Post Exam Comments on RO Question #49

RO Question 49

Given the following conditions:

- Unit 2 is operating at 50% power.
- 24MS167, Main Steam Isolation Valve Fast Close PB is depressed on the control console.

Which of the following describes SG pressures 5 minutes after the PB is depressed with NO other operator action?

Main Steamline Pressure in 21-23 loops is...

a. lower because of the increased steam flow from 21-23 SGs ONLY.
b. higher because Hi Steamline Flow Safety Injection signal has occurred.
c. higher because a Steamline Delta Pressure Safety Injection signal has occurred.
d. lower because of the increased steam flow from 21-23 SGs AND the steam flow from 24MS10.

Answer: a

Explanation of Answers: The closure of the 24MS167 will cause its pressure to rise. The 3 other loops will have to make up for the loss of flow from 24 loop, and the increased steam flow from each loop will lower steam header pressure as the MT governor valves open to maintain first stage pressure constant. The steamline D/P SI signal is on one SG LOWER than all the others, not higher. The High Steamline flow SI would not occur because even if the high steamline FLOW were present, it is coincident with LOW steam pressure (600 psig) or Lo Lo Tave (543°F), neither of which will be present.

Facility Comment

Question does not have a correct answer.

Justification: The 24MS167 closure will cause 24 SG NR level to rapidly shrink and the Rx will trip on 1/4 SG with 2/3 NR levels <14%. This response was verified in the Salem simulator and the shrink was rapid and substantial with 24SG NR levels dropped rapidly to <14% resulting in a reactor trip. With no operator action as stated in the stem, full AFW flow was admitted to the SGs which continued to drive SG pressures down below normal post trip values. This results in steam dump demand closing the steam dumps resulting in no steam flow from the 21-23 SGs 5 minutes after 24MS167 closes. Temperature response is consistent with NFS Calculation DS1.6-0515, "ANSI/ANS-3.5-1993 Benchmarking Project for Salem: Transient "A" – Manual reactor Trip" using RETRAN to provide a best estimate plant behavior based on plant design.

Five minutes after the PB for 24MS167 was depressed, all SG pressures are essentially matched, which would not be discernible on the instrumentation in the simulator. If the reactor had not tripped, there would have been a significant difference between the SGs due to the high steaming rate which is what the premise of the question is based on.

Although no specific plant data exist, the simulator response is validated against major plant transients, including full power reactor trips. A comparison against all NSSS system parameters is performed and where deltas exist modifications are made to tune simulator response where appropriate. (cont. next page)

Additional insights can be gained from a review of industry OPEX. The following three plant events have occurred that are on 4 loop Westinghouse plants and all resulted in reactor trips.

<u>Millstone 3 – 12-11-1998 – MSIV closure at power</u>

On December 11, 1998, with Millstone Unit 3 at 100 percent power, main steam isolation valve (MSIV) A went fully closed during partial-stroke testing. The subsequent shrink in steam generator level caused an automatic reactor scram on low-low steam generator level. Two main steam safety valves momentarily opened; otherwise, the scram was uncomplicated.

<u>Zion – 8-18-1996 – MSIV closure at power</u>

On August 18, 1996, Zion Unit 1 automatically scrammed from 100 percent power during testing of the main steam isolation valves (MSIVs). The 1D MSIV closed further than expected, causing the 1D steam generator (S/G) level to shrink to the low-low scram setpoint. The nuclear station operator (NSO) was performing a quarterly operability check of the MSIVs.

<u>Seabrook – 1-25-1994 – MSIV closure at power</u>

On January 25, 1994, with the unit at 100 percent power, the reactor scrammed due to the inadvertent closure of a main steam isolation valve (MSIV) during quarterly surveillance testing. The reactor trip was due to low steam generator level in conjunction with a safety injection and main steam isolation. Quarterly MSIV testing is peformed that only partially closes the MSIVs. In this case, inspite of an operator stationed to monitor and control the valve if necessary, the valve went fully closed resulting in the transient. The cause of the MSIV malfunction was a combination of contaminated hydraulic fluid, a sticking main dump valve and other solenoid valve problems. The station had several past problems with the MSIVs attemting to fully close during surveillance testing and in one case this resulted in another reactor scram. Root cause analysis of previous failures was insufficient and did not determine the fundamental material problems with the MSIVs. Additionally, the preventive maintenance program on the valves was inadequate, and mainteance performed in 1991 resulted in hydraulic fluid contamination and deviation from material requirements

Based on the reactor trip, main turbine stop closure with full AFW flow, steam pressures will equalize and RCS Tavg will lower with steam dump closure, hence, none of the answers are correct.

Based on this information and simulator response the facility recommends deleting RO Q49.

References:

- S2.OP-AR.ZZ-0006, Overhead Annunciators Windows F
- Simulator parameter response data for 24 SG MSIV closure from both 100% power and 50% power initial conditions
- 2004 RETRAN modeling of reactor trip and load rejection transients

NRC Response

The NRC agrees with the proposed change. RO Question #49 has NO correct answer. This question will be deleted from the exam.

RO Question #49 describes the full closure of the main steam isolation valve on 24 Steam Generator from an initial plant power level of 50% and asks the applicant to determine main steam line pressure on the other 3 steam lines 5 minutes later as compared to pre-event conditions and select the reason for the change. The key answer, lower pressure because of increased steam flow from 21 thru 23 SGs, assumed the plant could ride out the transient and remain at power without an automatic trip signal.

RO Q#49 was a new question developed for this exam. This question was missed by 4 of the 8 SRO applicants. Two of the four applicants that missed this question selected Choice C and the other two selected Choice D. None of the applicants asked any questions about this test item during exam administration.

The licensee contends there are no correct answers because the MSIV closure causes a level transient on the affected SG of sufficient magnitude to initiate an automatic steam generator low level reactor trip. The normal plant response to a reactor trip (substantial reduction in reactor power generation and corresponding secondary steam flow and RCS temperature stabilization at no-load conditions) would result in higher steam line pressures in 21-23 loops, but not for any of the reasons given in the available answer choices.

The licensee observed and documented the modeled response to the closure of 24 SG Main Steam Isolation Valve 24MS167, from both 100% and 50% power.

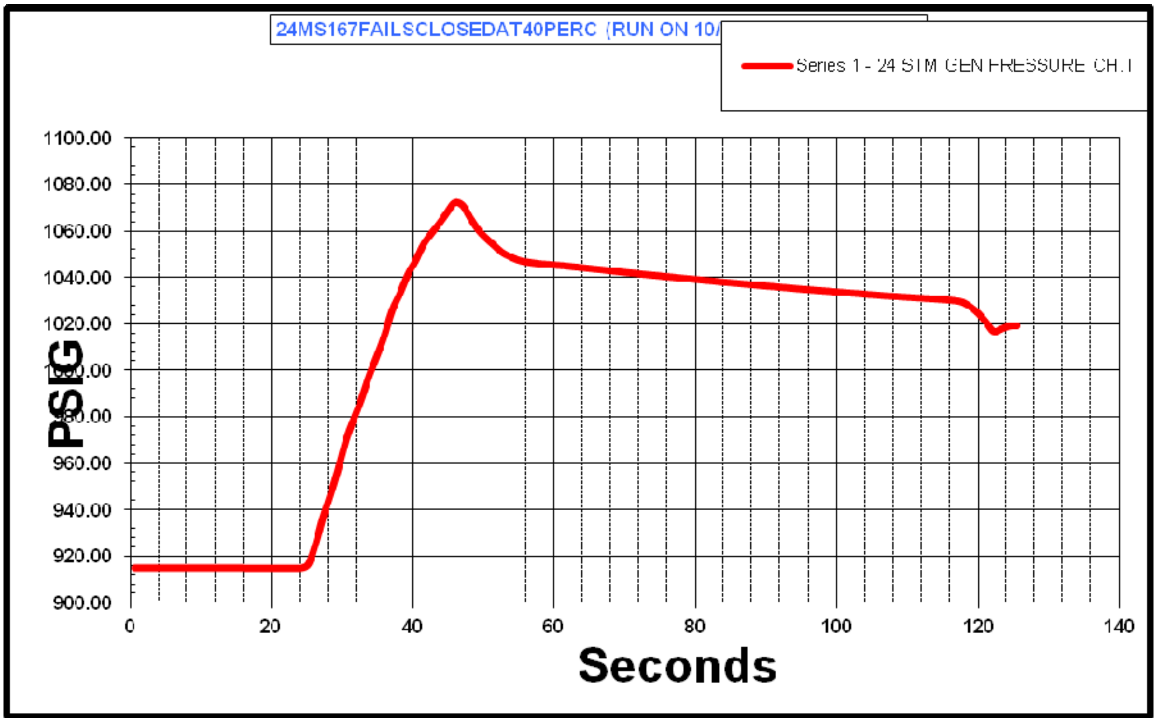
In the 100% power case, the initial RCS pressure transient caused 24 SG narrow range level to rapidly lower to below the automatic low level reactor trip set point due to shrink. The reactor tripped. RCS temperature trended to no-load values and steam flow tapered off to normal post-trip reduced values.

In the 50% power case, the initial RCS pressure transient resulted in 24 SG narrow range level dropping due to shrink. However, the level stopped decreasing at value slightly above the reactor trip setpoint. The reactor remained at power as the 24 Steam Generator Feed Regulating Valve 24BF19 opened to try to restore level. 24 SG pressure rise to a value of 1070 psig resulted in opening of Atmospheric Steam Dump 24MS10 and one of the 24 Main Steam Line safety valves, resulting in continuous steam discharge from 24 SG to atmosphere. The feedwater header pressure, initially at 1015 psig, lowered over a 90 second period to about 920 psig. With 24 SG pressure at 1070 psig and feed header pressure at less than 1015 psig, feedwater header pressure was insufficient to force feedwater into 24 SG, even though 24BF19 was opening. At about 90 seconds into the event, 24 SG narrow range level dropped below the low level trip setpoint, initiating a reactor trip signal. Following the reactor trip, as in the 100% power case, RCS temperature trended to no-load values and steam flow tapered off to normal post-trip reduced values.

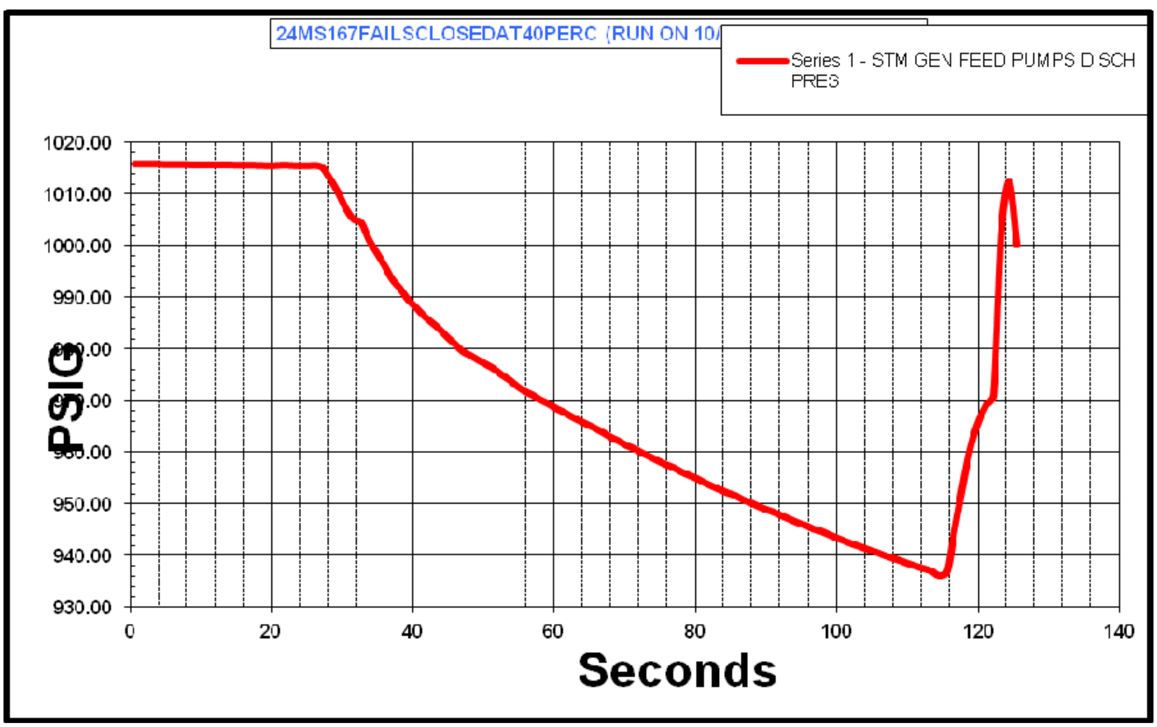
The digital feedwater control system compares feed regulating valve differential pressure to a feed pump speed control ΔP set point and raises or lowers feed pump speed to maintain FRV ΔP in a range between 50 and 151 psid as reactor power varies from 15% to 100%. Selected steam flow signals from each loop are summed and processed to develop this average feed pump speed control ΔP set point. The average loop steam pressure signal is compared to a selected feed water header pressure to produce an actual ΔP signal. Feed header pressure actually lowered during this event simulation, based on control system demand, because of the drop in average steam generator pressure. Level remained nearly constant in SGs 21, 22 and 23, but dropped in 24 SG because of insufficient pressure to feed 24 SG as shown in the following 2 trend graphs.

This control system response is expected per plant design for an inadvertent MSIV closure from any power level where saturation pressure for RCS hot leg temperature is greater than atmospheric valve set point, such that SG pressure stabilizes above the pressure of the feedwater header.

.



24 Steam Generator Pressure



Feed Header Pressure

Post Exam Comments on SRO Question #4

SRO Question 4

Given the following conditions:

- 21 CVCS Monitor Tank is in recirc.
- 21 CVCS Monitor Tank will be released via the Waste Discharge Cross Connection to Unit 1 Liquid Radwaste system IAW S2.OP-SO.WL-0001, Release of Radioactive Liquid Waste from 21 CVCS Monitor Tank.

Which of the following describes an action which would result in an unmonitored Radioactive Liquid Release if it was performed AFTER the Liquid Release was started, and which would require operator action to stop the release?

a. Unit 2 CRS authorizes a tagging request which results in the 2R18, Liquid Waste Disposal Rad monitor losing its power source.
b. Unit 1 CRS authorizes a tagging request which results in the 1R18, Liquid Waste Disposal Rad monitor losing its power source.
c. Unit 2 CRS authorizes a tagging request which results in the 2FR1064, Rad Waste Liquid Monitor Pumps Discharge flowmeter losing its power source.
d. Unit 1 CRS authorizes a tagging request which results in the 1FR1064, Rad Waste Liquid Monitor Pumps Discharge flowmeter losing its power source.

Answer: c

Explanation of Answers: The Salem ODCM is a supporting document to the Unit 1 and Unit 2 Technical Specifications. The previous LCOs that were contained in Radiological Effluent Tech Specs are now included in the ODCM. Using ODCM 3.3.3.8, IAW Salem Tech Specs 6.8.4.1.g.1 and Table 3.3-12 on page 19, the R18 and the FR1064 required to be operable. The R18 is interlocked with the WL51 so that if the R18 loses power, the WL51 will shut, preventing an unmonitored release, and no operator action would be required to stop the flow of water overboard. The Flow recorders (FR1064) are also required for releases. There is no comparable interlock between the flow recorder and the WL51, so the release would be considered unmonitored based on the fact that a required component was not available, and the release is ongoing. Loss of the flow recorder during the release requires operators to close the 2WL51 IAW step 5.5.9. The Unit 1 distracters require knowledge of the flow path for a release which is directed through the cross connect line.

Facility Comment

Question choices a and c are correct.

Justification: Authorizing a tagging request to remove power to the 2R18 would require the Controls Group to remove the 2R18 rad monitor from service first. Partial procedure usage of S2.IC-FT.RM-0028 or S2.IC-CC.RM-0028 would be performed to remove 2R18 from service, and requires the RO to place the 2WL51 Bypass Switch in the control room to the block position per prerequisites 2.4 and 2.7 respectively. Removing the 2R18 from service, in conjunction with placing the bypass switch in the block position (Dwg 203679, G-6 location) would result in an unmonitored release being performed since the compensatory actions required for the 2R18 being out of service have not been performed, and would require operator action to stop the release, since the Bypass Switch in the block position bypasses the High Radiation contact 74/1-R18 which normally opens on a high radiation signal. The switch is powered from the 125VDC associated with the 2WL51 and not the 2R18. Both of these procedures have a prerequisite which allows a Liquid Release to be performed even while the 2R18 is being taken out of service.

Facility recommends accepting both a and c as correct answers.

References:

- S2.OP-SO.WL-0001, Release of Radioactive Liquid Waste from 21 CVCS Monitor Tank
- Drawing 203679 No. 1 and 2 Units Waste Disposal System No. 1WL51 and 2WL51 Disch. Valves
- S2.IC-CC.RM-0028, 2R18 Liquid Waste Disposal Process Radiation Monitor
- S2.IC-FT.RM-0028, 2R18 Liquid Waste Disposal Process Radiation Monitor.

NRC Response

The NRC does NOT agree with the proposed change. SRO Question #4 Key Answer Choice C is correct and there are no other correct answers. This question will remain as-is on the exam.

SRO Question #4 gives a situation where the station is preparing to discharge 21 CVCS Monitor Tank to the Unit 1 radioactive waste system. It asks whether the release would be unmonitored and whether operator action would be required to stop the release if, after it was started, one of four choices of actions is taken. Each action choice states that the control room supervisor subsequently authorized a tagging request, which resulted in loss of power to a given component. The question clearly intends for the applicant to evaluate the consequences related to the radioactive waste discharge which would result from the unintended consequence of loss of power to a particular component. It tests 1) if the applicant knows whether the discharge flow path goes through Unit 2 or Unit 1 listed components, 2) if the loss of power to the given flow path components would result in an unmonitored discharge, requiring operator action to stop the release.

SRO Q#4 was a new question developed for this exam. This question was missed by 3 of the 7 SRO applicants. All three of the applicants that missed this question selected Choice A. None of the applicants asked any questions about this test item during exam administration.

The licensee's post-exam submittal contends that, in authorizing a tagging request to remove power from Liquid Waste Process Radiation Monitor 2R18, the CRS would have ensured specific procedural actions were taken to block the 2R18 release termination function prior to removing power from the radiation monitor. If these were the only actions taken related to the liquid waste discharge, then an unmonitored release would be in progress, requiring operator action to stop. They therefore propose Choice A as a second correct answer.

The NRC does NOT agree. The licensee's comments are based on the assumptions that 1) power is being deliberately removed from the 2R18 radiation monitor channel for unspecified reasons, 2) a maintenance procedure not mentioned in the question is partially implemented to defeat the 2R18 high radiation auto isolation function of the 2WL51 Liquid Waste Release Isolation Valve, and 3) that, although the power removal is intentionally authorized, the CRS would allow the removal of 2R18, creating a situation where an unmonitored discharge is now in progress, necessitating immediate termination of the release. This is not a reasonable string of assumptions for reasons described below.

The question choices are structured to indicate the tagging request is for unspecified work which has the consequence of removing power from the given liquid waste discharge component. No information is provided to indicate the purpose or scope of the tagout or what, if any, plant realignment actions are taken. In the absence of such information, there is no basis for the applicant to make assumptions about how or why the tagout is applied. In fact, the absence of amplifying information simplifies the analysis for the applicant, essentially reducing the question to one of evaluating the direct effect on the liquid waste release of loss of power to a particular component. This interpretation, intended by the facility's question developer, is supported by the original question justification, by the phrases, "which results in", and "losing its power source", and by the lack of detailed information related to the tagouts. The choices are phrased in a manner to communicate they are actions that result in unintended consequences affecting the ongoing liquid waste discharge. The choices are not phrased in a manner to indicate that the tagging request was knowingly, intentionally de-energizing the listed components.

Although the NRC disagrees with the licensee's recommendation for SRO Question #4, it is useful to follow the assumption string proposed by the licensee to its logical conclusion. If an applicant assumed the power removal was intentional then, in the absence of any information to the contrary, it would also be reasonable to assume the

CRS performed all expected actions prior to removing power from 2R18, not just those actions that block the auto valve closure. These expected actions would include evaluation of Offsite Dose Calculation Manual (ODCM) action requirements for a de-powered 2R18 radiation monitor. ODCM Table 3.3-12 Action 26 indicates an effluent release may continue provided that prior to initiating a release, additional sampling and assessment actions are implemented. The ODCM directs suspension of the release otherwise. Following this intentional de-powering in accordance with requirements line of reasoning, since the expanded sampling requirements have not been met, the discharge would be secured prior to tagging 2R18. An unmonitored release would not be in progress and no action would be required to stop an already stopped discharge. However, as stated before, given the wording of the choices and the absence of more detailed information, the NRC thinks it is not reasonable to assume the CRS expected to lose power to the stated component when hanging the tagout, Post Exam Comments on SRO Question #14

SRO Question 14

Given the following conditions:

- Unit 1 is operating at 100% power.
- 1PR5, PZR Safety Valve, fails open.

Which of the following identifies what procedure will be used upon the transition out of 1-EOP-LOCA-1, Loss of Reactor Coolant, and the actions in that procedure which will mitigate the effect of the event.

a. 1-EOP-LOCA-2, Post LOCA Cooldown and Depressurization. Establish a <100°F/hr RCS cooldown to restore subcooling. Depressurize the RCS and stop the depressurization when PZR level reaches 77%.
b. 1-EOP-TRIP-3, Safety Injection Termination. Determine minimum required subcooling, then reduce ECCS pumps to minimum require per Table C to equalize ECCS and break flow.
c. 1-EOP-LOCA-2. Establish a continuous RCS cooldown to restore subcooling. Depressurize the RCS ONLY until PZR level is >25%.
d. 1-EOP-TRIP-3. Sequentially stop ECCS pumps while checking RCS pressure stable or rising to equalize ECCS and break flow.

Answer: c

Explanation of Answers: The PZR Safety failing open will cause a SBLOCA. Pressure will rapidly lower, and a Rx trip will occur, followed shortly by a Low Pressure SI. The Safety Injection will actuate, and break flow will equal ECCS flow at ~ 800 psig. With no subcooling, the transition to TRIP-3 cannot be made, and LOCA-2 will initiate a cooldown/depressurization. The depressurization in ONLY performed until positive control of RCS inventory is established at 25% PZR level, as any continued depressurization may cause a loss of subcooling. If subcooling is lost during the depressurization, it will be restored as the CONTINUOUS cooldown continues. The LOCA-2 distractor contains the correct cooldown rate limitation (which is intentionally absent from the correct choice) but the 77% PZR level is wrong. It is the level at which a depressurization in LOCA-5 (step 24.1) would be stopped. Choice B is wrong because wrong procedure and wrong action. Table C is found in LOCA-2. Choice D is wrong because wrong procedure, with correct action for that procedure.

Facility Comment

Question does not have a correct answer.

Justification: During this PZR vapor space leak, the PZR will be filled solid due to the PORV failing open. After initiating a cooldown to restore subcooling, LOCA-2 contains steps (15-16) to depressurize the RCS to refill the PZR. With the PZR full, these steps will not be performed, since the depressurization termination criteria of PZR level >25% is present before the step is performed. Both of the choices with the correct procedure state to depressurize the RCS.

Additionally, if the operator were to answer Step 11 (is RCS subcooling >0°F) as NO based on not having cooled down enough to establish subcooling when that step is read, then steps 15 and 16 would be bypassed and no depressurization would be performed. Subcooling is <0°F at the onset of the event and requires a significant period of time to be reestablished. Step 11 transitions to step 33 and bypasses the depressurization actions.

Additional insights from the ERG – "Testing the ES 1.2 (LOCA-2) guideline for this case is important for several reasons. First, some of the actions in this guideline are dependent upon maintaining PRZR level on span, above some specified minimum (i.e., above the heaters). If the open PORV is left unblocked, the PRZR fills soon after trip and the minimum level criteria in ES 1.2 are, of course, always satisfied. Second, the break area for the stuck open PORV is twice as large as the break area for the one inch diameter cold leg break case studied previously."

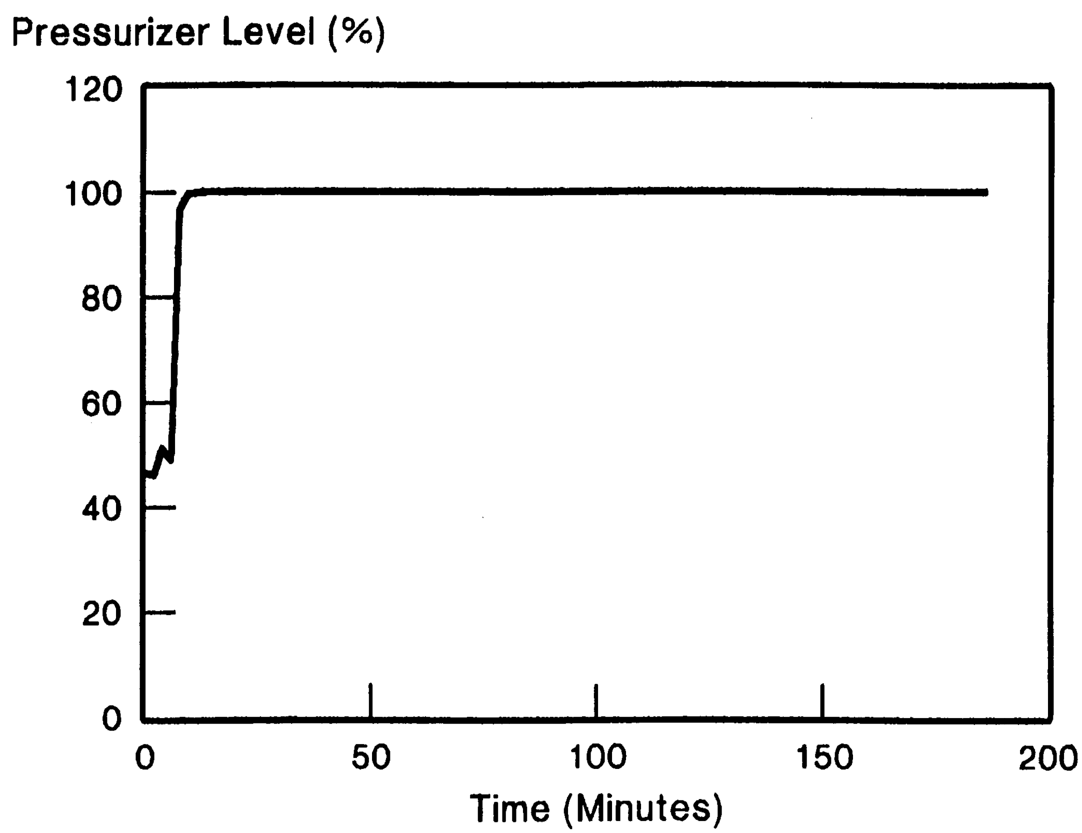
A graph from ERG is on the next page showing a full pressurizer for the entire duration of the event.

Following the graph are the corresponding EOP basis steps and statements as to why the steps would not be performed.

A pressurizer code safety response is similar since a code safety is slightly larger and would have a similar pressure response by rapidly lowering the RCS to saturation and full pressurizer.

Facility recommends deleting SRO Question 14.

Figure 13. PRESSURIZER LEVEL FOR STUCK OPEN PORV WITH ES-1.2 GUIDELINE APPLICATION



EOP Step No: Step 11

ERG Step No: Step 8

EOP Step:

IS RCS SUBCOOLING GREATER THAN 0°F
[SI FLOW EVALUATION]

Purpose:

To determine if the RCS is subcooled so that subsequent actions dependent upon subcooling can be performed.

ERG Basis:

If RCS subcooling can be verified, the LOCA is most likely small and controllable, i.e., ECCS flow equals or exceeds break flow. Subsequent steps that may be allowed include deliberate RCS depressurization, RCP restart, and make-up (ECCS) flow reduction. If subcooling cannot be verified, the transition to EOP Step 33 bypasses these actions.

This step is contained within the main cooldown loop (EOP Steps 5 to 47). Consequently, it is possible that subcooling could be verified later as the cooldown continues.

EOP Basis:

Same as ERG basis.

Supplemental Information:

None

Setpoints and Numerical Values:

Value	Setpoint	Description
0°F	R.01	The sum of temperature and pressure measurement system errors including allowances for normal channel accuracies, translated into temperature using saturation tables - based on Subcooling Margin Monitor.

ERG Deviations:

No deviation from the ERG.

For a failed open Safety valve there would be no possibility of subcooling, thus the Depressurization steps would be bypassed

EOP Step No: Step 14

ERG Step No: Caution 11-1

EOP Step:

CAUTION VOIDING MAY OCCUR IN RCS DURING DEPRESSURIZATION AND CAUSE RAPIDLY RISING PZR LEVEL
[RCS DEPRESSURIZATION TO REFILL PZR]

Purpose:

To alert the operator of possible void formation in the RCS during the RCS depressurization.

ERG Basis:

During a depressurization, the hotter regions of the RCS (upper head) will tend to void and cause PZR level to increase rapidly. This effect would be more likely to occur when RCPs are not running.

EOP Basis:

Same as ERG basis.

Supplemental Information:

None

Setpoints and Numerical Values:

None

ERG Deviations:

No deviation from the ERG.

With a safety valve failed open the depressurization is essentially in effect from the time of the failure and the voiding effect described in the ERG Basis above would have occurred and the pressurizer would be filled completely

References:

- 1-EOP-LOCA-2, Post LOCA Cooldown and Depressurization

NRC Response

The NRC agrees with the proposed change. SRO Question #14 has NO correct answer. This question will be deleted from the exam.

SRO Q#14 was a new question developed for this exam. This question was missed by 5 of the 7 SRO applicants. Three of the five applicants that missed this question selected Choice A and the other two selected Choice D. One of the applicants that missed the question and selected Choice A told the proctor during the exam that, "step in LOCA-2 won't be performed since PZR Safety is stuck open depressurizing the RCS." He was directed to use the information in the question stem to answer the question. No other questions were asked about this test item during exam administration.

The licensee contends there are no correct answers because the open pressurizer safety valve causes a pressurizer steam space LOCA, which results in rapid RCS depressurization to saturation conditions. The depressurization and flow out the top of the pressurizer leads to voiding in the reactor vessel and indication of a full pressurizer. The expected RCS parameter values would result in a path through LOCA-2 which would bypass the step that directs depressurization to fill the pressurizer. None of the choices provided are correct for the expected conditions.

The NRC agrees that this question should be deleted because none of the answer choices are correct. The ERG background document data supports the conclusion that a failed open pressurizer safety will result in an indicated level increase as the RCS voids out the top of the pressurizer. LOCA-2 is the correct procedure after transition out of LOCA-1. However, the steps for depressurize the RCS will not be implemented because of the expected progression of this steam space LOCA event.