but basic components of the operator action models.

If the RCPs fail to trip, and there is a tube rupture, the operators will enter the tube rupture procedure, AE-3. Since the leak flow will stop very quickly (as discussed in the previous pages), the need to actuate the ADS is not expected to arise. If one assumes that the leak flow does continue, contrary to the T/H analysis, then the CMT level will drop and the operators will be instructed by the procedures to depressurize the RCS. The previous discussions show there is a large amount of time for the operators to do this.

The basis for the success criteria in the baseline PRA and the focused PRA are the AP600 integral tests, the AP600 safety analyses, and T/H analyses. These analyses have included computer simulations of the transients modeled in the success criteria.

PRA Revision: None.

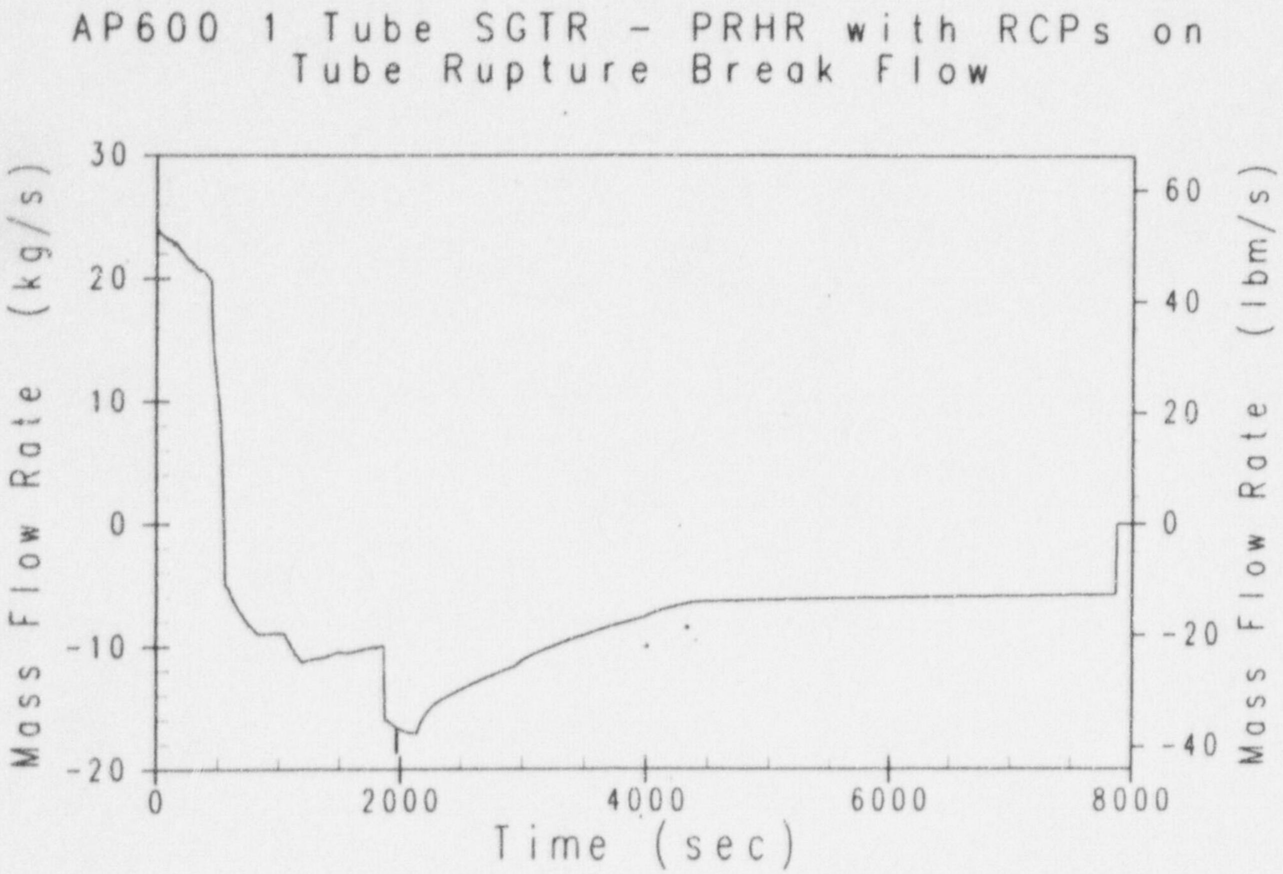


Figure 720.371-1

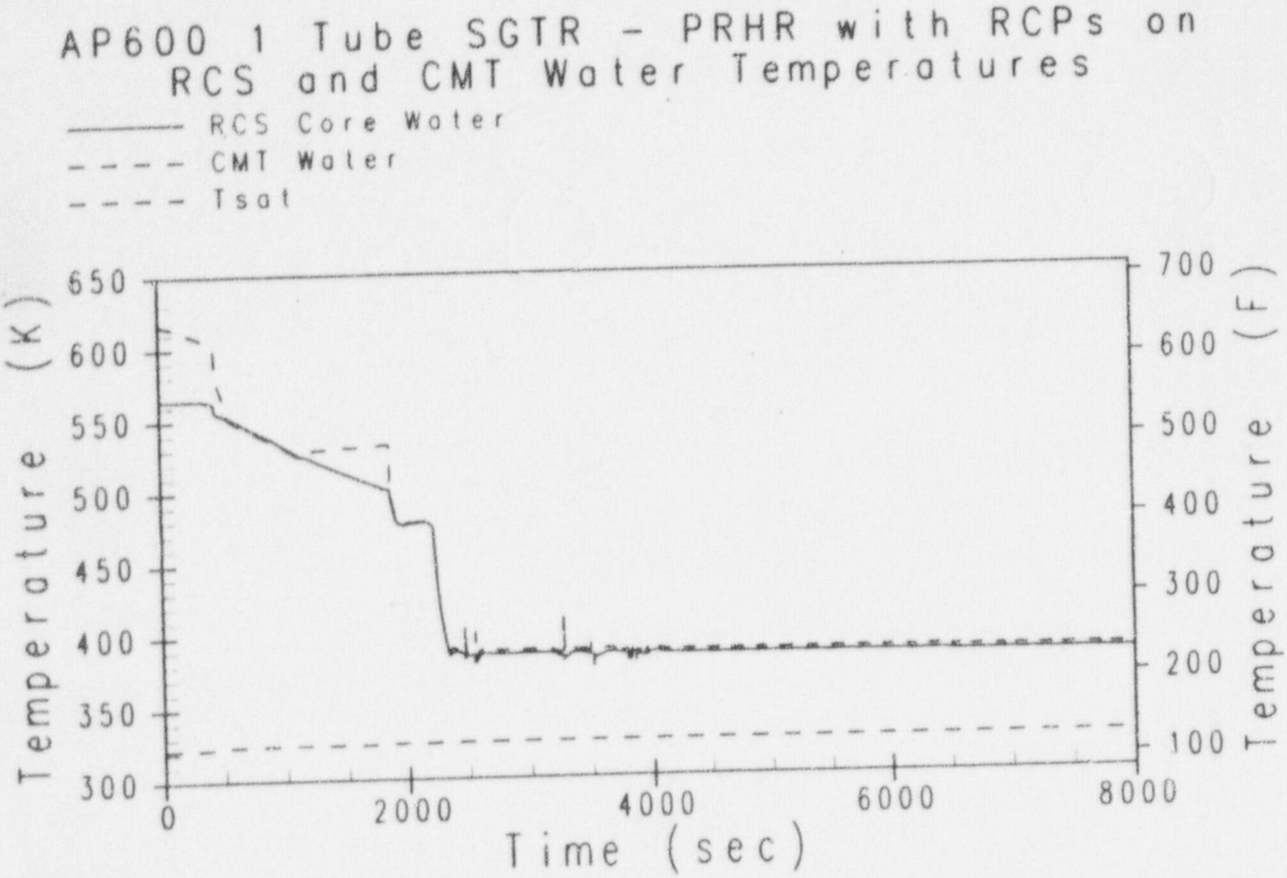


Figure 720.371-2

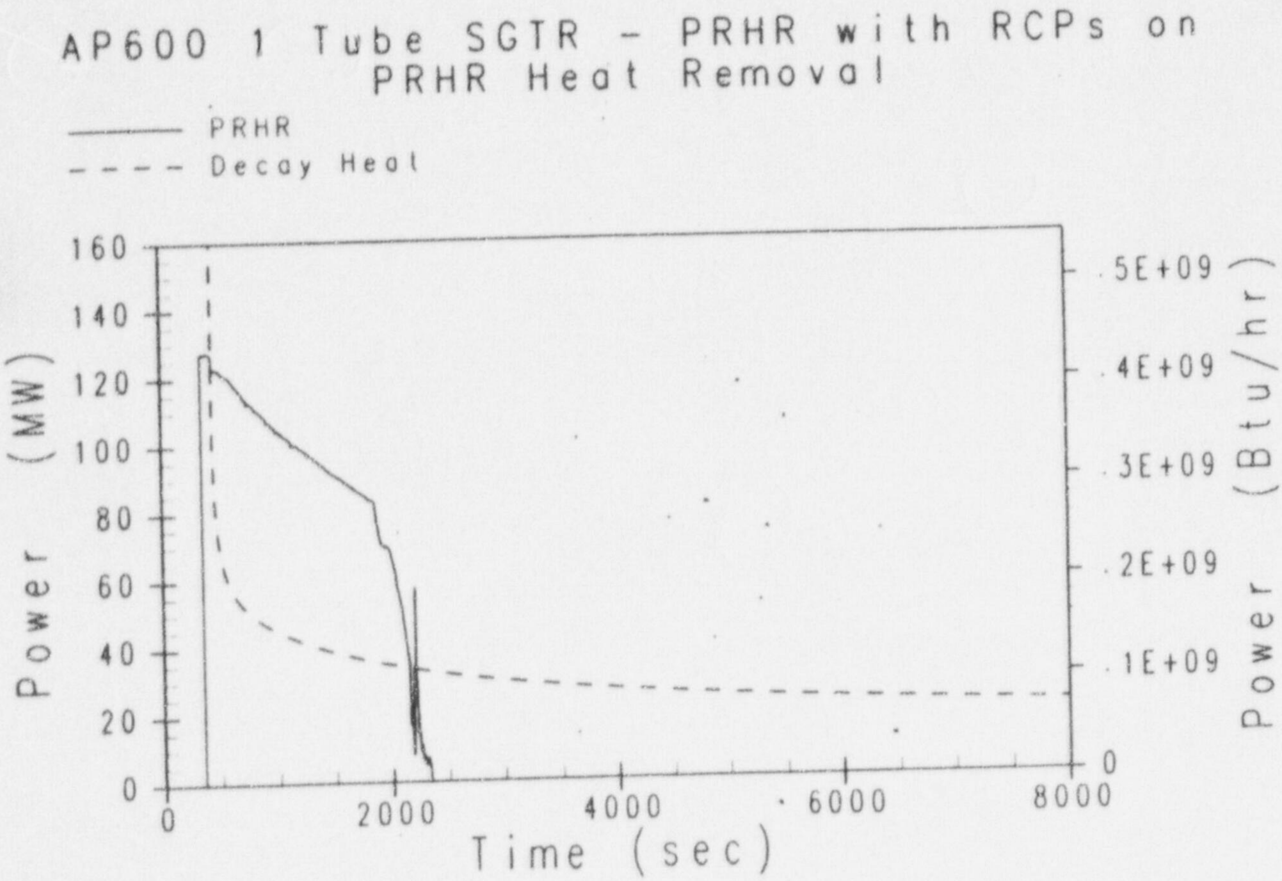


Figure 720.371-3

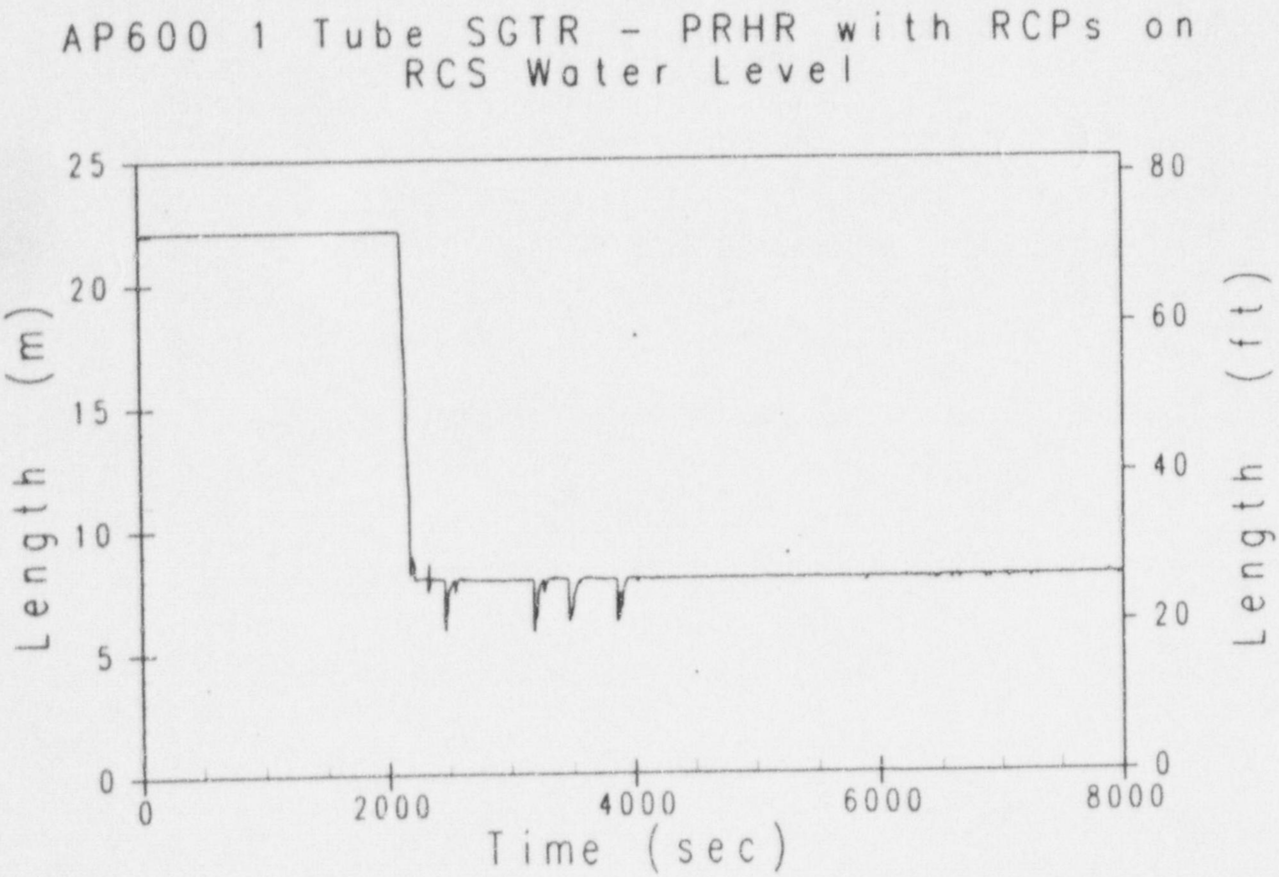


Figure 720.371-4

Question: 720.372 (OITS #5120)

Several cut sets (such as #3, #6 and #7) include common cause software failure across both PMS and PLS (event CCX-SFTW). It is not clear why event CCX-SFTW is considered, given that PLS is not supporting any system credited in the analysis, instead of common cause software failure within PMS only (event CCX-PMXMOD1-SW). Please explain.

Response:

The cutsets from the sensitivity study of concern were generated using the criteria and method specified and agreed to by the staff. The method included the assumed failure of the PLS. Other conservative events, such as CCX-SFTW, were not removed from the cutsets. Event CCX-SFTW is considered, because it was in the cutsets that formed the basis for the sensitivity study (according to the specification of the sensitivity) and not in the set of systems that were removed. The CCX-PMXMOD1-SW event is the common cause failure of the output part of the PMS. (This does not include the failure of the inputs. The failure of CCX-PMXMOD1-SW would not prevent the operators from getting good indications of the plant conditions and taking whatever action is possible.)

There are no cutsets with this type of situation in the focused PRA.

PRA Revision: None.

Meeting Action Items from NRC summary of January 21 and 22, 1997 Meeting on AP600 PRA

Action Item: 1. (OITS #5014) The staff asked Westinghouse to perform thermal-hydraulic analyses as necessary to support the success criteria assumed in the SGTR event tree. [A specific staff concern is ...] Lack of thermal-hydraulic analyses supporting the assumption that the AP600 design can rely on non-safety related systems only as a first line of defense to mitigate SGTR accidents (sequence 1 SGR-OK1 in event tree, see page 4-128 of PRA).

Response:

The AP600 provides three different responses to a SGTR. One response uses active features and operator actions that are similar to current plants. This is the only success path without safety systems in the PRA models. It is shown in Chapter 4 of the PRA. The other two responses use passive features that are automatic and independent of AC power. These three responses provide significantly greater redundancy and diversity than is provided by current plants which results in significant risk improvements. Figure 1 illustrates these different responses including several possible variations of the AP600 ADS based response.

Active System Response:
This response uses non-safety related active features and operator actions to equalize the RCS and SG pressure and terminate the leak. Initially the leak into the SG results in the pressurizer (Pzr) level and pressure dropping because the CVS makeup flow can not keep up with the leak flow. The reactor automatically trips on a low pressurizer (PZR) pressure. The AP600 startup feedwater pumps are automatically actuated to provide SG feedwater that is automatically controlled based on SG level; this minimizes the potential of overfilling the faulted SG. One startup feedwater pump feeding the intact SG provides sufficient water to remove decay heat and support the RCS cooldown operations conducted during SGTR mitigation. One CVS makeup pump is automatically actuated and controlled based on pressurizer level. Although the pressurizer may empty, the pump provides sufficient RCS makeup to prevent further voiding of the RCS.

In this response, the operator is required to perform several manual actions to terminate the leak. One action is to isolate the faulted SG by closing the associated main steam isolation valve. Another action is to cool the RCS using the intact SG. To cool down the RCS, the operator reduces the control setpoint for the turbine bypass valves or the SG power operated relief valve to vent sufficient steam from the intact SG to reduce its pressure and temperature. The RCS pressure is also reduced by the operator using the CVS auxiliary spray. The operator can also partially open and re-close one ADS stage 1 valve to make a limited reduction in RCS pressure.

The use of these active features provides the operator sufficient time to terminate the leak without overfilling the faulted SG assuming the rupture of one to five SG tubes. This is shown in Figures 2 through 6. These figures are from a MAAP4 simulation of the success sequence with active systems modeled in the PRA.

Figure 2 shows the leak flow from the ruptured tube for this success path. This should be contrasted with Figure 3, which is the CVS flow into the RCS. It can be seen that the leak flow quickly drops to the same as the CVS flow (in about 17 minutes). This is supported by Figure 4, which shows the rapid equalization of the RCS and steam generator pressure. Once the primary and secondary pressures are the same, the leak stops. Figure 5 shows the pressure in the faulted steam generator and the safety valve setpoint. That figure shows the safety valves are not challenged in the transient. Figure 6 shows the level in the faulted steam generator. The computer simulation was terminated at 5.7 hours when the CVS is automatically isolated due to the faulted steam generator level reaching the 79% narrow range level.

While this response is similar to current plants, there are some differences. One notable difference is that current plants have high head safety injection pumps that have greater capacity than the AP600 CVS makeup pumps. This greater pumped capacity maintains a higher RCS pressure relative to the SG pressure. As a result, the leak flow into the SG is greater in current plants, and the time available for the operators to take action before the SG

overfills is less in current plants.

The first success path (event tree path 1) in the SGTR event tree requires success at the following nodes to be successful:

Reactor trip (RTRIP),
CVS makeup pumps (CVS),
Startup feedwater pumps (SFW),
RCS cooldown via intact SG (SGDEP),
RCS pressure reduction (PRDEP), and
Isolation of the faulted SG (SGISO).

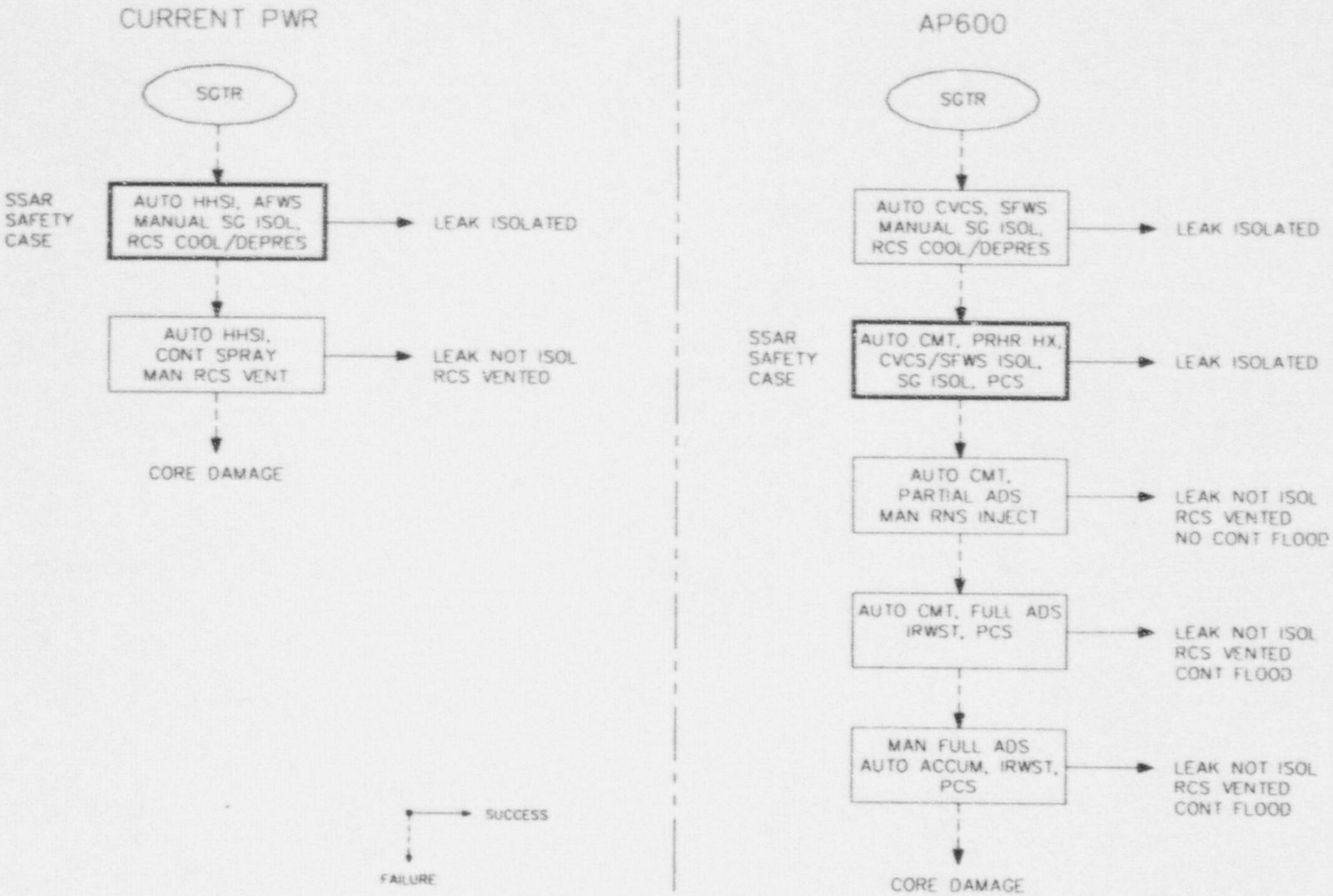
If any of these nodes fail then the passive systems are assumed to be necessary for success.

Action Item: 3. (OITS #5016) The staff asked Westinghouse to perform thermal-hydraulic analyses as necessary to support the success criteria assumed in the SGTR event tree. [A specific staff concern is ...] Lack of thermal-hydraulic analyses supporting the success criteria and modeling of sequences involving failure to trip of one or more RCPs.

Response:

This is addressed in the response to RAI 720.371.

Figure 1 - AP600 SG Tube Rupture Levels of Defense



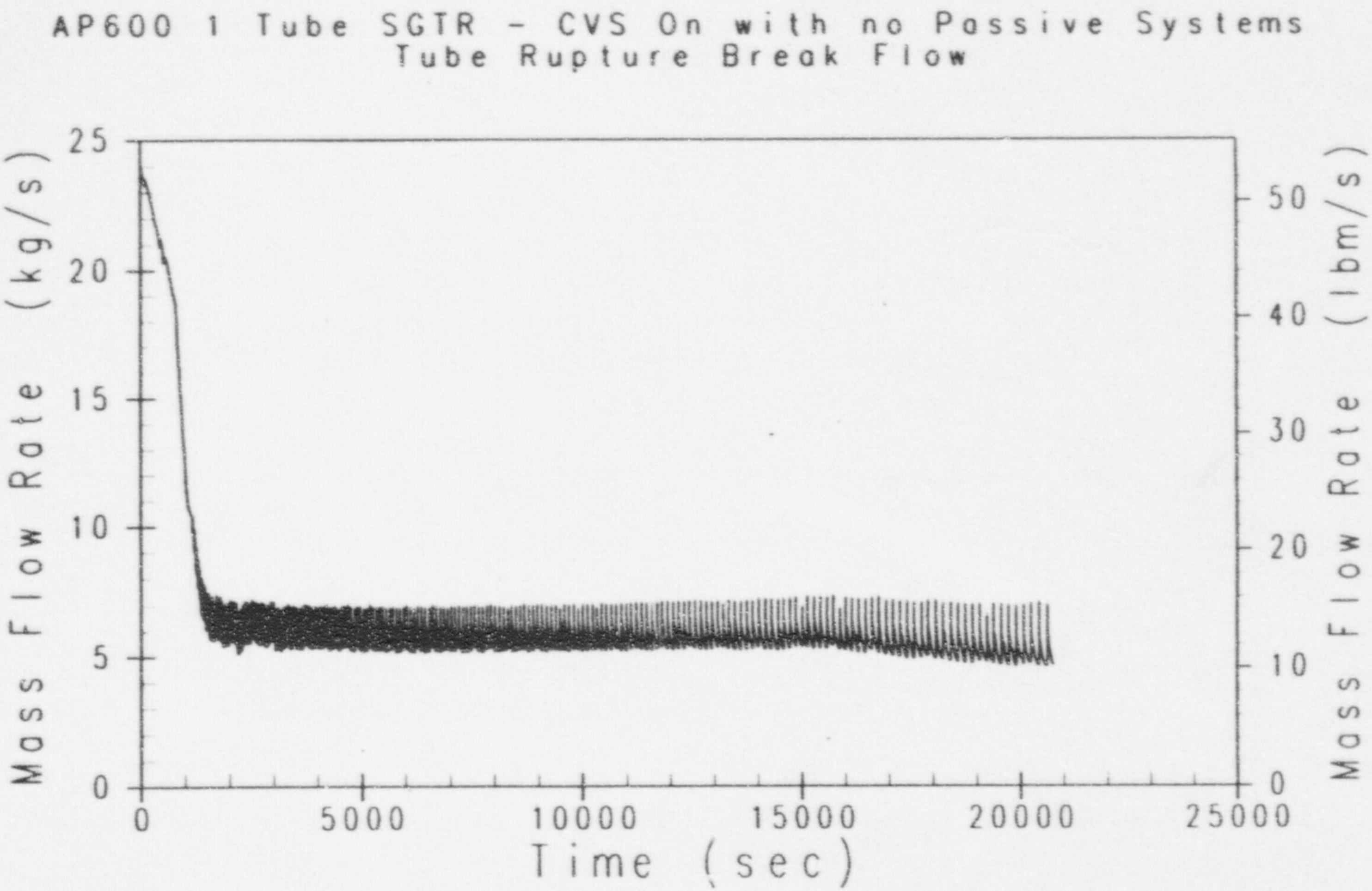
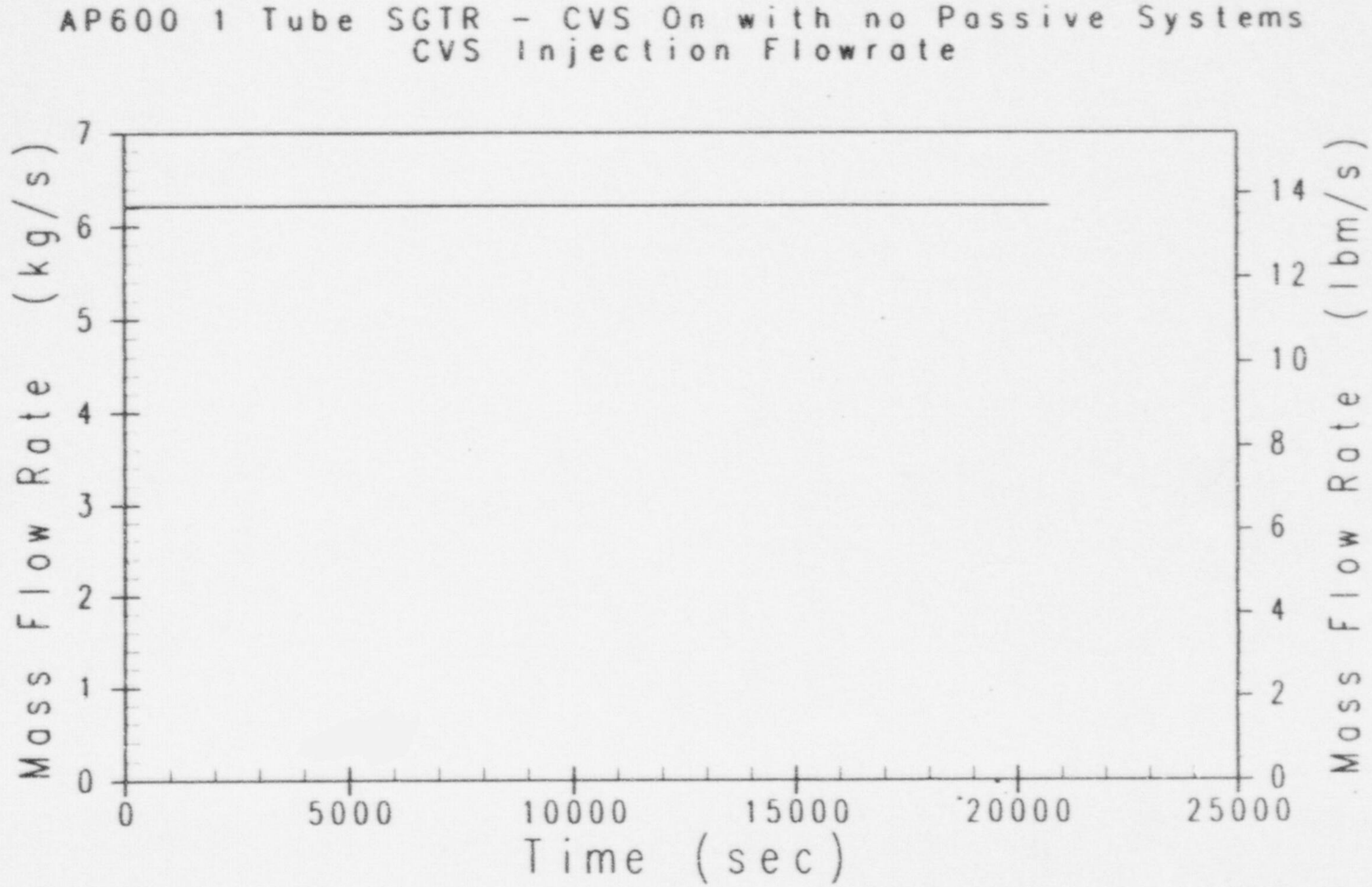

AP600 1 Tube SGTR - CVS On with no Passive Systems
Tube Rupture Break Flow
Mass Flow Rate (kg/s)
Mass Flow Rate (lbm/s)
Time (sec)
0
5
10
15
20
25
0
10
20
30
40
50
0
5000
10000
15000
20000
25000

Figure 2



AP600 1 Tube SGTR - CVS On with no Passive Systems
CVS Injection Flowrate
Mass Flow Rate (kg/s)
Mass Flow Rate (lbm/s)
Time (sec)
0
1
2
3
4
5
6
7
0
2
4
6
8
10
12
14
0
5000
10000
15000
20000
25000

Figure 3

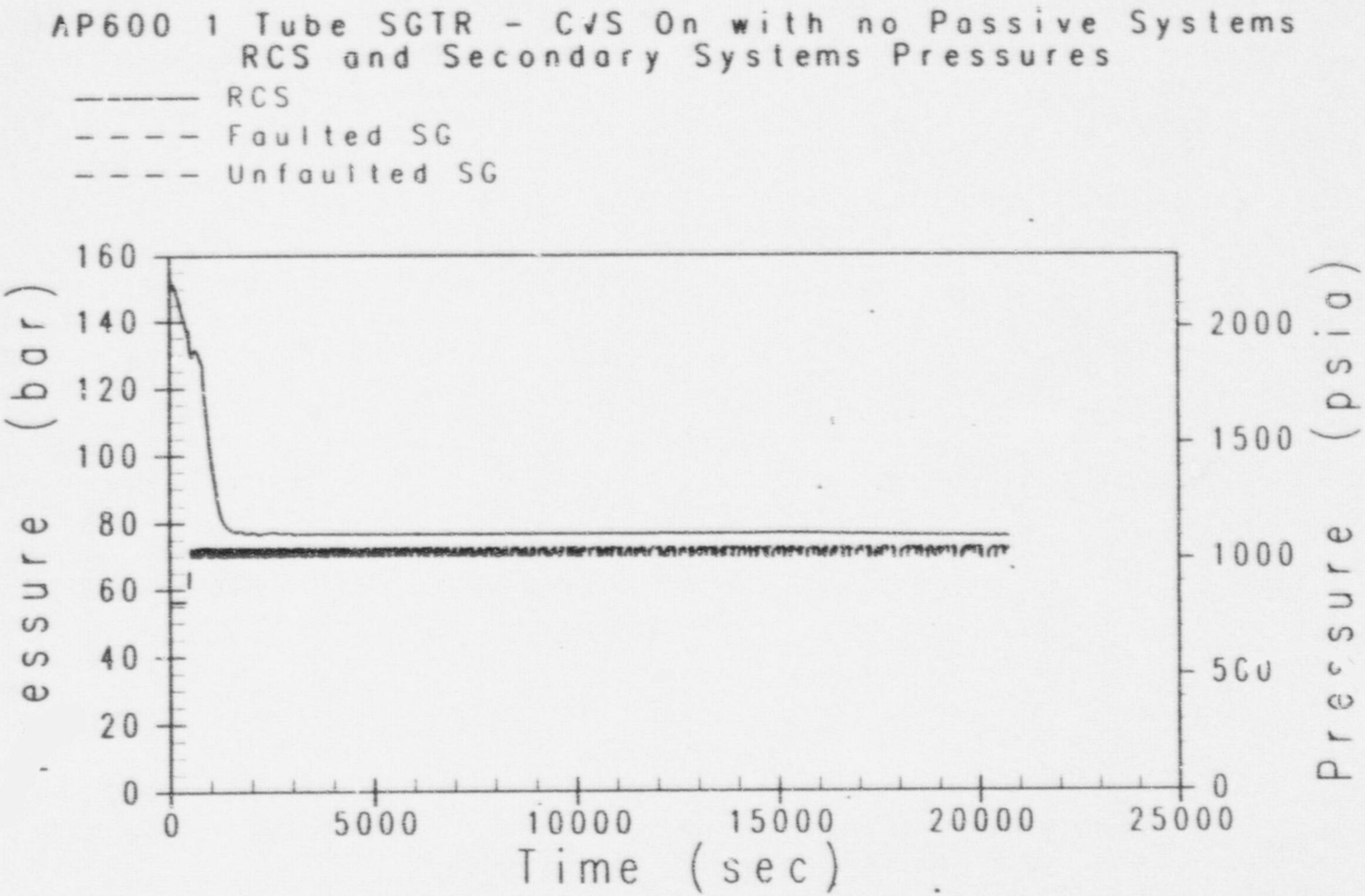

AP600 1 Tube SGTR - CVS On with no Passive Systems
RCS and Secondary Systems Pressures
RCS
Faulted SG
Unfaulted SG
Pressure (bar)
Pressure (psia)
Time (sec)
0
5000
10000
15000
20000
25000
160
140
120
100
80
60
40
20
0
2000
1500
1000
500
0

Figure 4

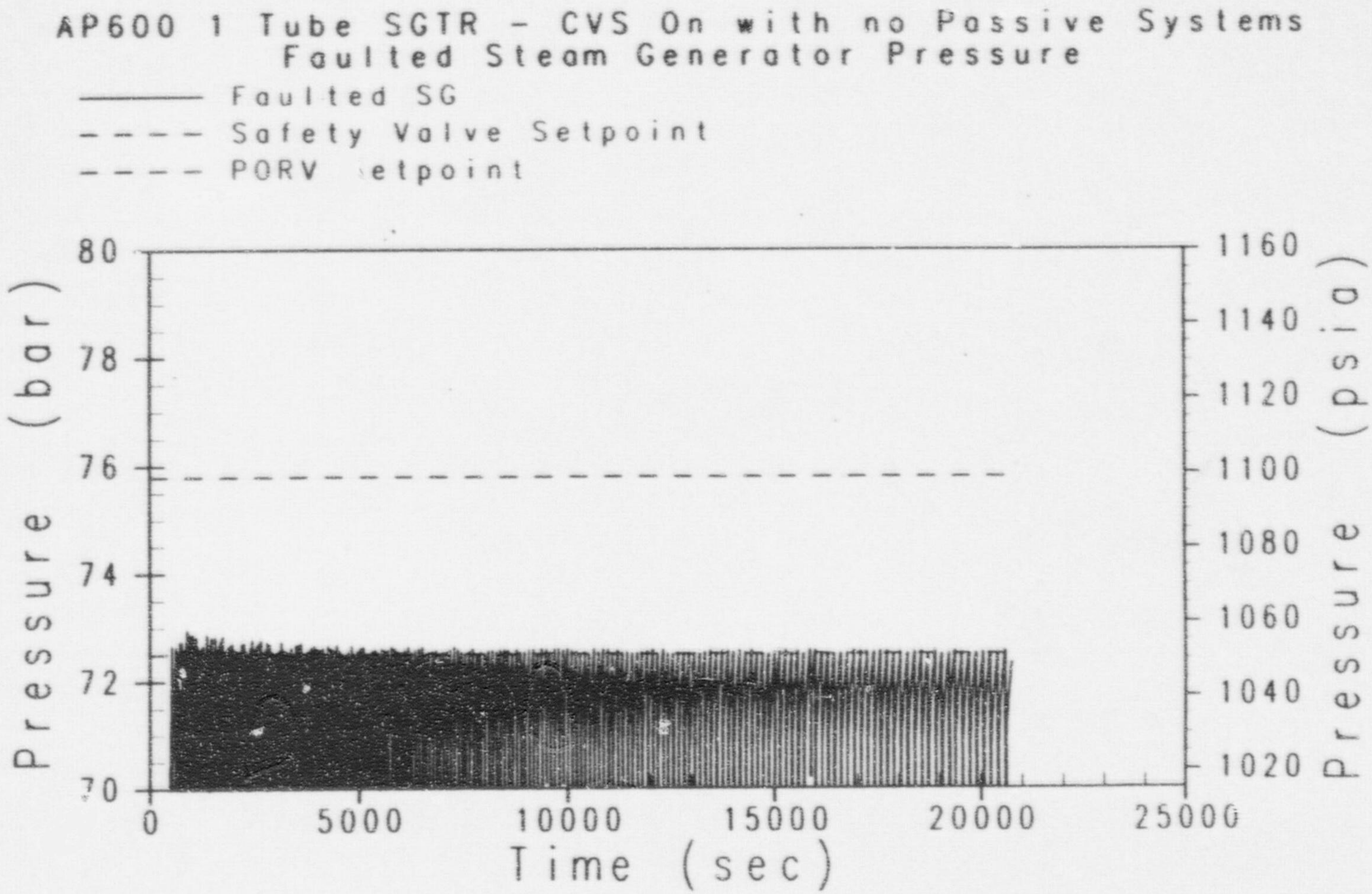

AP600 1 Tube SGTR - CVS On with no Passive Systems
Faulted Steam Generator Pressure
Faulted SG
Safety Valve Setpoint
PORV Setpoint
Pressure (bar)
Pressure (psia)
Time (sec)
80
78
76
74
72
70
1160
1140
1120
1100
1080
1060
1040
1020
0
5000
10000
15000
20000
25000

Figure 5

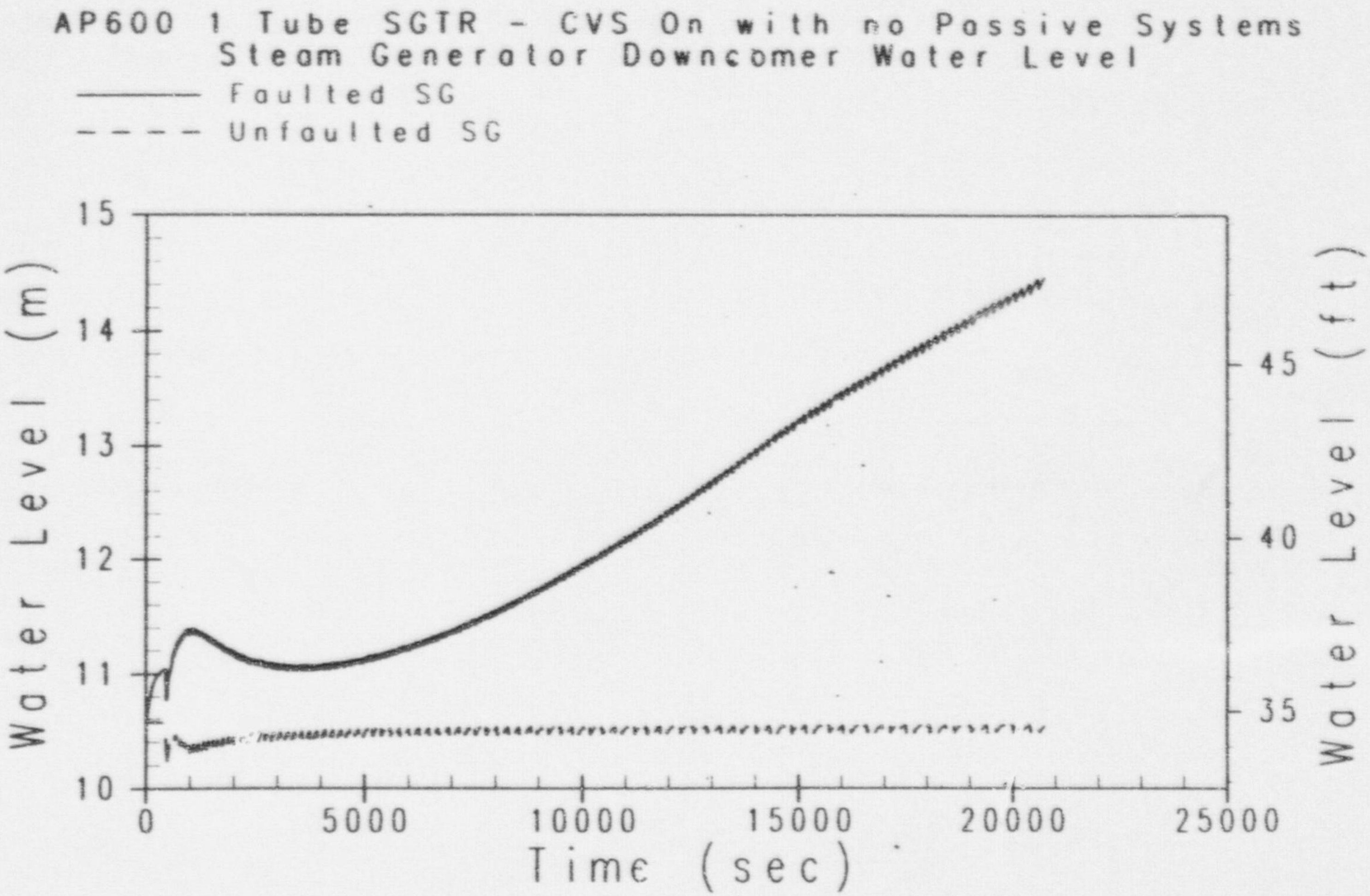

AP600 1 Tube SGTR - CVS On with no Passive Systems
Steam Generator Downcomer Water Level
Faulted SG
Unfaulted SG
Water Level (m)
Water Level (ft)
Time (sec)
10
11
12
13
14
15
35
40
45
0
5000
10000
15000
20000
25000

Figure 6