Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-5056
DCP/NRC800
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April 4, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: REVISED RESPONSES TO NRC QUESTIONS REGARDING THE ADVERSE SYSTEMS INTERACTION EVALUATION REPORT, WCAP-14477.

REFERENCES:

1. Letter from NRC to Westinghouse (Huffman to Liparulo), Discussion Items for AP600 Meeting on Adverse Systems Interactions, dated 10/3/96.

2. Letter from NRC to Westinghouse (Huffman to Liparulo), Summary of Telephone Conference to Discuss AP600 Adverse Systems Interactions Questions (ASI), dated 1/8/97.

3. Letter from NRC to Westinghouse (Huffman to Liparulo), Summary of Telephone Conferences to Discuss Westinghouse Responses to AP600 Adverse Systems Interaction Report Discussion Items, dated 3/4/97.

Dear Mr. Quay:

Reference 1 provided discussion items regarding the Adverse Systems Interaction Evaluation Report, WCAP-14477. Initial responses were provided to the NRC by fax on 12/4/96, 12/19/96, and 1/13/97; and telecons were held on 12/20/96, 1/29/97, 1/30/97, and 2/19/97 to discuss those responses. References 2 and 3 provide a summary of those telecons and include a copy of the initial responses.

During those telecons. Westinghouse agreed to revise certain initial ASI responses and provide those revised responses to the NRC. Attached are the promised response revisions, except for discussion item 4, which relates to design and operation of the AP600 fan coolers and chilled water system. As stated in the initial response for discussion item 4, upon completion of an ongoing review, Westinghouse will provide a response under separate cover. This work will be completed before 4/30/97.

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This letter provides a partial completion of Open Item Tracking System (OITS) item 3961. Completion of the response to discussion item 4 will complete Westinghouse action for OITS item 3961.

The NRC is requested to review the attached response revisions and contact Robin K. Nydes at (412) 374-4125 if additional clarification or discussion is needed.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

jml

Attachment

cc: W. C. Huffman, NRC (w/Attachment)
A. Levine, NRC (W/Attachment)
N. J. Liparulo, Westinghouse (w/o Attachment)

# AP600 Open Item Tracking System Database: Executive Summary

**Date: 4/3/97**

**Selection:** [item no] between 3961 And 3961 Sorted by Item #

| Item No. | Branch | DSER Section/ Question | Type | Title/Description Detail Status | Resp Engineer | (W) Status | NRC Status | Letter No. / | Date |
|---|---|---|---|---|---|---|---|---|---|
| 3961 | | 20. | TEL-OI | | RTNSS/ASI/Corletti | Action W | Action W | NTD-NRC-97-5056 | |

The NRC sent a letter on 10/3/96, Discussion Items for AP600 Meeting on Adverse System Interactions, which contains 51 questions.

Westinghouse to draft responses to discussion items by Nov 18 to support a Dec 2 resolution meeting. rkn 10/14/96
Responses faxed 12/4, 12/19, and 1/13/97. Telecons held 12/20, 1/29, 1/30, and 2/19. All issues technically resolved. Owe revised responses.
All revised responses, except Q4 for fan coolers, went to mgt review 4/2/97 for 4/4 issuance to NRC. Bechtel input for Q4 (regarding design and operation of the AP600 fan coolers and chilled water system) due 4/18 for Westinghouse evaluation. rkn 4/2/97
Response to all but discussion item 4 sent by NSD-NRC-97-5056 dated April 4, 1997.
Action W to evaluate Becthel input then provide response to Q4 to NRC. rkn 4/3/97

 Total Records: 1

REVISED RESPONSES TO QUESTIONS, COMMENTS, AND DISCUSSION ITEMS
CONCERNING THE WESTINGHOUSE AP600
ADVERSE SYSTEMS INTERACTION REPORT

**Introduction:**

The Adverse Systems Interactions report does a reasonably good job in describing interactions between single components and/or systems in the AP600. However, there appears to be relatively little consideration of "integral" effects: for instance, the cumulative impact of operating several non-safety systems for a period after which the passive systems might be required to operate. The staff recognizes that the actual plant response to a transient or accident may depend upon operator (or automatic) actuation of the "defense-in-depth" (DID) systems. The staff is concerned, however, that the integral-systems impact of operating several DID systems, either in parallel or serially, may affect the plant's thermal-hydraulic state sufficiently to compromise the ability of the passive systems to "take over" should they be required at a later time, in the event that an accident or transient worsens, or if some or all of the DID systems were to fail or be shut off. Has Westinghouse evaluated these types of integral and time-dependent effects?

**Revised Response:**

The question addresses two types of interactions associated with the operation of the active nonsafety systems: integral effects and time-dependent effects. Integral effects of the active systems on the performance of the passive systems in performing their safety functions has been considered. However, the integral effects are minor, compared with the individual effects of the system interactions accounted for in the report, presented on a system by system basis. For example, a primary function of the CVS makeup pumps is to maintain inventory in the RCS. The CVS makeup pumps operate based on the RCS parameter of pressurizer water level. Their potential adverse interactions on the passive safety systems is discussed in sections 2.2.3 and 2.2.5 of the ASI report. The SFW pumps operate to maintain an RCS heat sink via the steam generators based on the parameter of steam generator water level. These interactions are also discussed in section 2.2.9. If these systems were to operate in conjunction (as would be expected), their operation, although effecting RCS conditions, would not adversely interact with the operation of the passive systems moreso than already has been discussed in the report. Since these systems operate in response to thermal-hydraulic conditions in the RCS, as do the passive systems, their operation in tandem does not place the RCS conditions outside of those that have already been considered in their operation independently. For example, the CVS can not raise the water level in the pressurizer to a greater level with or without the SFW pumps operating. The SFW can not overfill the SGs to a greater degree with or without the CVS makeup pumps operating. In addition, their operation in tandem during events where it could be a factor has been considered, such as a SGTR event. That is why both the SFW pumps and the CVS pumps are isolated on a high SG water level, to reduce the potential for either pump overfilling the SG following a SGTR.

With regards to time-dependent effects, these have been addressed on a system by system basis in the report. For instance, operation of the SFW following a loss of normal feedwater can delay PRHR operation As noted in the report, this is regarded as a favorable interaction. Consistent with the discussion above, the operation of the CVS in conjunction with the SFW has no real impact on any time dependent effects of the SFW and /or CVS operation.

These systems were chosen as examples because they are the active systems that have the most direct impact on the operation of the passive systems, especially in the short term following a postulated event. This discussion illustrates that since operation of the active systems is generally dependent on single, independent RCS parameters, their operation in an integral fashion poses no additional adverse interactions than those which have already been addressed in the report.

3. Injection of saturated water due to energy input to IRWST is mentioned several times, as is the beneficial ("quenching") effect of injecting subcooled water. There are a number of interactions that can cause the IRWST water to become saturated prior to injection to the RCS. Westinghouse should confirm that in the Chapter 6/15 analyses, that saturated conditions are assumed in the IRWST to minimize the beneficial effect of subcooled injection or explain why the assumption is not necessary.

**Revised Response:**

The range of initial IRWST temperatures assumed in the design basis safety analyses is: 50°F < T < 120°F. Of all the Chapters 6 and 15 analyses, IRWST injection only occurs for a LOCA. For the LOCA analyses, the maximum initial IRWST temperature (120°F) is assumed for conservatism. During the course of a LOCA, the IRWST is heated up by PRHR heat transfer and ADS injection. Test data shows that there is significant temperature stratification in the IRWST, such that the portion of the IRWST below the PRHR tubes remains subcooled for an extended length of time. This phenomenon is modeled by the LOCA analysis codes (WCobra/Trac and NOTRUMP) for the IRWST injection in the small and large break LOCA analyses of Chapter 15.6. Saturated water injection is postulated to occur during long-term cooling following a LOCA event and is modeled in the Chapter 15.6 long term cooling analysis as appropriate. During the one-half inch equivalent diameter cold leg break presented in SSAR Chapter 15.6, operation of the PRHR for more than 15,000 seconds prior to ADS actuation heats the upper IRWST to near saturation. When ADS stages 1-3 actuate, the region of the IRWST above the bottom PRHR tube elevation boils. ECCS performance under this condition is bounded by the small break LOCA long-term cooling sump injection window of SSAR Section 15.6.5.4C. In that case, a saturated sump is modeled with a much lower liquid driving head available (approximately 23 feet lower liquid level) and similar decay heat present.

11. For the Plant Control System (Section 2.2.13), what impact does the loss of offsite power followed by starting and sequential loading of these systems on the non-safety diesels have on system interactions?

**Revised Response:**

There are no additional adverse system interactions associated with the nonsafety active systems due to a loss of offsite power and sequential loading of these systems on the nonsafety-related diesels. As discussed in this report, nonsafety-related systems are assumed to operate following an accident if their operation can be shown to be detrimental to plant safety. Therefore, for a loss of offsite power coincident with an accident, if operation of a nonsafety system was shown to be detrimental, then it is assumed in the SSAR Chapter 15 analysis.

To avoid any potential adverse interactions with regards to automatically loading the nonsafety-related standby diesel generators, the following guidelines were used in developing the loading sequence.

1. Lighting loads are sequenced on first because of their importance to the ability of the operators to respond.

2. Regulating transformers for Class 1E and non-1E systems are sequenced in the first block. If a regulating transformer is in use, then the associated inverter is inoperable and control will be impaired until the regulating transformer is powered.

3. Equipment required for direct support of the diesel generator are sequenced as soon as possible to minimize the possibility of diesel generator failure.

4. The startup feedwater pump is sequenced next because of its importance to steam generator heat removal.

5. Cooling water systems are sequenced such that the heat sink is in operation before a system is started, i.e., service water before component cooling before normal residual heat removal. Likewise, the air cooled chiller pump is started before the air cooled chiller.

6. The makeup pump is allowed to start toward the end of the sequence. This pump starts on demand, not on the sequencer signal. If the makeup pump were sequenced early, there would be a danger that the makeup pump demand signal would be coincident with the start of another pump and create a load step too large for the diesel generator.

7. Systems which have an inherent "ride-through" capability are sequenced last. This group of loads includes battery chargers, air compressors, heaters, and selected fans.

Although no adverse interactions have been identified with the loading of the nonsafety-related standby diesel generators, these guidelines afford the AP600 with a highly reliable standby ac power supply.

**12.** Isolation of the RCDT to prevent overpressurization (p. 2-26) is indicated to be non-safety-related. What are the implications of failure to accomplish this isolation? Is a tank rupture credible? If so, what potential systems interactions might occur?

**Revised Response:**

If RCDT isolation was not accomplished, then the RCDT relief valve would open spilling the contents of the tank into the containment. However, if ADS discharge continues, the RCDT relief valve would not be sufficient to prevent the tank from being overpressurized (design pressure 10 psig).

Due to the small 1" line that connects the ADS header to the RCDT, the amount of bypass would be insignificant. In addition, any steam flow that bypassed the sparger would condense and collect in the containment. Since the RCDT is located at the bottom of containment, and

will flood anyway due to the LOCA, the effect is negligible.

Rupture of the RCDT would not adversely effect any safety-related system, structure or component. The resultant peak pressure that would occur in the RCDT prior to tank rupture is much less than the pressure necessary to cause an internally generated missile (275 psig) and therefore its rupture does not pose a risk to any safety-related components.

17. The discussion on CMT/PRHR interactions does not take into account the possible role of the PRHR in system-wide interactions, such as those observed in OSU testing.

**Response:**

The system-wide CMT / PRHR interactions were considered in development of Section 2.3.1.4.

CMT recirculation operation cools the RCS and, therefore, impacts PRHR HX effectiveness since the PRHR heat removal capability will decrease with lowering RCS temperatures as the temperature difference between the RCS and IRWST decreases. When the CMT recirculation and associated RCS cooling slows the RCS can heat up somewhat because PRHR HX may not be able to match decay heat at lower RCS temperatures. The RCS then subsequently heats up slightly to a temperature where the PRHR can match the core decay heat rate.

In tests of the AP600 passive safety systems performed at Oregon State University, oscillatory behavior in the operation of the passive safety systems was observed. Various system performance characteristics were noted as oscillatory in nature including PRHR flow rates during LOCAs and long-term CMT / IRWST injection flow rates. These oscillatory behaviors were studied (Reference 1), and it was determined that their occurrences in the test facility is partly due to the scaling of these tests. These oscillatory patterns may exist in the plant, but their magnitude will be low. Nonetheless, these oscillations cause no adverse system interaction in and of themselves, but rather represent the operation of the passive systems in response to the thermal-hydraulic conditions present in the RCS following significant dynamic events such as a LOCA and or ADS.

A potential concern would be the response of the operators during an event when such oscillatory behavior patterns would occur. Although these oscillatory patterns are real and may be expected to occur in the AP600, they are expected to be of such a small magnitude, that they would not cause the operators to take actions that would be detrimental to the plant. For example, during IRWST injection, where oscillations in RCS pressure of 1-2 psi would result in oscillatory safety injection flow rates. However, the pressure instrumentation that the operators would use to monitor the RCS has a range of 0-3300 psig and such oscillations would be undetectable. In addition, no safety injection flow rates are provided. During this phase, the operators will be using core exit thermocouples as the primary indication of adequate core cooling, and will be verifying valve position indications to ensure proper operation of the passive safety systems. Backup indications include RCS hot leg level and IRWST water level. Oscillations in hot leg water level will be noted by the operators, however, there are no additional operator actions qued on hot leg level once the passive safety systems (including ADS) have actuated. During this scenario, the operators will be trying to establish operation of the normal residual heat removal system (RNS) to provide low pressure RCS makeup and eventually restore the plant to normal conditions. Prior to RNS operation,

the actions operators take will not be affected by small oscillations in the plant parameters. For example, no actions are taken based on core exit thermocouples until their indication would reach 700°F. Once RNS pump operation is established, the oscillatory behaviors noted in the tests will not be a factor, due to the high RNS injection flow rate.

Reference 1 - WCAP-14292, Rev. 1, "Low-Pressure Integral Systems Test at Oregon State University, Test Analysis Report," September 1995.

**26.** Section 2.3.5.5 (PRHR/RCS) does not consider the effect of the PRHR on RCS inventory and thermal-hydraulics. Further, the issue of stratification in the primary system and its possible impacts are not addressed.

**Revised Response:**

Subsection 2.3.5.5 does specifically address the PRHR effect on RCS inventory and high-lights some thermal-hydraulic effects, however, the specific aspects of interest in the first sentence above, beyond thermal stratification, need to be clarified.

Paragraph 2 discusses both forced flow heat removal from reactor coolant pumps and natural circulation flow heat removal, single-phase and two-phase heat transfer, and the IRWST heatup. Paragraph 2 continues with the statement that "As the RCS cools down and contracts, RCS voiding may occur, which can eventually change the PRHR HX transfer process to steam condensation heat transfer..."

The issue of thermal stratification is not specifically discussed, however, the final statement in this section is intended to bound the identified design interactions. The last sentence states that "These design interactions have been confirmed as part of the testing and plant analyses."

The discussion in Subsection 2.3.5.5 will be revised to specifically address this comment by including information related to thermal stratification interactions. The response to RAI 440.358 addresses the issue of inadvertent ADS due to flashing in the cold leg due to thermal stratification in the cold leg.

**28.** Section 2.3.7 addresses PCCS/containment interactions, and discusses only spurious operation of the PCCS. It does not address at all the failure of the PCCS, e.g., effects on containment if no water cooling were available to augment external containment heat transfer.

**Revised Response:**

The failure of passive containment cooling to actuate can only occur with beyond-design-basis component failures due to multiple failures of redundant safety-related valves. The containment interactions that occur for the beyond-design-basis failure of the passive containment cooling system has been accounted for in the AP600 PRA. Section 2.3.7 of the ASI report has been updated to address this issue.

29. In addition to human factor related concerns raised in question 1, 4, 6, and 15 above, several other human reliability issues could be elaborated on in this report.

a. The operator has the capability, in a station blackout, to override the automatic ADS actuation just prior to 24 hours. The rationale for including this actuation, as understood by the staff, is to ensure that sufficient battery power is available to open the ADS valves. Suppose that an operator overrides the actuation at 24 hr (minus), but then finds at some time thereafter (say, 36 hours) that it is necessary to actuate the ADS. What are the effects of such a scenario? Is sufficient power available? When does power cease to be available? What alternatives would the operator have if power were not available? Are there other situations in which delaying an action (either actuating a system or overriding its actuation) could have a significant impact on plant response?

b. At the end of Section 2.3.6.2, Westinghouse states that the AP600 ERGs provide guidance on manual actuation of ADS-1 to terminate an SGTR event. How is this addressed in terms of human reliability? What if the operator makes an error and causes actuation of the entire ADS system?

**Revised Response**

a. During a prolonged station blackout event, the emergency procedures provide the operators with criteria to manually override the timer-driven ADS actuation for scenarios where the plant is in a stable condition. In this situation, if the CMTs are full, the pressurizer water level is stable, and the IRWST water level is stable, the operators would be permitted to manually override the timer-driven ADS actuation. It is intended that the manual override of the timer-driven ADS actuation would be implemented about 20-22 hours from the start of the event. Loads supplied by the 24-hour battery, including the PMS actuation cabinets would be de-energized at the time of the override decision. This would preserve the battery power of the two electrical divisions powered by the 24-hour battery and these batteries would have at least a two-hour charge for a later ADS actuation, if needed.. The operators would continue to monitor plant parameters via instrumentation that is powered from one of the two electrical divisions with 72-hour battery capacity.

If the event continues beyond 24 hours, the 72-hour batteries allow the operators to monitor the plant parameters to decide if ADS actuation is needed. Also, there is ample time before the 22nd hour is reached to evaluate the plant conditions and decide whether the manual ADS override is needed or not. The AP600 ERGs will be revised to address the above described operator actions of overriding and re-establishing ADS actuation following a prolonged station blackout event.

Since the batteries are conserved, as described above, sufficient power is available for actuation of ADS during the 22-72 hour time frame. After 72 hours, the ancillary diesel generators will allow continued monitoring the plant conditions. If ancillary diesel generators can not be started, and other ac power sources can not be brought to the site from other locations, the emergency procedures for this beyond design basis scenario would direct the operators to manually actuate the automatic depressurization system while post-accident monitoring capability is still available.

In the spectrum of events studied for AP600, the above-discussed action is the only known case where overriding a safety-related automatic system actuation is envisioned, and it applies

to a rare event (e.g. station blackout) which is made even more improbable by considering it be prolonged over 20 hours.

b. The AP600 ERGs guide the operators on the use of ADS in the recovery from a steam generator tube rupture. Following such an event, the recovery strategy includes isolating the faulted steam generator and depressurizing the primary side to equalize its pressure with the secondary side, to terminate the loss of coolant. The preferred method of depressurizing the primary side is by use of the auxiliary spray which requires operation of the CVS makeup pumps. If auxiliary spray is unavailable, the operators will use a single, first stage ADS valve to manually depressurize the system. These valves are designed to be used in such a manner, and are capable of being controlled by the operators to throttle flow, and thereby allow the operators to carefully control the depressurization rate. If the operator makes an error and actuates the entire ADS system in the above mentioned case, the ADS operation will successfully reduce the RCS pressure, thereby terminating the primary to secondary leak, and in conjunction with the passive core cooling system, will maintain core cooling. This success path is modeled in the AP600 PRA in the SGTR event trees, as one of the multiple success paths in a SGTR event.

Boron dilution following RCS depressurization during tube rupture events has been evaluated in Reference 2. Reference 2 shows that during SGTR accidents, the AP600 provides multiple layers of defense to equalize the RCS and faulted SG pressures, isolate the break and terminate the loss of RCS coolant without draining the CMTs or manually actuating the ADS. To evaluate the likelihood and degree of dilution of these SGTR scenarios, various scenarios were evaluated including:

- An unisolated secondary system
- Loss of primary side heat removal
- Operator error to actuate full ADS

Boron dilution resulting from these scenarios was evaluated in Reference 2, and return to power will not occur following full ADS during a steam generator tube rupture event.

In conclusion, actuation of the ADS during a steam generator tube rupture requires multiple system failures or operator errors of commission. Furthermore, ADS actuation results in successful mitigation of the accident, and results in no excessive boron dilution that would cause a return to criticality.

Reference 2 - NSD-NRC-97-5035, "AP600 Multiple SGTR Analysis Report and Response to Request for Additional Information," March 24, 1997

**32.** The emphasis in the report is on passive-passive and passive-active interactions that can directly affect the passive safety systems. Are there any "active-active" interactions of interest, i.e., an interaction between two active systems that could act to impede (or cause spurious actuation of) a passive system? This could include secondary or tertiary effects, such a feedback from the turbine/generator system through the steam generator to the primary loop. For example: Table 2-1 does not show any systems beyond the secondary side of the RCS. How about effects deriving from: turbine/generator (e.g., closing of stop valves causing level transient in SG that actuates PRHR); other indirect interactions--such as transients caused by malfunctions of the EHC system; interactions caused by spurious actuation of the reactor

- protection system or failure of turbine-over-speed protection.

**Revised Response:**

Spurious actuation of passive systems are evaluated in the SSAR Chapter 15 analysis. Therefore, any combination of active systems that could cause a spurious actuation of a safety system is bounded by the analysis presented in Chapter 15 of the SSAR. This report identifies interactions that could occur that could degrade the performance of the safety systems once they have been actuated. These interactions are then accounted for in the SSAR Chapter 15 accident analysis.

Interactions between secondary and / or other nonsafety-related systems have been considered in the determination of the initiating event frequencies for spurious actuation of the passive systems presented in the AP600 PRA.

**47.** Is it possible for a secondary side break or rupture (within containment) to cause actuation of the ADS system? Under such circumstances, significant additional water inventory will be added to containment; are there any adverse conditions possible from such a scenario (such as boron dilution)?

**Revised Response:**

As demonstrated in the SSAR Chapter 15 Analyses, there are no design basis secondary side breaks that would result in ADS actuation. As ADS actuation on secondary side breaks is undesirable for various reasons (degradation of RCS pressure boundary for Condition II events, increased mass and energy release for steamline breaks inside containment), the sets of assumptions considered in the SSAR analyses have been selected to maximize the potential for ADS operation. In all cases, ADS actuation is not predicted for any design basis secondary side break.

However, even if ADS is postulated following a secondary side break, the potential for a boron dilution resulting in a return to criticality is not a concern. For example, a feedline break inside containment would result in the contents of a single steam generator being dumped to the containment. Following ADS, the accumulators, core makeup tanks, and IRWST would inject into the reactor vessel and spill into the containment via the fourth stage ADS valves. There it would mix with the unborated contents of the steam generator. The following table presents the relative masses of the various sources of water inside containment. As shown in the table, the mass of unborated water from a steam generator is very small compared with the available inventory of borated makeup from the accumulators, CMTs and IRWST. As the contents of the steam generator would be mixed with that of the PXS tanks prior to entering the RCS (via containment recirculation) the effect on RCS boron concentration during this phase of the event would be very small. This dilution mechanism is bounded by the results of the analysis of the dilution resulting from ADS following a SGTR, where the unborated contents of a steam generator is injected directly into the RCS via the broken SG tube discussed in the response to discussion item 29b.

| System / Component | Approximate Mass (lbm) | Minimum Boron Concentration (ppm) |
|---|---|---|
| Reactor Coolant System | 340,000 | 0 |
| Steam Generator Secondary Side | 107,210 | 0 |
| IRWST | 4,619,000 | 2600 |
| Core Makeup Tanks (2) | 248,000 | 3500 |
| Accumulators | 206,700 | 2600 |

50. Related to the human factors aspects of question 29 above concerning actions following an extended station blackout, there may be a need for operators to take some additional actions if temperature limits are being approached in the I&C cabinets due to lack of normal control room cooling. What actions could the operator be expected to take and is there a potential for errors of commission or omission? What would be the consequences of a such errors?

**Revised Response**

The main control room habitability system is described in SSAR section 6.4. As discussed in this section, the design basis for the main control room is a maximum occupancy of up to 11 persons (subsection 6.4.1.1). The heatup evaluation of the control room presented in section 6.4.4 concludes that the temperature in the control room will remain within limits for up to 72 hours. This evaluation assumed the maximum occupancy limit of 11 persons.

As discussed below, there is no need for operator actions for cooling I&C cabinet rooms, or the control room during the 72-hour period following an extended station blackout or loss of ventilation system. Thus, there are no postulated omission or commission errors.

The temperature in the I&C rooms following a loss of the nuclear island non-radioactive ventilation system remains below 120°F over a 72 hour period. If cooling is required beyond 72 hours, the room doors are opened and cooling is achieved via ancillary fans powered from the ancillary diesels. Alternatively, portable fans may be obtained from offsite, and used to deliver convection cooling to each of the electrical equipment rooms using ambient air, as needed.

The same discussion applies to the main control room.