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U. S. Nuclear Regulatory Commission
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Clinton Power Station, Unit 1
Facility Operating License No. NPF-62
NRC Docket No. 50-461

Subject: Additional Information Supporting the License Amendment Request to Revise Suppression Pool Water Level and Upper Containment Pool Water Level Requirements in Mode 3

Reference: Letter from K. R. Jury (Exelon Generation Company, LLC) to U.S. NRC, "Request for License Amendment to Appendix A, Technical Specifications to Revise Suppression Pool Water Level and Upper Containment Pool Water Level Requirements in Mode 3," dated November 16, 2001

In the above referenced letter, AmerGen Energy Company (AmerGen), LLC submitted a request for changes to the Facility Operating License No. NPF-62 and Appendix A to the Facility Operating License, Technical Specifications (TS), for Clinton Power Station (CPS) to revise the suppression pool water level and upper containment pool water level requirements for Mode 3. Specifically, the proposed changes in the referenced letter requested the revision of the allowable operating range for the suppression pool water level and the modes of applicability for the upper containment pools. The affected specifications are TS Section 3.6.2.2, "Suppression Pool Water Level," and TS Section 3.6.2.4, "Suppression Pool Makeup (SPMU) System." The NRC, in a conference call, requested additional information regarding the proposed changes in the referenced letter. The attachment to this letter provides the NRC requested information.

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Should you have any questions related to this information, please contact Mr. Timothy A. Byam at (630) 657-2804.

Sincerely,

T. W. Simpkin

for Keith R. Jury
Director – Licensing
Mid-West Regional Operating Group

Attachments:

Affidavit
Attachment 2: Additional Information Supporting the License Amendment Request to Revise Suppression Pool Water Level and Upper Containment Pool Water Level Requirements in Mode 3
Attachment 3: Clinton Power Station EOP-1
Attachment 4: Clinton Power Station EOP-2

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Clinton Power Station
Office of Nuclear Facility Safety – Illinois Department of Nuclear Safety

STATE OF ILLINOIS	**)**	
COUNTY OF DUPAGE	**)**	
IN THE MATTER OF	**)**	
AMERGEN ENERGY COMPANY, LLC	**)**	**Docket Number**
CLINTON POWER STATION, UNIT 1	**)**	**50-461**

SUBJECT: **Additional Information Supporting the License Amendment Request to Revise Suppression Pool Water Level and Upper Containment Pool Water Level Requirements in Mode 3**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

T.W. Simpkin
T. W. Simpkin
Manager – Licensing

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 4th day of

October, 2002.

Notary Public

"OFFICIAL SEAL"
TIMOTHY A. BYAM
COMMISSION EXPIRES 12/04/05
NOTARY PUBLIC STATE OF ILLINOIS

ATTACHMENT 2

Additional Information Supporting the License Amendment Request to Revise Suppression Pool Water Level and Upper Containment Pool Water Level Requirements in Mode 3

Question 1
What is the difference in SPMU hold-up volume between the current license basis assumption and licensee's proposed Mode 3 reduced pressure assumption (i.e., what is the floodable volume of the reactor upper dome and cylinder above the bottom of the main steam nozzles)?

Response 1
The suppression pool makeup (SPMU) volume required in the current licensing basis is 14,748 ft^3. The volume available from the upper pool when the reactor cavity is fully drained is 3694 ft^3. The difference is 11,054 ft^3. This is not the same volume as the floodable volume of the reactor upper dome and cylinder at greater than 1 inch above the bottom of the main steam nozzles, which is approximately 3760 ft^3. The decrease in the SPMU volume is more than the dome volume because of the proposed increase in the suppression pool level, which reduces the make-up requirement. The difference in the volumes is discussed further in the response to Question 6. These values refer to the alternate SPMU requirements of proposed Technical Specification (TS) Surveillance Requirement (SR) 3.6.2.4.1.e, which allows the reactor well pool to be fully drained (see Reference 1).

Question 2
Assuming that a design-basis LOCA occurred with reactor pressure less than 235 psig in Mode 3 with the proposed alternate SPMU requirement SR 3.6.2.4.1.d in effect: Using design-basis analysis methods, what would be the minimum top vent coverage if the operator were unable to control the reactor vessel level and the vessel were inadvertently filled to the upper dome?

Response 2
If the suppression pool level were assumed to be reduced by the volume from the top of the vessel dome to 1 inch above the bottom of the main steam lines, the change in level would be less than 6 inches. This would result in top vent coverage of approximately 18 inches. Note that this does not correspond to the methods used in the design basis containment analysis. The design basis containment analysis does not fill the vessel dome or upper cylinder. Using the methods of the design basis containment analysis, the minimum vent coverage would be greater than 2 feet even with the proposed alternate SPMU requirements of SR 3.6.2.4.1.d in effect.

The design basis SPMU volume is calculated separately from the containment analysis by combining all of the volumes that may be filled in different accident scenarios. In addition, the design basis SPMU volume contains conservatism that is documented in the response to Question 6. With no operator action, no single design basis accident (DBA) scenario would fill the vessel to the top of the dome, fill the bottom of the drywell and require containment spray during the early part of an accident. In the reactor recirculation line break analysis, the bottom of the drywell fills quickly, but maximum Emergency Core Cooling System (ECCS) flow will not completely fill the top portion of the vessel cylinder and the dome during the first few days of the Loss of Coolant Accident (LOCA). In the design basis analysis for the main steam line break, vessel level is controlled to less than Level 8 (i.e., approximately 214 inches above top of active fuel) and the bottom of the drywell is filled slowly. Very small breaks fill the bottom of the drywell slowly because less fluid is lost through the break into the drywell fill volume. Containment spray is credited only in the design basis analyses that consider steam bypass of the drywell. Additional margins included in the SPMU make-up volume are discussed in the response to Question 6. In the postulated scenario of a LOCA in Mode 3 at reduced vessel pressure, there are no anticipated actions of higher priority than control of vessel level. In a Mode 1 event, the higher priority action would be to ensure reactor shutdown. This action would not be required in a Mode 3 event, leaving the control of vessel level as the highest priority. The operator actions required to control level are discussed further in the response to Question 23. In summary, these actions are simple, may be accomplished from the control room, are part of routine operator training, and are not new as a result of this amendment. Therefore, it is expected that operators would be able to control vessel level to less than Level 8.

Question 3

Assuming that a design-basis LOCA occurred with reactor pressure less than 235 psig in Mode 3 with the proposed alternate SPMU requirement SR 3.6.2.4.1.e in effect: Using design-basis analysis methods, what would be the minimum top vent coverage if the operator were unable to control the reactor vessel level and the vessel were inadvertently filled to the upper dome?

Response 3

The upper pool and suppression pool levels in the proposed TS SR 3.6.2.4.1.d and SR 3.6.2.4.1.e were chosen to provide the same post-LOCA suppression pool volume. Therefore, the response to this question is the same as Question 2.

Question 4

Assuming that a LOCA occurred with reactor pressure less than 235 psig in Mode 3 with the proposed alternate SPMU requirement SR 3.6.2.4.1.d in effect: If the minimum top vent coverage would be less than 2 feet, how would adequate pressure suppression be assured for the containment?

Response 4

It was not the intent of Reference 1 to assume complete condensation with less than two feet of vent coverage, but rather to ensure that 2 feet of vent coverage was always provided. The minimum top vent coverage will be greater than 2 feet because the operators will control vessel level to less than Level 8. In addition, as discussed in the response to Question 6, margins are available in the calculation of the SPMU volume such that even if the operators allowed the dome to fill completely, the long term vent coverage would not be expected to be significantly less than 2 feet.

If the manual method of reducing the suppression pool level by the volume of the upper portion of the cylinder and dome is employed, the resulting minimum vent coverage of 18 inches would provide condensation of any steam passing through the vents. In addition, as mentioned in the response to Question 2, the SPMU volume calculation is based on a combination of events that do not occur concurrently. The amount of time required to fill the bottom of the drywell and the vessel dome is dependent on the size of the line break. In the design basis recirculation line break (i.e., large break), the lower part of the drywell fills quickly, but the vessel dome does not. The SAFER/GESTR

ATTACHMENT 2

Additional Information Supporting the License Amendment Request to Revise Suppression Pool Water Level and Upper Containment Pool Water Level Requirements in Mode 3

ECCS analysis shows that reactor pressure vessel (RPV) water level stabilizes well below Level 2 (i.e., approximately 116 inches above top of active fuel). In this case, the drywell pool overflows the weir wall into the suppression pool early in the event and the suppression pool level remains well above 2 feet over the top of the top vents. In very small breaks, the drywell fills up slowly, but the vessel fills up quickly. When the vessel is filled, even if operators do not act to throttle flow, the pump flow will be reduced due to the increased pressure. For very small breaks, this condition can exist for a considerable time before the break flow will deposit sufficient water on the floor of the drywell to lower the suppression pool level to less than 2 feet above the top vents.

When the vessel fills to the level of the break, the water spilling out of the break into the drywell condenses the steam in the drywell. In both the recirculation and main steam line breaks, this condensation reduces the pressure in the drywell to less than in the containment. This can be compared to the drywell pressure response in the DBA recirculation line break accident submitted for the extended power uprate license amendment in Reference 6. Attached Figure 1 shows that the drywell pressure drops below the containment pressure at about 600 seconds into the accident. After 600 seconds the drywell and containment pressure gradually increase. For the duration of the event, the drywell-to-containment differential pressure does not reach 2 psid. In the analysis for the design basis main steam line break, the drywell pressure does not drop below the containment pressure because the vessel is not allowed to flood to the height of the break. The power uprate analyses were performed assuming minimum ECCS flow. With full ECCS flow, 2 feet of vent coverage is reached earlier. However, because of the larger ECCS flow out of the break, drywell pressure is also reduced more quickly.

A series of GOTHIC computer code runs were performed to determine the drywell to containment differential pressure profile, including the associated vent flows, as compared to the suppression pool water level. From these GOTHIC runs, the flow through the drywell vents was determined. These runs use the containment model previously prepared and used in Reference 1, with certain adjustments to provide a more conservative prediction of suppression pool level. GOTHIC runs were made modeling the break flow with both the Moody slip flow model and the homogeneous flow model. The Moody model is generally used in containment analyses because it predicts conservatively high break flow. Runs were made using the homogeneous model to provide a more conservative prediction of the vessel fill time over the range of vessel pressures being considered in Mode 3. These runs included various break sizes, from the double ended guillotine break of the recirculation and main steam lines down to a small break of 0.05 ft^2. Runs were made using both minimum and maximum ECCS flows. The runs show that during any period of time when the vent coverage is reduced to less than 2 feet, there is no steam flow through the drywell vents because the drywell pressure has been reduced to below the containment pressure by condensation. With no operator action to reduce ECCS flows, the suppression pool drawdown is complete in about 10 to 20 minutes, depending on the ECCS flow and other assumptions used in the model. In all cases evaluated, the vent flow stopped several minutes before the suppression pool level dropped below 2 feet. This shows that there is margin in the GOTHIC models to accommodate minor changes in inputs and assumptions without changes in the result. Therefore, when condensation is required, more than two feet of vent coverage will be available.

The vent coverage of 2 feet was chosen in the early part of the General Electric (GE) containment design to balance pressure suppression against hydrodynamic loads. However, as discussed in Section 6 of Reference 2, testing has demonstrated that complete condensation will occur with less than 2 feet of vent coverage. Reference 2 cites the documentation of testing contained in Reference 3. Attachment A to Reference 3 is GEAP-3596, which documents the Humboldt Bay Full Scale Tests. These tests showed that complete condensation occurred even if the vents were not covered. The Humboldt Bay Tests were for vertical vents rather than the horizontal vents that are used in the Mark III containment design. References 3 and 4 provide information on the Mark III containment testing program to confirm adequacy of vent coverage. Complete condensation was demonstrated in small-scale tests with vent coverage as low as 4 inches in some very limited test series and as low as 18 inches in others. These tests were conducted with the vent coverage at the beginning of the event being less than 2 feet so that the initial rapid discharge of steam passed through less than 2 feet of water. In a postulated Mode 3 LOCA, the initial vent coverage would be much greater than 2 feet. Vent coverage would not drop below 2 feet until about 10 minutes or longer into the event even if the operators take no action. This far into the event, if there were any flow through the vents it would be a slow, bubbling flow that would condense more easily. Since at least 18 inches of vent coverage will still be available, complete condensation is still expected if steam is transported through the drywell vents when the suppression pool level is reduced. This vent coverage assumes that the drywell and containment suppression pool water levels are equal.

In summary, the minimum top vent coverage will be greater than 2 feet since the operators are expected to control reactor vessel level to less than Level 8. If the operator does allow the vessel level to exceed Level 8 and flood the dome, GOTHIC models predict that there will be 2 feet of vent coverage while there is flow through the vents and therefore, adequate pressure suppression will be assured. In addition, should the operators flood the dome, vent coverage never decreases to less than 18 inches, which is expected to provide complete condensation and pressure suppression.

<u>Question 5</u>
Assuming that a LOCA occurred with reactor pressure less than 235 psig in Mode 3 with the proposed alternate SPMU requirement SR 3.6.2.4.1.e in effect: If minimum top vent coverage would be less than 2 feet, how would adequate pressure suppression be assured for the containment?

<u>Response 5</u>
The upper pool and suppression pool levels in the proposed TS SR 3.6.2.4.1.d and SR 3.6.2.4.1.e were chosen to provide the same post-LOCA suppression pool volume. Therefore, the response to this question is the same as Question 4.

<u>Question 6</u>
Near the top of page 4, the licensee's submittal states that the volume required to be considered for design basis hold-up "contains margins when considering an accident in Mode 3." Please identify the conservatisms inherent in the DBA hold-up volume assumption when considering a Mode 3 accident and explain how

these conservatisms bound the proposed reduced hold-up volume assumption.

Response 6

The design basis suppression pool make-up volume is based on volumes that may be filled in any accident scenario and is not based on any single design basis event. In the DBA recirculation line break analysis with minimum ECCS flow, the bottom of the drywell will fill with water, but the RPV water level does not reach Level 8 or the main steam lines. In the design basis main steam line break (MSLB) analysis, the vessel fills to Level 8, but only a small amount of water accumulates in the bottom of the drywell and the vessel dome is not filled. Containment spray is not credited in the design basis analyses for either the recirculation line break or the MSLB. Therefore, including all of the hold-up volumes in the design basis analysis provides margin for suppression pool make-up.

The current design basis accident analysis for a MSLB assumes that operators control vessel level between Level 3 (i.e., approximately 171 inches above top of active fuel) and Level 8. This method accommodates the issue raised in Humphrey concerns 4.1 and 4.2. These Humphrey concerns identified the potential for the containment analysis results to be more severe if the operators controlled vessel level to less than Level 8 and the drywell pool did not form. The concern and the Clinton Power Station (CPS) response are documented in Reference 11. Therefore, the current design basis analytical methods of assuming the operators do control vessel level and prevent overflow into the drywell for a MSLB provide conservative containment analysis results.

In Mode 1, the DBA begins with a scram from a vessel pressure of 1040 psia. For the Mode 3 event, the rods are already inserted and the vessel pressure starts at 250 psia. The differences in the transient vessel level response for these two conditions are significant based on many factors, including significant initial density and mass differences for the fluid in the vessel and break flow differences. These differences, while not accounted for in the CPS design basis make-up volume, provide additional margin in the Mode 3 event. In addition, the decay heat load due to an accident in Mode 3 would be less than in Mode 1. Therefore, a smaller heat sink volume would be required to maintain the same peak suppression pool temperature.

The design basis SPMU make-up volume has additional margins included. The design basis SPMU make-up volume conservatively assumes that the vessel water level is at low Level 2 at the beginning of the event. This is more than 5-1/2 feet below the normal water level. Using the more realistic assumption that vessel water level starts at the normal water level provides almost 1500 ft^3 of additional water volume. In addition, the design basis make-up volume uses the GE standard calculation for vessel fill rather than the CPS specific value. The GE standard calculation vessel fill value has been determined to be conservative by about 375 ft^3 for CPS. The sum of these conservatisms provides about 3 inches of margin in the suppression pool level.

In the calculation of make-up volume for the proposed technical specification change, CPS dimensions are used and the starting vessel water level is assumed to be Level 3 rather than Level 2. The design basis requirement is to provide make-up from normal water level. The "normal" water level control band is from Level 3 to Level 8; however, the design basis value is calculated with additional margin. Level 3 is approximately 2 ½ feet lower than normal water level and still provides adequate margin for operator control of level. Using normal water level instead of Level 3 reduces the make up requirement by about 1180 ft^3. The volume of the vessel cylinder at greater than 1 inch above the bottom of the main steam lines and the volume of the dome is equal to approximately 3760 ft^3. The sum of these two values is 4940 ft^3 and represents the difference in vessel related fill volume used for the design basis SPMU and that used for the proposed alternate make-up volume. The total available make-up volume in the proposed Mode 3 reduced pressure case is as follows:

Design basis SPMU volume	14,748 ft^3
Margin in GE standard vessel fill value to CPS vessel fill volume	-375 ft^3
Volume between Level 2 and Level 3	-854 ft^3
Volume of vessel cylinder at greater than 1" above bottom of main steam lines and vessel dome	-3760 ft^3
Additional volume available from the suppression pool due to increased minimum level for the proposed amendment from 18' 11" to 19' 9".	-6358 ft^3
Total required Make-up volume	3401 ft^3

Make-up available from upper pool after reactor well is drained	3694 ft^3

The difference between the required make-up volume and the make-up available from the upper containment pool is from rounding up when converting the required volume of water to inches of suppression pool level. This provides approximately one-half inch of additional margin in the increased minimum suppression pool level. The above table refers to the volume available with the proposed make-up requirements of proposed TS SR 3.6.2.4.1.e in effect. With the proposed TS SR 3.6.2.4.1.d in effect, the additional volume available from the suppression pool is changed to 0 ft^3 and the make-up available from the upper pool is 10,052 ft^3. This is the same as the volume of make-up available with the proposed TS SR 3.6.2.4.1.e in effect.

The design basis calculation for required SPMU volume does not credit any feedwater or condensate addition, although some feedwater will flow into the vessel after the LOCA. Although the High Pressure Core Spray (HPCS) System is normally lined up with its suction from the Reactor Core Isolation Cooling (RCIC) System storage tank, no credit is taken for any volume transferred from the RCIC storage tank to the vessel. The calculations also do not consider the changes in water density that would increase the volume of water from the suppression pool as the pool heats up. Each of these items, if considered, would increase the long term suppression pool level and vent coverage.

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Additional Information Supporting the License Amendment Request to Revise Suppression Pool Water Level and Upper Containment Pool Water Level Requirements in Mode 3

Question 7

Please provide justification that it is credible for an operator to control reactor vessel water level in accordance with the Emergency Operating Procedures (EOPs), considering, for instance, the amount of time available, the long term nature of the action, and effects of post-accident conditions such as high stress and operator occupation with task of higher safety significance.

Response 7

According to the Emergency Operating Procedures (EOPs) and associated training, there are no tasks of higher safety significance than restoration and control of vessel level during a design basis LOCA in Mode 3 with the reactor at reduced pressure. The CPS procedures and training are prepared in accordance with Revision 2 of the BWR Owner's Group Emergency Procedure Guidelines (EPG)/Severe Accident Guidelines (SAG). Since the reactor is already shutdown, the higher priority actions associated with failure to scram will not be required. In addition, since the reactor begins at reduced pressure and a LOCA is assumed to have occurred, vessel overpressure would not be a primary concern in this event. With the reactor shutdown and vessel pressure under control, the highest priority actions relate to control of vessel level. The control band allowed by the EOPs is between Level 3 and Level 8. This is approximately 43 inches of vessel level, which provides a wide band for operator control.

As stated in the original submittal, the proposed make-up volume accommodates the operators not controlling level until after the vessel level reaches the steam lines and fills both the steam lines and the bottom of the drywell. The steamlines are approximately 44 inches above Level 8. The proposed make-up volume therefore allows just slightly more than 8 feet of vessel level as the control band.

Stopping and starting the ECCS pumps provides initial level control. HPCS and RCIC, if running, will stop injecting automatically at Level 8. The low pressure pumps are stopped and started from the control room. In the longer term, the EOPs provide guidance for throttling the injection valves to maintain level. After the injection valves are throttled, flow will be relatively constant for the duration of the event and will not require constant operator attention. If the vessel should start to overfill, an alarm will actuate at Level 8, approximately 44 inches (i.e., 3.7 feet) below the bottom of the main steam lines.

In summary, it is highly likely that the operators will be able to control vessel level between Level 3 and Level 8. The action is simple, it can initially be accomplished from the control room, there are no higher priority actions, the operators are extensively trained in the actions, and means are available for long term control that avoid a repetitive action.

Question 8

Please provide a copy of the EOP referenced in the submittal which directs the operator to maintain water level below Level 8.

Response 8

CPS EOP-1, "RPV Control," is included in Attachment 3 to this letter. This EOP provides the actions required for control of vessel level and shows that the operators are directed to control level between Level 3 and Level 8. EOP-2, "RPV Flooding," Attachment 4 to this letter, provides directions for flooding the vessel under the circumstances discussed in the response to Question 9.

Question 9

Please identify any circumstances under which the EOPs would not direct that vessel level be maintained lower than Level 8 and explain the implication of these situations in regard to the proposed alternate SPMU requirements and the analytical assumptions used to justify them.

Response 9

The operators are directed by EOP-1 and EOP-2 to flood the vessel in the event the reactor water level cannot be determined. A potential cause of the inability to determine vessel water level is unreliable reactor water level indication. Vessel level indication is considered unreliable if excessive drywell or containment temperature causes flashing in the reference legs. Graphs of the temperatures above which level indication is unreliable are provided in EOP-1. The design basis accident analyses for a Mode 1 LOCA predicts a maximum containment temperature of less than 160°F. At 160°F, the fuel zone and wide range level indications are reliable at any vessel pressure and at water levels down to approximately the top of active fuel. Similarly, the drywell temperatures and vessel pressures predicted for a Mode 1 LOCA do not present the risk of flashing for the shutdown, upset and narrow range level indication when vessel level is near Level 8. The GOTHIC analyses for an accident in Mode 3 predict peak temperatures under 150°F for the containment and under 250°F for the drywell. The analyses show that there is considerable margin to the containment and drywell temperatures that may cause unreliable vessel level readings. Therefore, the only event in which the operators would be required to flood the vessel dome would be a multiple failure event where all vessel level indication is lost. In this event, the operators would be directed to fill the vessel by all available means. For large break LOCAs, ECCS flow alone would be insufficient to fill the vessel to the top of the dome in a timely manner because hot liquid would initially flow out of the break faster than the ECCS injection rate. If additional systems such as feedwater or condensate were available, the vessel could be filled to the top of the dome, but there would be no water inventory concerns based on the availability of the extra feedwater or condensate.

Question 10

The staff reviewer notes (see page 11-1 of Reference 6 to the licensee's submittal) that the EOPs direct the operator to maintain reactor vessel water level below Level 8, not just for a Mode 3 LOCA, but also for a Mode 1 LOCA (e.g., a DBA). However, the NRC staff has nevertheless required the DBA analysis to consider filling the reactor vessel to the upper dome in calculating the required SPMU hold-up volume. Is it the licensee's intention to distinguish between the credibilities of these two apparently similar operator actions?

Response 10
As discussed in the response to Question 2, the assumptions used in calculating the design basis SPMU volume differ from the actions the EOPs direct the operators to take to cope with an event in either Mode 1 or Mode 3. These differences provide an inherent margin in the Mode 3 event as well as the Mode 1 event. These margins are discussed in the response to Question 6. The proposed amendment uses some of these margins, but does not change the assumptions in calculating the design basis SPMU volume. The ability of the operator to accomplish level control early in a Mode 3 event contributes to the acceptability of reducing the margins. The lack of concern for items such as reactor shutdown, as discussed in the response to Question 7, contributes to the ability of the operators to accomplish control of vessel level early in the event.

Question 11
If the licensee intends to distinguish between the credibilities of the two apparently similar operator actions of the operator controlling vessel water level following a Mode 3 LOCA and the operator controlling vessel water level following a DBA, please provide a basis for the requested distinction.

Response 11
The primary basis for the need for less margin in Mode 3 as compared to Mode 1 is the inherent margin in the suppression pool make-up volume. As discussed in the response to Question 7, since a Mode 3 LOCA is a less complicated scenario than a Mode 1 LOCA and the control of vessel level would be a higher priority in Mode 3 than in Mode 1, there is less potential for interference with the operators accomplishing the goal of maintaining vessel level between Level 3 and Level 8. Therefore, there is a lower potential for needing the extra margin provided in the design basis SPMU volume.

Question 12
What is the minimum available dump volume with the proposed alternate SPMU requirement in effect that the reactor cavity pool level at an elevation greater than or equal to 824'7"?

Response 12
The minimum available dump volume with the reactor cavity pool level at an elevation greater than or equal to 824'-7" is equal to the volume of the separator pool between the dump line and elevation 824'-7" plus the volume in the reactor cavity pool above elevation 821'-3". Please refer to Figure 2 of Reference 1. The sum of these volumes is 10,052 cubic feet.

Question 13
Is the volume difference between the current upper containment pool required dump volume (14,652 cubic feet) and the minimum available dump volume with the proposed alternate requirement of the reactor cavity pool at an elevation of 824'7" equal to the difference in assumed entrapment volume between the DBA analysis and the revised Mode 3, reduced pressure analysis (i.e., equal to the floodable volume of the reactor vessel upper dome and cylinder above the main steam nozzles)?

Response 13

The difference between the current upper containment pool dump volume and the minimum available dump volume with the proposed alternate requirement of the reactor cavity pool at an elevation of 824'-7" is 4696 cubic feet. The current upper pool dump volume was changed to 14,748 cubic feet due to recalculation of the drywell fill volume. This change is included in the TS Bases changes made in support of the extended power uprate amendment. The change in value is unrelated to this proposed amendment. The volume of the vessel dome and cylinder above the main steam nozzles is equal to 3760 ft^3.

Question 14

If the two volume differences in the preceding question are unequal, please explain the derivation of the proposed alternate requirement of maintaining the reactor cavity pool at a level corresponding to elevation 824'7".

Response 14

The reason for the difference in the two volumes is detailed in the response to Question 6. These differences are based on the reanalysis using the CPS specific data for the vessel volume and considering the vessel level starting at Level 3 rather than Level 2. The sum of the volume of the dome and cylinder at greater than one inch above the bottom of the main steam nozzles, the volume between Level 2 and Level 3, and the volume decrease due to using CPS specific dimensions is 4989 ft^3. This is more than the volume of 4696 ft^3 by which the SPMU volume is reduced because the required suppression pool level was rounded up to the nearest inch (i.e., the analysis includes approximately ½ inch of additional margin due to rounding up the minimum required suppression pool level for the Mode 3 event).

Question 15

Is the SPMU hold-up volume accounted for by the proposed alternate SPMU requirement of maintaining the reactor cavity pool minimum level at or above an elevation of 824'7" equal to 38,995 cubic feet?

Response 15

Yes, the hold-up volume accounted for in the proposed SPMU requirement of maintaining the reactor cavity pool minimum level at or above an elevation of 824'-7" is 38,995 cubic feet, the same as that used for the case with the reactor cavity fully drained.

Question 16

If the SPMU hold-up volume accounted for by the proposed alternate SPMU requirement of maintaining the reactor cavity pool minimum level at or above an elevation of 824'7" is not equal to 38,995 cubic feet, please provide justification.

Response 16

As stated in the response to Question 15, the hold-up volume is equal to 38,995 cubic feet, and therefore, no response is required.

Question 17
What is the reactor cooldown rate used in the analysis provided in the licensee's submittal regarding a small break LOCA with steam bypassing the suppression pool?

Response 17
The analysis used 100°F/hour, which is the same cooldown rate used in the licensing basis scenario documented in the CPS Updated Safety Analysis Report (USAR). The vessel pressure/temperature curves provided in TS 3.4.11, "Reactor Coolant System Pressure/Temperature Limits," are also based on a maximum cooldown rate of 100°F/hr.

Question 18
What is the reactor cooldown rate recommended in the EOP applicable to a small break LOCA?

Response 18
The maximum recommended cooldown rate recommended in EOP-1 is 100°F/hour, except in extreme circumstances. For extreme circumstances, such as high containment pressure, the EOPs allow the operators to exceed 100°F/hour. In the Mode 3 drywell bypass analysis, the containment pressure would exceed the allowable value for emergency depressurization in less than 10 minutes. The maximum cooldown rate of 100°F/hour is conservatively used in the analysis to model a normal shutdown in accordance with Appendix A to Standard Review Plan section 6.2.1.1.C.

Question 19
If the analytical reactor cooldown rate exceeds the cooldown rate recommended in the applicable EOP, then please explain how this difference has been taken into account.

Response 19
The cooldown rate used in both the licensing basis and alternate suppression pool make-up analyses is the same as the maximum recommended cooldown rate in the EOPs for all but extreme circumstances. For conservatism, the analyses do not use the higher allowed cooldown rates.

The following questions relate to the time period during Mode 3 that the reactor cavity portion of the upper containment pool is drained.

Question 20
Contrast the amount of response time available prior to manual action for the operator in the control room between situation 1 and situation 2, where situation 1 is a LOCA in Mode 3 with water in the UCP and situation 2 is a LOCA in Mode 3 without water in the UCP. Discuss the consequences of a failure on the part of the operator to take action under the two types of situations.

Response 20

For a LOCA in Mode 3, both with and without water in the upper containment pool, the first action dictated by the EOPs would be to confirm automatic actions, such as start of ECCS systems, have occurred. After confirming these actions and completing them if they have not occurred automatically, both scenarios require the operators to control vessel level between Level 3 and Level 8. The response time expected is the same for both events. The next manual action required in either scenario is the start of suppression pool cooling within 30 minutes.

There are other manual actions that would be directed by the EOPs to minimize the consequences of the accident. The actions would vary depending on availability of equipment, especially non-safety equipment. However, these actions would not be different for the two specified scenarios. In addition, these actions would not interfere with or impact the time available for the control of vessel level between Level 3 and Level 8 in the Mode 3 scenario.

For both situations, the impact of the operator failing to control vessel level to less than Level 8 would be to flood the steam lines assuming the LOCA is a steam line break or a small recirculation line break. In the event of a large recirculation line break, the minimum ECCS system flow cannot flood the vessel up to Level 8, so no operator action is required to prevent vessel level from rising above Level 8. Flooding the steam lines following a MSLB would result in formation of the drywell pool. In both cases, the volume of water required to flood the steam lines and form the drywell pool is accommodated in the make-up volume available. If the break is small enough and operators do not reduce ECCS flow, the vessel dome would be filled after the steam lines are flooded. In situation 1, the suppression pool level would drop to slightly over 2 feet above the top of the top row of drywell vents. In situation 2, the level in the suppression pool would also remain at more than 2 feet above the top of the top row of vents until after the flow through the drywell vents has stopped. In the long term, the suppression pool level may drop slightly below 2 feet above the top of the top vents. However, since there is no flow through the vents during this time, there is no impact to containment pressure. Therefore, even if the operators did not control level to less than Level 8 there would be little impact to the event scenario.

Question 21

Describe any new operator actions required as a result of a LOCA in Mode 3 with the upper containment pool (UCP) drained.

Response 21

There are no new operator actions required as a result of draining the upper pool in Mode 3.

Question 22

Describe any changes to any current operator actions covered by emergency (EOPs) or abnormal procedures that may occur as a result of this amendment. Describe any evaluation of the EOPs for potential modifications.

Response 22
There are no changes proposed to the EOP's or abnormal procedures as a result of this amendment. The responses to Questions 7, 8, 9, 10, 11, 18, 19 and 20 above discuss evaluations performed of the EOPs to determine impacts from this proposed amendment. The result of these evaluations is that there is no need to change any of the actions in the EOPs as a result of the proposed amendment. The current EOP actions are consistent with those required for the proposed amendment and actions that the operator would be expected to take to cope with a LOCA in Mode 3 regardless of whether or not the reactor cavity pool was drained.

Question 23
With regard to the maximum time available prior to mandatory manual action by the operator to control water level in the reactor for the two situations discussed above, describe information required by the control room operator to determine whether such operator action is required. What is the ability to recover from credible errors in performance of manual actions, and the expected time required to make such a recovery?

Response 23
The information needed by the operator to control RPV water level between Level 3 and Level 8 is the vessel level indication and the status of the ECCS systems being used to inject. This vessel level information is available from multiple level indicators in the control room that are designed to be available in post-accident situations. The status of the ECCS systems being used to inject are also available in the control room and are designed to be available in post-accident situations.

As described above, the potential for the RPV water level to increase above Level 8 prior to the operators controlling injection has been addressed in the available suppression pool make-up volume with the reactor cavity drained (i.e., the proposed alternate requirements of SR 3.6.2.4.d and SR 3.6.2.4.e). Controlling RPV water level in the band between Level 3 and Level 8 is an activity that the operators are routinely trained to accomplish, procedures provide adequate guidance, and the action is not new for this amendment. The EOPs direct the operators to control vessel level to less than Level 8 by cycling pumps if throttling is not successful or is not a capability of the available pumps. The ECCS pumps may be stopped and started from the main control room. HPCS and RCIC automatically stop injecting at Level 8 without operator action. Therefore, control of ECCS flow to maintain vessel level may be accomplished immediately. In the longer term, if needed, the HPCS, Low Pressure Core Spray (LPCS), and Low Pressure Coolant Injection (LPCI) injection valves may also be throttled. Throttling these injection valves requires cutting jumpers in the breaker cubicles for the valves to defeat the full open valve signal. Once it has been decided to throttle an injection valve and an operator has been assigned the task, the completion time is less than 20 minutes. This completion time is based on job performance monitoring that is conducted as part of the operator training program.

Question 24
Describe the difference in the level of difficulty with respect to any manual actions between the two situations discussed above? What is the band of water level in

the reactor that the operator must control within under this amendment?

Response 24

As described above, the actions for both situations are the same. The control of RPV water level to prevent overfill is a simple action with multiple success paths, some of which may be accomplished from the control room. In both situations the analyses assume the manual action of controlling vessel level is accomplished. The band of water accommodated by this amendment is broader than that allowed by the EOPs. The EOPs require control of level between Level 3 and Level 8, while this amendment ensures there is sufficient makeup volume to allow level to increase up to 1" above the bottom of the steam lines. The elevation difference between Level 3 and 1" above the bottom of the steam lines is just over 8 feet as compared to 4-1/3 feet between Level 3 and Level 8.

Question 25

Describe any changes the proposed amendment will have on the operator training program and provide the implementation schedule for making the changes.

Response 25

Prior to implementation of the proposed amendment, operators will be trained in the process of draining the upper pool during Mode 3, while raising the suppression pool level. No changes are anticipated to the training program for coping with accidents and events.

Question 26

What analyses have you performed with regard to the interaction of the amendment on the upper containment pool and the amendment on the inclined fuel transfer if an event were to occur at the time both amendments were in effect?

Response 26

No numerical analyses have been performed since there is not expected to be any interaction between the two amendments. The opening of the Inclined Fuel Transfer System (IFTS) blind flange requires installation of the gate between the reactor cavity and the dryer/fuel storage pool (References 9 and 10). The IFTS pool connects to the dryer/fuel storage pool. Installation of this gate ensures there is a sufficient volume of water available for SPMU by preventing leakage through the IFTS tube to the spent fuel pool. This requirement has been incorporated into the plant procedures for removing the IFTS blind flange during Mode 1, 2, or 3 (i.e., CPS procedures 3702.01, "Inclined Fuel Transfer System" and 8109.01, "Inclined Fuel Transfer System Blind Flange Rotation") and the proposed IFTS amendment adds this requirement to the Technical Specifications. In addition to supporting the opening of the IFTS blind flange in Modes 1, 2, or 3, the steam dryer pool to reactor cavity pool gate must also be installed and inflated prior to allowing draining of the reactor cavity pool. With the gate installed the two pools are separate and do not communicate. The suppression pool make-up volume is calculated assuming that this gate is installed and no water is available from the dryer/fuel storage or fuel transfer pools. Refer to Figure 2 of Reference 1 for upper pool configuration.

ATTACHMENT 2

Additional Information Supporting the License Amendment Request to Revise Suppression Pool Water Level and Upper Containment Pool Water Level Requirements in Mode 3

The analyses that were performed for the IFTS amendment are not affected by the draining of the upper pool. None of the contingency actions proposed in support of the IFTS amendment request will be impacted by the draining of the upper containment pool in Mode 3. The containment pressure that results from a LOCA in Mode 3 with vessel pressure less than or equal to 250 psia and the reactor cavity pool drained is less than the containment pressure that results from a design basis LOCA at full power. The evaluations that were performed for flooding, severe accidents, and other concerns are also not impacted by the draining of the reactor well pool on Mode 3.

References:

1. Letter from K. R. Jury (Exelon Generation Company, LLC) to U.S. NRC, "Request for Amendment to Appendix A, Technical Specifications to Revise Suppression Pool Water Level and Upper Containment Pool Water Level Requirements in Mode 3," dated November 16, 2001
2. NEDO-10320, April 1971, "The General Electric Pressure Suppression Containment Analytical Model"
3. NEDE-10182, May 1970, "Additional Information Pressure Suppression Concept Test Data Report"
4. NEDM-10848, April 1973, "Mark III Confirmatory Test Program Progress Report"
5. NEDO-10571, April 1972, "General Electric Boiling Water Reactor Mark III Containment Concept"
6. Letter from K. R. Jury (Exelon Generation Company, LLC) to U.S. NRC, "Additional Plant Systems Information Supporting the License Amendment Request to Permit Uprate Power Operation at Clinton Power Station," dated November 20, 2001
7. Letter from J. M Heffley (AmerGen Energy Company, LLC) to U.S. NRC, "Request for Amendment to Technical Specifications to Permit Inclined Fuel Transfer System Blind Flange Removal During Power Operations," dated April 2, 2001
8. Letter from K. R. Jury (Exelon Generation Company, LLC) to U.S. NRC, "Response to Request for Additional Information Regarding Risk Aspects of Inclined Fuel Transfer System Blind Flange Removal During Power Operations Amendment Request for Clinton Power Station," dated January 15, 2002
9. Letter from J. M. Heffley (AmerGen Energy Company, LLC) to U. S. NRC, "Request for Amendment to Technical Specifications to Permit Inclined Fuel Transfer System Blind Flange Removal During Power Operations," dated April 2, 2001
10. Letter from T. W. Simpkin (Exelon Generation Company, LLC) to U. S. NRC, "Additional Information Supporting the License Amendment Request to Permit Inclined Fuel Transfer System Blind Flange Removal During Power Operations," dated August 23, 2002
11. Letter from G. E. Wuller (Illinois Power Company) to U.S. NRC, "Humphrey Concerns," dated February 4, 1983

ATTACHMENT 2

Additional Information Supporting the License Amendment Request to Revise Suppression Pool Water Level and Upper Containment Pool Water Level Requirements in Mode 3

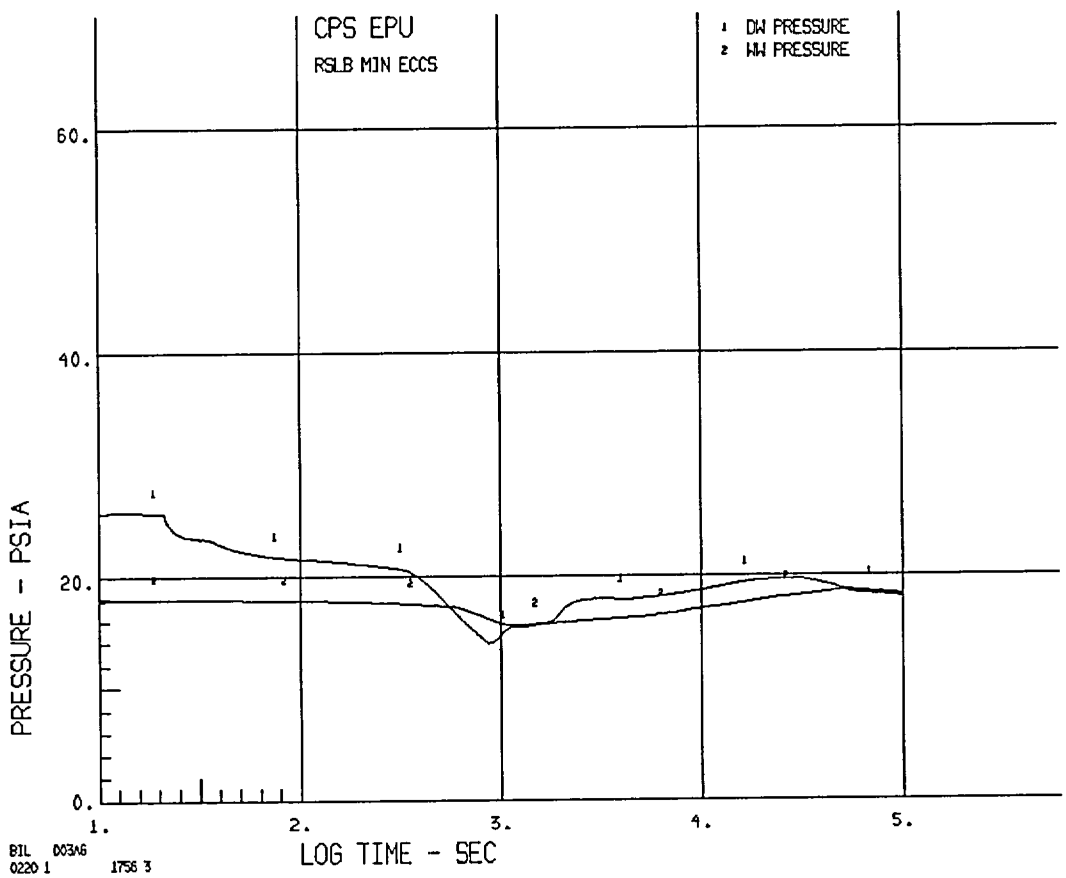
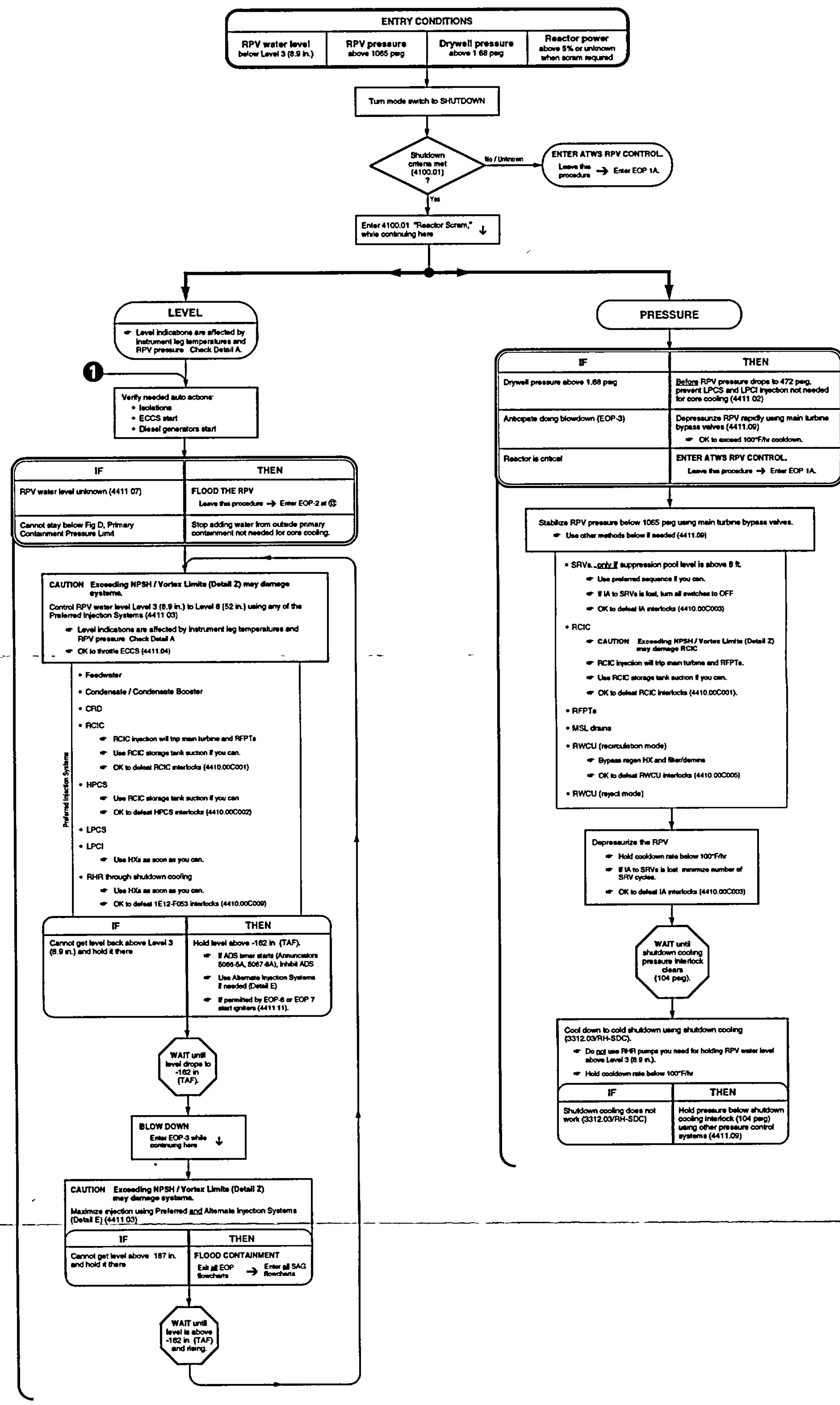


Figure 1
Containment Pressure Response
for Long-term Recirculation Suction Line Break (RSLB)
(originally submitted as Figure 5.7-7 in Reference 6)

ATTACHMENT 3

Clinton Power Station EOP-1
RPV Control

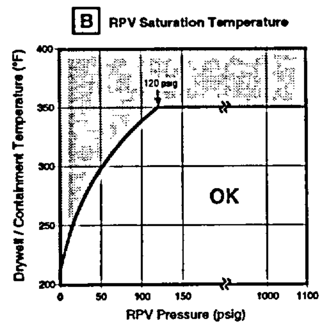
EOP-1 RPV CONTROL



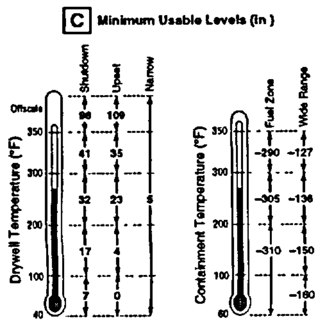
A RPV Water Level Instruments (4411 07)

- RPV water level instruments may be unreliable due to boiling in the instrument legs if drywell or containment temperature is above Fig B, RPV Saturation Temperature
- Do not use an RPV water level instrument if level is at or below Fig C, Minimum Usable Levels.
- Use the Fuel Zone instrument only if the Wide Range instrument is unavailable

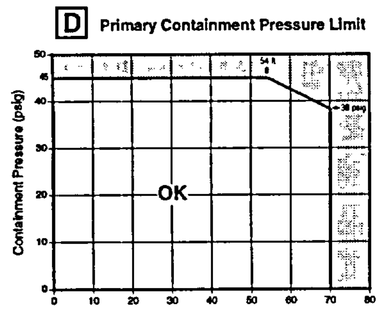
B RPV Saturation Temperature



C Minimum Usable Levels (in)



D Primary Containment Pressure Limit



E Alternate Injection Systems

NOTE Instructions in 4411 03.

- SLC (test or storage tank)
- FP
- SX through RHR B
- FC
- ECCS water leg pumps
- CY

Z NPSH / Vortex Limits

NOTE Limits apply only when suction is from suppression pool

HPCS, LPCS, RHR limits

- Minimum suppression pool level....................11 ft

RCIC limits:

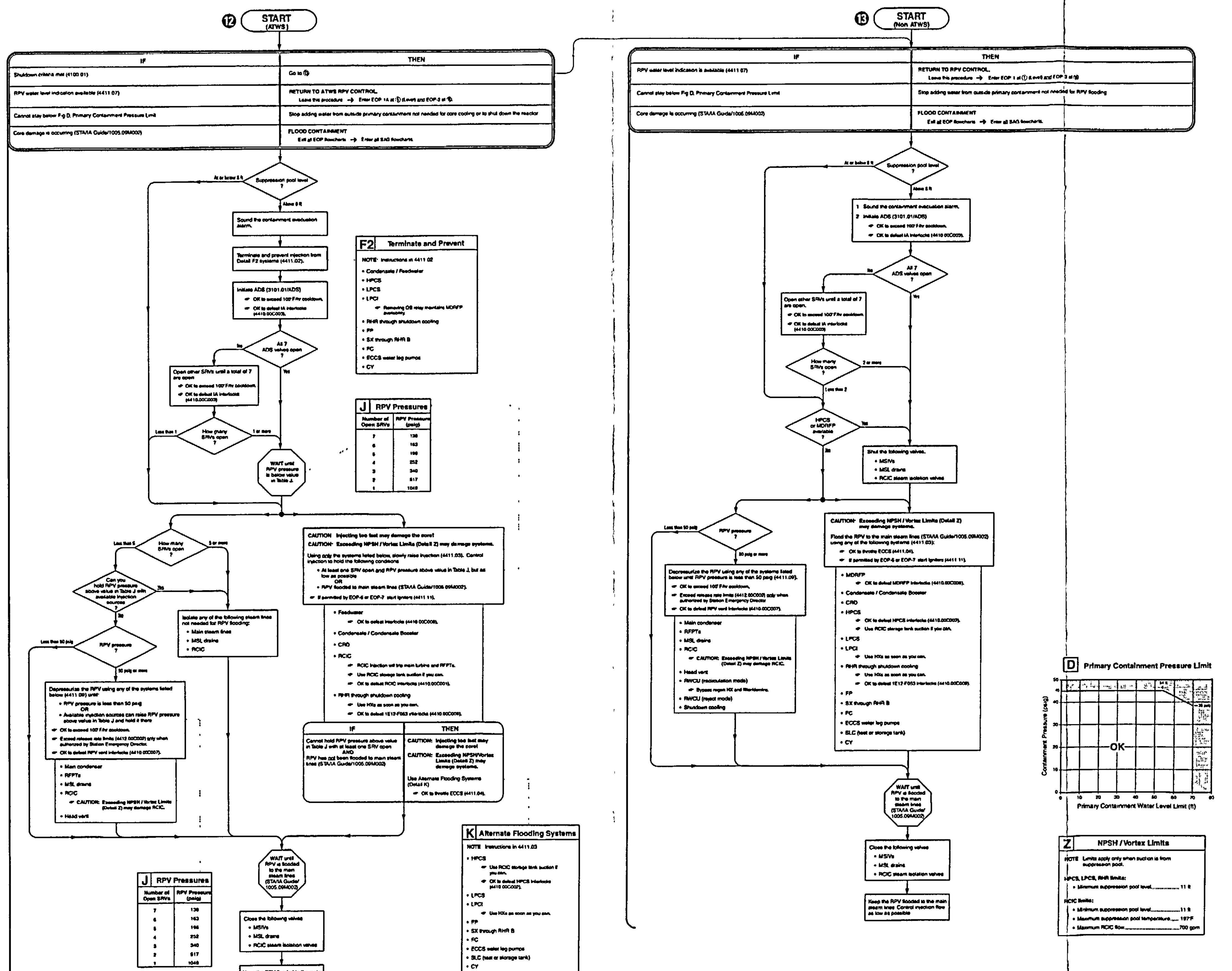
- Minimum suppression pool level....................11 ft
- Maximum suppression pool temperature.......197°F
- Maximum RCIC flow ....700 gpm

EMERGENCY OPERATING PROCEDURE	CLINTON POWER STATION	
RPV CONTROL	PROCEDURE NO 4401 01	REV 26
Approval Documentation on File	SHEET 1 OF 1	

ATTACHMENT 4

Clinton Power Station EOP-2
RPV Flooding

EOP-2 RPV FLOODING



EMERGENCY OPERATING PROCEDURE	CLINTON POWER STATION	
RPV FLOODING	4403.01	26
Approval Documentation on File	Sheet 1 of 1	