COVER SHEET

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MAR - 7 1997
Brian A. Mc Intyre

FAX

To: Bill Huffman - NRC

Subject: NRC/W NOTRUMP Meeting

Date: March 5, 1997

Pages: 13, including this cover sheet.

COMMENTS:

Bill,

Attached is a partially complete table on the "road-map" to close OIS/RAIs/etc. for NOTRUMP V&V related issues. Please give Ralph Landry a copy so he can have a chance to review before we meet. We'll discuss this as part of Thursday meeting. Thanks.

Earl

cc: L. Hochreiter (FAX), B. Osterrieder, M. Young, B.MCINTYRE (NRC Informal correspondence File), A.Gagnon

From the desk of...

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SDSER Confirmatory Item #	Description of Item	Reference Where Answered
DSER-CN 21.6.2.4-1	The application of SIMARC drift-flux is acceptable pending confirmation of the model through benchmark and assessment of code to be provided in NOTRUMP Final Validation Report (FVR).	The NOTRUMP FVR (WCAP-14807, Revision 1) has been submitted.
DSER-CN 21.6.2.4-2	The modifications made to the NOTRUMP drift-flux correlations are acceptable pending confirmation of the model through benchmark and assessment of code to be provided in NOTRUMP FVR.	The NOTRUMP FVR (WCAP-14807, Revision 1) has been submitted.
DSER-CN 21.6.2.4-3	Westinghouse needs to verify that the NOTRUMP code does not use the Bjornard and Griffith modification.	
DSER-CN 21.6.2.4-4	Westinghouse needs to verify that heat link methodology for transition boiling is not used in AP600 NOTRUMP calculations.	
DSER-CN 21.6.2.5-1	The acceptability of the PRHR model used in NOTRUMP is contingent on a finding that the PRHR data are applicable.	
DSER-CN 21.6.2.7-1	Comparisons of the NOTRUMP code simulations to the OSU and SPES-2 test data in the NOTRUMP FVR should confirm the applicability or insensitivity of the NOTRUMP flow regime models to the key system response parameters.	The NOTRUMP FVR (WCAP-14807, Revision 1) has been submitted.

SDSER Open Item #	Description of Item	Reference Where Answered
DSER-OI 21.6.2.2-1	Westinghouse needs to identify which information from the NOTRUMP-related RAI responses will be incorporated into NOTRUMP-related documentation.	This table identifies where RAI information is captured and closes the OI. Note that the NOTRUMP FVR is intended to be the only NOTRUMP related documentation summarizing the NOTRUMP code for use on AP600 plant calculations.
DSER-OI 21.6.2.2-2	Westinghouse needs to submit the NOTRUMP FVR.	The NOTRUMP FVR (WCAP-14807, Revision 1) has been submitted.
DSER-OI 21.6.2.4-1	Westinghouse needs to explain provisions to ensure that volumetric-based momentum equations will be used for all AP600 calculations.	
DSER-OI 21.6.2.4-2	Westinghouse needs to submit the assessment cases demonstrating acceptability of casting equations in net volumetric form.	The NOTRUMP FVR (WCAP-14807, Revision 1) has been submitted. Section 3.5 contains the assessment cases.
DSER-OI 21.6.2.4-3	Westinghouse needs to submit the assessment cases for the Horizontal Stratified Flow Model.	After the preliminary calculations, this model was no longer used. The preliminary calculations were redone without this model, and therefore the model description is not included in WCAP-14807. As a result, the assesments are not needed and not performed.
DSER-OI 21.6.2.4-4	Westinghouse needs to explain provisions to ensure that options to override the default flow partitioning will be used for all AP600 calculations.	
DSER-OI 21.6.2.4-5	Final acceptance of Mixture Overshoot Logic must await completion of benchmark and assessment calculations to be included in NOTRUMP FVR	The NOTRUMP FVR (WCAP-14807, Revision 1) has been submitted.
DSER-OI 21.6.2.4-6	Determination of additional (to G-2 tests) separate effects level swell tests necessary for code qualification.	Section 4 of WCAP-14807, Revision 1 contains GE and ACHILLES separate effect level swell test simulations in addition to G-2.
DSER-OI 21.6.2.4-7	Acceptance of modified pump model must await submittal of benchmark calculations.	Benchmark submitted in Section 3.7 of WCAP-14807, Revision 1
DSER-OI 21.6.2.4-8	Acceptance of implicit treatment of gravitational head await staff review of the benchmark calculations.	Benchmark submitted in Section 3.4 of WCAP-14807, Revision 1

DSER-OI 21.6.2.4-9	Acceptance of the horizontal flow levelizing model must await submittal and staff review of benchmark calculations.	Benchmark submitted in Section 3.3 of WCAP-14807, Revision 1
DSER-OI 21.6.2.4-10	The staff cannot determine the adequacy of the birthing logic until benchmark is submitted and reviewed.	After the preliminary calculations, this model was no longer used. The preliminary calculations were redone without this model for inclusion in WCAP-14807, Revision 1. As a result, no benchmark was performed and the staff does not need to review the birthing logic.
DSER-OI 21.6.2.4-11	Acceptance of the Zuber critical heat flux correlation for AP600 SBLOCA analysis will be determined after review of the NOTRUMP FVR.	The NOTRUMP FVR (WCAP-14807, Revision 1) has been submitted.
DSER-OI 21.6.2.4-12	Acceptance of the smoothing logic between choked and unchoked flow must await submittal and review of the Final NOTRUMP Validation Report.	The NOTRUMP FVR (WCAP-14807, Revision 1) has been submitted.
DSER-OI 21.6.2.4-13	Acceptance of the logic schemes for application of fluid node stacking, mixture level overshoot, and bubble rise changes must await the submittal of the assessment cases in the NOTRUMP FVR.	The NOTRUMP FVR (WCAP-14807, Revision 1) has been submitted.
DSER-OI 21.6.2.5-1	The NOTRUMP code tended to overpredict the ADS flow rates in the preliminary OSU and SPES-2 comparisons. The models affecting the fluid entering the ADS piping, particularly for the hot legs and pressurizer, need to be reviewed in the NOTRUMP FVR.	The NOTRUMP FVR includes the OSU and SPES-2 simulations which were redone after the preliminary calculations. Included in the report are comparisons (test data to simulation) of ADS flows.
DSER-OI 21.6.2.5-2	CMT thermal stratification was not captured in the CMT tests. Westinghouse will further investigate inability to properly characterize CMT thermal stratification and these assessments will be provided in the NOTRUMP FVR.	Section 6 of the NOTRUMP FVR contains the CMT test simulations which were redone after the preliminary calculations.
DSER-OI 21.6.2.5-3	The staff must receive and evaluate the CMT and ADS test simulations that were identified in Table 21.7 of the SDSER.	Sections 5 and 6 of the NOTRUMP FVR contain these test simulations.

DSER-OI 21.6.2.6-1	The staff must receive and evaluate the benchmark calculations that were identified in Table 21.8 of the SDSER.	Section 3 of the NOTRUMP FVR contains these benchmarks with the exception of the Birthing Logic and Horizontal Stratified Flow ones which were not performed because the coding was not used in the NOTRUMP FVR calculations and will not be used in AP600 plant calculations.
DSER-OI 21.6.2.6-2	The staff must receive, review, and evaluate the adequacy of the separate-effects testing relative to level swell and void fraction distribution.	Section 4 of the NOTRUMP FVR contains the level swell related test simulations.
DSER-OI 21.6.2.6-3	The staff must receive and evaluate the integral test simulations that were identified in Table 21.10 of the SDSER.	Sections 7 and 8 of the NOTRUMP FVR contain the integral test simulations.
DSER-OI 21.6.2.7-1	Westinghouse needs to address PRHR primary-side heat transfer comparisons between NOTRUMP and OSU/SPES-2 data in the NOTRUMP FVR.	The comparisons for SPES-2 are contained in Section 7 of the NOTRUMP FVR. OSU comparisons were not included because comparable test data was not available.
DSER-OI 21.6.2.7-2	Effects of non-condensible gases on PRHR heat transfer should be addressed in NOTRUMP FVR.	
DSER-OI 21.6.2.7-3	Clarify the use of the COSI condensation model in the AP600 code.	

RAI #	Description of Item	Reference Where Answered
RAI 440.325	Questions on NOTRUMP CAD (WCAP-14206) related to PIRT, NOTRUMP modeling of noncondensible gases, and NOTRUMP 1-D model.	Westinghouse Letter NTD-NRC-95-4594; WCAP-14807, Revision 1 Section 1.3 contains final SBLOCA PIRT.
RAI 440.326	Should include an AP600 plant nodalization and reference to SAR calculations.	Westinghouse Letter NTD-NRC-95-4587; WCAP-14807, Revision 1 Section 1.2 contains AP600 plant noding diagram.
RAI 440.327	Provide a matrix of tests that will be used for assessing each of the PIRT items. Also, identify the models that are to be validated for each test.	Westinghouse Letter NTD-NRC-95-4610; WCAP-14807, Revision 1 Section 1.4 contains table of tests and parameters selected for validation of NOTRUMP for highly ranked PIRT items.
RAI 440.328	Explain what analyses were performed to determine the limiting failure.	Westinghouse Letter NTD-NRC-95-4587
RAI 440.329	Describe the low flow correlations applicable to the prediction of the single and two-phase friction factors in NOTRUMP for AP600 and identify the test data that will be used for the assessment.	Westinghouse Letter NTD-NRC-95-4610
RAI 440.330	Describe the enhancements made to the NOTRUMP code for AP600.	Westinghouse Letter NTD-NRC-95-4587; WCAP-14807, Revision 1, Section 2 contains the NOTRUMP code changes for AP600 calculations.
RAI 440.331	Provide the specific inputs for the code externals used to perform the analyses in the SSAR calculations done in January 1994.	Westinghouse Letter NTD-NRC-96-4630
RAI 440.332	Provide a document describing the methods and models comprising the long term cooling code and describe how the code is initialized from the NOTRUMP code.	Westinghouse Letter NSD-NRC-96-4780
RAI 440.333	Justify the use of a fixed containment pressure boundary condition since the response of the safety systems depend on containment pressure.	Westinghouse Letter NSD-NRC-96-4780

RAI 440.334	Provide a test matrix showing the separate effects and integral tests to be used in the validation of NOTRUMP for AP600.	Westinghouse Letter NTD-NRC-95-4610; WCAP-14807, Revision 1, Section 1.4 contains table of tests and parameters selected for validation of NOTRUMP.
RAI 440.335	Justification for using constant friction factors, particularly at low flow, flow pressure conditions are needed.	
RAI 440.336	Describe if momentum flux is included in AP600 analyses and justify its omission if it is not used.	
RAI 440.337	Demonstrate that the Macbeth correlation is adequate for the low flow and pressure conditions expected for AP600.	
RAI 440.338	Demonstrate that the NOTRUMP pump model can predict the AP600 pump coastdown. Describe and justify the use of the two-phase pump degradation curves for AP600 analyses.	
RAI 440.339	Provide time step and nodalization studies to justify the AP600 nodalization.	
RAI 440.340	Discuss the potential for boric acid build-up and precipitation during long transients for AP600.	Westinghouse Letter NSD-NRC-96-4780
RAI 440.341	Describe in detail the IRWST model including how the sparger and plumes are handled as well as their influence on IRWST injection and PRHR heat removal.	Westinghouse Letter NTD-NRC-95-4587
RAI 440.342	Provide documentation for a) NOTRUMP coding changes along with model benchmarks, b) a description of the containment modeling approach with calculations justifying model, c) a description of the "Long Term Cooling Code", d) a section presenting calculative methods including sensitivity studies and the full break spectrum analysis, and e) a test matrix listing the pertinent separate and integral tests used to benchmark the AP600 small break LOCA code package.	
RAI 440.432	Identify where choking occurs in the ADS tests and discuss why the asymetric effects can be ignored in modeling the three ADS valves as a single flow path.	Westinghouse Letter NTD-NRC-95-4610

RAI 440.433	Explain the effect of not modeling air in the ADS lines on the ADS system pressure, flow, and quality responses.	Westinghouse Letter NTD-NRC-95-4610
RAI 440.434	Demonstrate the ability of the NOTRUMP code to accomodate single phase steam critical flow since the ADS system is expected to transition to high quality steam flow discharge.	
RAI 440.435	Questions related to ADS modeling including explain how NOTRUMP treats the void distribution and release of steam from the two-phase regions in the ADS lines.	Westinghouse Letter NTD-NRC-95-4594
RAI 440.436	Explain how NOTRUMP uses equation 4-1 of RCS-GSR-003 in computations of fluid quality.	Westinghouse Letter NTD-NRC-95-4598
RAI 440.437	Questions on ADS test simulation depressurization rates and length of test simulations.	Westinghouse Letter NTD-NRC-95-4594
RAI 440.438	Explain the inconsistency in the discussion of the effect of tank pressure on quality in the ADS Preliminary Validation Report.	Westinghouse Letter NTD-NRC-95-4587
RAI 440.439	Has the NOTRUMP code been assessed against single-phase and two-phase pressure drop test data in piping systems with expansions and contractions present?	Westinghouse Letter NTD-NRC-95-4610
RAI 440.440	Provide the results of a noding study used to justify the CMT noding in the CMT Preliminary Validation Report. Also, provide the plots of the fluid driving heads calculated by NOTRUMP for each side of the loop.	Westinghouse Letter NTD-NRC-96-4622
RAI 440.441	Were wall temperatures measured in the facility in the CMT and piping? If so, provide comparisons with the NOTRUMP code and discuss the results.	
RAI 440.442	Were wall heat structures modeled in the piping and reservoir? If not, justify the omission; if so describe the model.	
RAI 440.443	Justify the reservoir nodalization and explain the effects of thermal stratification and mixing, or lack thereof, in the S/W reservoir on the NOTRUMP results.	

RAI 440.444	Was a time step study performed for the CMT tests? Discuss and show that the time steps used do not contribute to the error in the NOTRUMP predictions. Are the time steps consistent with those used in the plant model?	
RAI 440.445	The early CMT flow rates appear to be overpredicted even though the time averaged flows show good comparisons. Discuss the NOTRUMP behavior given that the early overprediction of flow may affect the RCS loop temperatures and system behavior later in the event.	Westinghouse Letter NTD-NRC-96-4626
RAI 440.446	Explain why the CMT inlet flow uncertainty is higher than the outlet flow uncertainty measurement for the test. Explain this uncertainty in light of the NOTRUMP inlet flow rate predictions.	
RAI 440.463	Justify use of single node for SG secondary side.	Westinghouse Letter NTD-NRC-95-4587
RAI 440.464	Perform two-phase level swell simulations to justify core noding.	WCAP-14807, Revision 1 Section 4 for level swell, Sections 4.2.5 and 4.3.4 for noding
RAI 440.465	Justify omission of wall heat transier from loop piping and secondary components.	Westinghouse Letter NTD-NRC-95-4594
RAI 440.466	For SIMARC drift flux model ... Please describe how the void fraction is computed for countercurrent flow conditions.	WCAP-14807, Revision 1 Section 2.2
RAI 440.467	Two drift flux models were added to NOTRUMP. Which model is to be used for AP600 calcs? Explain models.	Westinghouse Letter NTD-NRC-95-4587; WCAP-14807, Revision 1 Section 2.3
RAI 440.468	Provide benchmark calcs for level swell and counter current flow data to evaluate flooding.	WCAP-14807, Revision 1 Section 4 for level swell, Sections 3.2 & 3.3 for flooding
RAI 440.469	Provide volumetric flow based momentum equations and code benchmarks for this model change.	WCAP-14807, Revision 1 Section 2.4 for equations, Section 3.5 for benchmark
RAI 440.470	Questions on Horizontal Stratified Flow Model in preliminary NOTRUMP report LTCT-GSR-001	After the preliminary calculations, this model was no longer used. The preliminary calculations were redone without this model, and therefore the model description is not included in WCAP-14807. As a result, the RAI no longer applies.

RAI 440.471	Discuss the use of partitioning models for AP600 calculations and show that there use would not adversely affect the level swell results.	Westinghouse Letter NTD-NRC-95-4598
RAI 440.472	Please explain the liquid reflux flow links and how their use affects level swell, bubble rise, steam production, and fuel cooling.	Westinghouse Letter NTD-NRC-95-4594
RAI 440.473	Please explain how the mixture level overshoot logic does not introduce errors into the NOTRUMP solution that could change the results or conclusions of an AP600 analysis.	Westinghouse Letter NTD-NRC-95-4587; WCAP-14807, Revision 1 Section 2.8
RAI 440.474	Provide the derivations and the expressions for the equations comprising the implicit bubble rise model. Provide level swell calculations verifying this model.	WCAP-14807, Revision 1 Section 2.9 for equations, Section 3.6 for benchmark, Section 4 for level swell
RAI 440.475	Provide a mathematical description of modified pump model and comparison of the old to new model.	WCAP-14807, Revision 1 Section 2.10 for equations, Section 3.7 for comparison
RAI 440.476	Describe mathematically the implicit treatment of gravitational head and provide verification analysis.	WCAP-14807, Revision 1 Section 2.11 for equations, Section 3.4 for verification benchmark
RAI 440.477	Provide new levelizing drift velocity correlation and provide a benchmark for model.	WCAP-14807, Revision 1 Section 2.12 for correlation, Section 3.3 for benchmark
RAI 440.478	Provide a sample calculation showing how the birthing region works.	After the preliminary calculations, this model was no longer used. The preliminary calculations were redone without this model, and therefore the model description is not included in WCAP-14807. As a result, the RAI no longer applies.
RAI 440.479	Provide a comparison of the NOTRUMP Shah condensation model prediction to condensation test data demonstrating applicability of the model to the range of conditions expected in AP600.	Westinghouse Letter NTD-NRC-96-4626
RAI 440.480	Provide a comparison of the results of the as implemented Zuber critical heat flux correlation to test data over the range of conditions expected for AP600 small break LOCAs.	Westinghouse Letter NTD-NRC-96-4626

RAI 440.481	Provide comparisons of the new NOTRUMP two-phase friction multiplier to separate effects and/or integral test data below 250 psia to justify the new models extrapolation formulation.	Westinghouse Letter NTD-NRC-95-4598; WCAP-14807, Revision 1, Section 2.16
RAI 440.482	Provide benchmark of the new critical flow model versus critical flow tests to verify the model. Describe how the model treats the transition from choked to unchoked conditions.	Westinghouse Letter NTD-NRC-96-4630; WCAP-14807, Revision 1, Section 2.13 describes the transition from choked to unchoked conditions.
RAI 440.483	Provide results of a sample fill and drain calculation to demonstrate the Fluid Node Stacking Logic and provide a mathematical description of the logic.	WCAP-14807, Revision 1 Section 2.18 for description, Section 3.8 for demonstration
RAI 440.484	Show the effect of the changes to the transition boiling correlation on peak clad temperature.	Westinghouse Letter NTD-NRC-95-4594
RAI 440.485	Describe the coding and model changes included in the preliminary ADS test simulations and CMT test simulations	Westinghouse Letter NTD-NRC-96-4630; These simulations were redone and included in WCAP-14807, Revision 1
RAI 440.486	Explain why in the preliminary OSU simulations the upper head drains prematurely in the tests.	Westinghouse Letter NTD-NRC-95-4598; These simulations were redone and included in WCAP-14807, Revision 1
RAI 440.487	For the analyses in the OSU Preliminary Validation Report (PVR), provide comparisons of the NOTRUMP liquid levels in the core and upper plenum versus test data.	
RAI 440.488	Discuss the NOTRUMP overprediction of the integrated break flow for the OSU PVR calculation.	
RAI 440.489	Explain why the NOTRUMP code underpredicts the PRHR heat transfer in the OSU PVR and justify how this model results in conservative AP600 SBLOCA ECCS performance predictions.	
RAI 440.490	Explain why the NOTRUMP code overpredicts the downcomer liquid level during this OSU PVR calculation and justify the model result for AP600 plant calculations.	

RAI 440.491	Provide the core inlet and core bypass mass flow rate predictions for the NOTRUMP code.	
RAI 440.492	Provide the core inlet and bypass mass flow rate predictions for the blind two inch cold leg balance line break in the OSU PVR. Also provide the liquid level plots for the upper plenum and core region and the void distribution in the core region.	
RAI 440.493	Discuss the NOTRUMP fast depressurization rate for OSU PVR calculations including the break flow discharge coefficient and the steam generator heat transfer.	
RAI 440.494	Discuss the impact of the delayed CMT-2 drainage on the core/upper plenum level response for the OSU PVR calculation.	
RAI 440.495	Provide the upper plenum and core liquid level plots for this test along with the void distribution in the core.	
RAI 440.496	For this OSU PVR calculation explain why the code overpredicts the liquid inventory in the downcomer and justify that this will not lead to non-conservative predictions of the liquid level in the vessel for AP600 plant calculations.	
RAI 440.497	Explain the statement that the NOTRUMP code allows a "short spurt of flow at the break" in reference to Figure 5.3-22 of the OSU PVR.	
RAI 440.498	For this OSU PVR case, explain the reasons for the highly oscillatory behavior in the PRHR inlet flow calculated by NOTRUMP and why the code predicts a much higher PRHR flow rate.	
RAI 440.499	Can the NOTRUMP code model nitrogen entering the RCS? If not, justify the omission of nitrogen effects on AP600 response following small break LOCAs.	
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Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	3-10-97	NAME:	Cindy Haag
TO:	Joe Sebrosky	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-4277
COMPANY:	US NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
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Cover + Pages 1 + 3

The following pages are being sent from the Westinghouse Energy Center, East Tower, Monroeville, PA. If any problems occur during this transmission, please call:

WIN: 284-5125 (Janice) or Outside: (412)374-5125.

COMMENTS: Joe -

Attached are markup page changes to the AP600 PRA. Specifically, pages 59-37, 59-38, and 59-41. These changes fix the typo pointed out by NRC in/ PRA [redacted] meeting (1/97). This addresses meeting action item 2 in NRC letter dated 3/3/97. W Status in OITS for this action item (#5015) will change to "Confirm-W" and these changes will be included in the next PRA revision unless we hear differently from the NRC.

Cindy

cc: pointers

C Haag
[redacted]
T. Buxter

Sequence 14: Steam Generator Tube Rupture Accident (SGTR-07)

Sequence Frequency: 2.4E-09/year
Contribution to Core Damage: 1.4 percent
Initiating Event Frequency: 5.2E-03/year
Conditional Core Damage Probability: 4.6E-07

Description of Sequence

A steam generator tube rupture initiating event (break size ~~1/8-inch~~ 0.6-inch equivalent diameter) occurs. Due to failures in nonsafety systems, such as startup feedwater or chemical and volume control systems, or failure to identify and isolate the faulted steam generator, the event continues as a challenge to passive core cooling systems, similar to that of a small loss-of-coolant accident event. One or more core makeup tanks inject into the reactor coolant system, and passive residual heat removal is successful. The automatic depressurization system fails and the pressurized reactor coolant system loses inventory through the break into the secondary side. The reactor coolant system inventory loss cannot be made up after the core makeup tanks inject, although the decay heat is being removed by passive residual heat removal. Core damage is postulated due to the inability to provide long-term reactor coolant system inventory makeup. This sequence is assigned to accident class 6E. In this accident class, the reactor coolant system is postulated to be at high pressure and a containment bypass path through the faulted steam generator exists.

Important Modeling Assumptions

The success criteria for this sequence are very conservative. This sequence may not be a core damage sequence since the decay heat is being removed by the passive residual heat removal and reactor coolant system inventory loss is made up by the core makeup tanks for a considerable time period. The loss of reactor coolant is expected to be stopped when passive residual heat removal cooling lowers the reactor coolant system pressure, thus terminating the loss-of-coolant accident and the need for automatic depressurization system and gravity injection.

Risk-Important Failures

Table 59-17 lists the dominant cutsets for this sequence. The dominant risk-important failure is the common cause failure of protection and safety monitoring system engineered safety feature output logic software and manual diverse actuation system actuation (51 percent). This is followed by various protection and safety monitoring system actuation common cause failures.

Credit is taken for the proceduralized operator action to manually actuate safety-related core cooling systems by using the diverse actuation system, if protection and safety monitoring system actuation fails.

Sequence 15: Steam Generator Tube Rupture Accident (SGTR-23)

Sequence Frequency: 2.3E-09/year
Contribution to Core Damage: 1.4 percent
Initiating Event Frequency: 5.2E-03/year
Conditional Core Damage Probability: 4.4E-07

Description of Sequence

A steam generator tube rupture initiating event (break size ~~1/8-inch~~ 0.6 inch equivalent diameter) occurs. Due to failures in nonsafety systems, such as startup feedwater or chemical and volume control systems, or failure to identify and isolate the faulted steam generator, the event continues as a challenge to passive core cooling systems, similar to that of a small loss-of-coolant accident event. One or more core makeup tanks are actuated to inject into the reactor coolant system, and passive residual heat removal is successful. However, reactor coolant pumps fail to trip and this is assumed to prevent core makeup tanks from injecting. The automatic depressurization system fails and the pressurized reactor coolant system loses inventory through the break into the secondary side. The reactor coolant system inventory loss cannot be made up, although the decay heat is being removed by passive residual heat removal. Core damage is postulated due to the inability to provide long-term reactor coolant system inventory makeup. This sequence is assigned to accident class 6E. In this accident class, the reactor coolant system is postulated to be at high pressure and a containment bypass path through the faulted steam generator exists.

Important Modeling Assumptions

The success criteria for this sequence are very conservative. This sequence may not be a core damage sequence since the decay heat is being removed by passive residual heat removal and reactor coolant pumps would be tripped eventually to allow core makeup tank injection. Then, this sequence would behave like the previously discussed SGTR-07 sequence.

Risk-Important Failures

Table 59-18 lists the dominant cutsets for this sequence. The dominant risk-important failure is the common cause failure of reactor coolant pump breakers to open and operator to manually actuate the automatic depressurization system via the protection and safety monitoring system or diverse actuation system (over 95-percent contribution). This is followed by various operator actions associated with failed nonsafety systems.

Credit is taken for the proceduralized operator action to manually actuate safety-related core cooling systems by using the diverse actuation system, if protection and safety monitoring system actuation fails.

Sequence 18: Consequential Steam Generator Tube Rupture Accident (SGTRC-03)

Sequence Frequency: 2.1E-09/year
Contribution to Core Damage: 1.0 percent
Initiating Event Frequency: 6.8E-05/year
Conditional Core Damage Probability: 3.1E-05

Description of Sequence

A consequential steam generator tube rupture initiating event (break size 0.6 inch ~~1/8-inch~~ equivalent diameter) occurs. The starting point of this event may be a transient or a secondary line break (or a stuck-open secondary-side valve). One or more core makeup tanks inject into the reactor coolant system, and passive residual heat removal and the automatic depressurization system are successful. Normal residual heat removal fails and reactor coolant system inventory makeup by in-containment refueling water storage tank gravity injection is successful. Sump recirculation fails. Core damage is postulated due to the inability to provide long-term reactor coolant system inventory makeup and core cooling following failures of the normal residual heat removal system and sump recirculation. This sequence is assigned to accident class 6L. In this accident class, the reactor coolant system is fully depressurized but a potential containment bypass path through the faulted steam generator exists.

Important Modeling Assumptions

This sequence may not be a core damage sequence since the decay heat is being removed by passive residual heat removal and reactor coolant system inventory loss is made up. The loss of reactor coolant is expected to be stopped, thus terminating the loss-of-coolant accident and the need for sump recirculation, due to low reactor coolant system pressure terminating the break flow.

Risk-Important Failures

Table 59-21 lists the dominant cutsets for this sequence. The dominant risk-important failure is the common cause failure to open of explosive valves on recirculation lines (83-percent contribution). This is followed by common cause failure of in-containment refueling water storage tank level transmitters and operator action to open sump recirculation valves (15-percent contribution).

Credit is taken for the proceduralized operator action to actuate normal residual heat removal, and sump recirculation (if automatic actuation fails). Credit is also taken for operator action to actuate core makeup tanks and the automatic depressurization system as a backup to automatic actuation. These actions are not risk-important in this sequence.

FAX to DINO SCALETTI

March 10, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Robin Nydes
Chip Suggs
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #169 (M5.2.5-26)

In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items from the smallest item number on. The relevant entry from OITS related to Open Item #169 (M5.2.5-26) is attached. This item is a good example of why we need to maintain some discipline in statusing our open items. This item as ORIGINALLY asked was satisfied in the August TechSpec revision. However, as a result of that revision, another question was asked on the same section. The new question is related in that it addresses ISLOCA, however, the simple request to correct the entry for a reference in TS 3.4.8 was resolved in August. This item (#169) should be closed on two counts. First, because we corrected the reference in August of 1996 and second, because a more relevant question has been asked by NRC as Q24 (OITS #4970). It seems a reasonable request that NRC at least acknowledge receipt of the change associated with item #169. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed". Thank you.

Jim

Jim Winters
412-374-5290

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
169	NRR/SPLB	5.2.5	MTG-OI		TECHSPEC/Suggs, C.	Closed	Action W		

M5.2.5-26 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) TS 3.4.8 bases section "Applicable Safety Analyses" refers to reference 4 instead of reference 3.

appropriate reference will be referenced. rkn 3/29
Closed - With issuance of the Tech Specs in SSAR Rev. 9.

Action W - We need to explicitly tell NRC where in TechSpec we cover ISLOCA, especially in light of earlier TechSpec RCS pressure boundary valve isolation 3.4.8 per Chapter 5 telecon with NRC on 12/2/96.

The original review comment was resolved in the August 1996 Tech Spec revision. The new question regarding where ISLOCA is addressed in the Tech Specs is logged as Q24 of OITS item 4970. In general, that response (being issued today) states that ISLOCA is a risk-based issue and therefore does not meet the NRC NUREG-1431 criteria for inclusion in the Tech Specs. Because this issue is a duplicate of 4970 Q24 and the original question for item 169 was addressed, this item is W-status = Closed. rkn 3/10/97.

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FAX to DINO SCALETTI

March 10, 1997

CC: Sharon or Dino, please make copies for: Diane Jackson
Ted Quay

Don Lindgren
Richard Orr
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR SSAR Chapter 2

This is a background package for the remaining open item for SSAR Chapter 2 for your action. SSAR Chapter 2 is of interest because by our joint NRC/W schedule, the FSER for this section should be turned into Projects by the end of April. There are 3 Open Items (547, 556 and 4997) with NRC Status of Action W. These items (OITS report attached) have been discussed repeatedly with NRC and the technical description is included in the item's "Status Detail." Items 547 and 4997 were discussed as part of the Senior Management Review on March 3, 1997. We will be providing a markup of Chapter 2 to address items 547 and 4997 by March 14, 1997. This markup and any subsequent discussions should be used for preparation of the FSER. The agreed to changes to the SSAR for Chapter 2 will be included in Revision 12. Item 556 was resolved by changes included in Revision 10 (December, 1996, over 2 months ago) of the SSAR. We believe that no further Westinghouse action is required for item 556. It seems a reasonable request that NRC acknowledge receipt of this information. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N" or "Closed". Thank you.

Jim

Jim Winters
412-374-5290

1 of 2

AP600 Open Item Tracking System Database: Executive Summary

Date: 3/10/97

Selection: [nrc st code]='Action W' And [DSER Section] like '2.*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
547	NRR/ECGB	2.5.4.3-2	DSER-OI		Orr /NRCBM	Action W	Action W	NSD-NRC-96-4738	

Westinghouse should consider and document in the SSAR, the effects of differential settlement.

Closed - Letter NSD-NRC-96-4738 provided a response for this issue to address differential settlement between buildings. In particular, the main steam line leaving the aux. building and entering the turbine building is considered.
NRC Status Update provided in September 5, 1996 letter:
As agreed during the telephone conference on May 1, 1996, Westinghouse should clarify in SSAR section 3.8.5 that the nuclear island basemat will be designed and constructed giving due consideration to the effects of construction sequential loading. This will be treated as Confirmatory Item 2.5.4.3-2.
Action Westinghouse
This will be addressed with the response to item 768 (DSER OI 3.8.5-10).

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
556	NRR/ECGB	2.5.4.8-1	DSER-OI		Orr/Lindgren/BPC	Closed	Action W	NTD-NRC-95-4433	4/3/95

Westinghouse should add COL Action Item 2.5.4.8-1 to the SSAR, requiring that the COL applicant discuss and evaluate site-specific static and dynamic lateral earth pressures and hydrostatic groundwater pressures acting on plant safety-related facilities.

Closed - Combined License item included in SSAR Rev 2, section 2.5.4.1.8.
NRC Status - Review response to request for specific interface requirement for lateral earth pressure. Response was that other site parameters provide for a site with acceptable lateral earth pressures.
Sept 5, 1996 letter incorrectly provided NRC position on item #556 and status was changed to Action - W. The comment was moved to item # 547 , OI# 2.5.4.3-2. NRC to confirm that item # 556 is resolved.
Action W - Westinghouse will identify SSAR reference to AP600 safety-related facilities. NRC letter of 12/9/96
SSAR Revision 10 included information in second paragraph of Section 1.2 on site-specific structures that addresses this question.

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
4997	NRR/ECGB	2.5	RAI-OI		Orr	Action W	Action W		

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RAI 231.34 - Site Design Parameters
Westinghouse uses the term "interface requirements" in SSAR Section 2.2 and Subsection 2.5.2.1 and "site interface criteria" in SSAR Subsection 2.5.4.5. This terminology is unacceptable. Westinghouse should use "site parameter" or "site design parameter," in accordance with 10 CFR 52.47(a)(iii) and past practice on the Advanced Boiling Water Reactor (ABWR) and System 80+ designs. Interface requirements are different than site parameters as distinguished in 52.47(a). Westinghouse is requested to review its SSAR, particularly those sections cited above, and revise it accordingly.

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	MARCH 10 1997	NAME:	JIM WINTERS
TO:	BILL HUFFMAN	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	US NRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 3

The following pages are being sent from the Westinghouse Energy Center, East Tower, Monroeville, PA. If any problems occur during this transmission, please call:

WIN: 284-5125 (Janice) or Outside: (412)374-5125.

COMMENTS:

BILL

ATTACHED ARE MARKUPS OF 3 SSAR PAGES WHICH SHOULD RESOLVE OITS ITEM # 153 ON RC LEAK DETECTION SENSITIVITY. IT WILL GO INTO SSAR REVISION 12 UNLESS WE HEAR FROM YOU. PLEASE ACKNOWLEDGE RECEIPT WITH AN "NRC STATUS" CHANGE.

Jim

cc: LINDGREN
MCINTYRE
CUMMINS
WINTERS
RONUIJUK
ISRAELSON
JEANNE EVANS

normal level signify a possible increase in unidentified leakage rates and alert the plant operators that corrective action may be required. Similarly, increases in containment sump level signify an increase in unidentified leakage. The following sections outline the methods used to collect and monitor unidentified leakage.

5.2.5.3.1 Containment Sump Level Monitor

Leakage from the reactor coolant pressure boundary and other components not otherwise identified inside the containment will condense and flow by gravity via the floor drains and other drains to the containment sump.

A leak in the primary system would result in reactor coolant flowing into the containment sump. Leakage is indicated by an increase in the sump level. The containment sump level is monitored by two seismic Category I level sensors. The level sensors are powered from a safety-related Class 1E electrical source. These sensors remain functional when subjected to a safe shutdown earthquake in conformance with the guidance in Regulatory Guide 1.45. The containment sump level and sump total flow sensors located on the discharge of the sump pump are part of the liquid radwaste system.

Failure of one of the level sensors will still allow the calculation of a 0.5 gpm in-leakage rate within 1 hour. The data display and processing system (DDS) computes the leakage rate and the plant control system (PLS) provides an alarm in the main control room if the average change in leak rate for any given measurement period exceeds 0.5 gpm for unidentified leakage. Unidentified leakage is the total leakage minus the identified leakage. The leakage rate algorithm subtracts the identified leakage directed to the sump.

The measurement interval must be long enough to permit the measurement loop to adequately detect the increase in level that would correspond to 0.5 gpm leak rate, and yet short enough to ensure that such a leak rate is detected within an hour. The measurement interval is less than or equal to 1 hour.

When the sump level increases to the high level setpoint, one of the sump pumps automatically starts to pump the accumulated liquid to the waste holdup tanks in the liquid radwaste system. The sump discharge flow is integrated and available for display in the control room.

Procedures to identify the leakage source upon a change in the unidentified leakage rate into the sump include the following:

- Check for changes in containment atmosphere radiation monitor indications.
- Check for changes in containment humidity, pressure, and temperature.
- Check makeup rate to the reactor coolant system for abnormal increases.

[The minimum detectable leak is 0.02 GPM.

- Check for changes in water levels and other parameters in systems which could leak water into the containment, and

- Review records for maintenance operations which may have discharged water into the containment.

5.2.5.3.2 Reactor Coolant System Inventory Balance

Reactor coolant system inventory monitoring provides an indication of system leakage. Net level change in the pressurizer is indicative of system leakage. Monitoring net makeup from the chemical and volume control system and net collected leakage provides an important method of obtaining information to establish a water inventory balance. An abnormal increase in makeup water requirements or a significant change in the water inventory balance can indicate increased system leakage.

The reactor coolant system inventory balance is a quantitative inventory or mass balance calculation. This approach allows determination of both the type and magnitude of leakage. Steady-state operation is required to perform a proper inventory balance calculation. Steady-state is defined as stable reactor coolant system pressure, temperature, power level, pressurizer level, and reactor coolant drain tank and in-containment refueling water storage tank levels. The reactor coolant inventory balance is done on a periodic basis and when other indication and detection methods indicate a change in the leak rate.

The mass balance involves isolating the reactor coolant system to the extent possible and observing the change in inventory which occurs over a known time period. This involves isolating the systems connected to the reactor coolant system. System inventory is determined by observing the level in the pressurizer. Compensation is provided for changes in plant conditions which affect water density. The change in the inventory determines the total reactor coolant system leak rate. Identified leakages are monitored (using the reactor coolant drain tank) to calculate a leakage rate and by monitoring the intersystem leakage. The unidentified leakage rate is then calculated by subtracting the identified leakage rate from the total reactor coolant system leakage rate. The minimum detectable leak is 0.13 GPM.

Since the pressurizer inventory is controlled during normal plant operation through the level control system, the level in the pressurizer will be reasonably constant even if leakage exists. The mass contained in the pressurizer may fluctuate sufficiently, however, to have a significant effect on the calculated leak rate. The pressurizer mass calculation includes both the steam and water mass contributions.

Changes in the reactor coolant system mass inventory are a result of changes in liquid density. Liquid density is a strong function of temperature and a lesser function of pressure. A range of temperatures exists throughout the reactor coolant system all of which may vary over time. A simplified, but acceptably accurate, model for determining mass changes is to assume all of the reactor coolant system is at $T_{Average}$.

The inventory balance calculation is done by the data display and processing system with additional input from sensors in the protection and safety monitoring system, chemical and volume control system, and liquid radwaste system. The use of components and sensors in systems required for plant operation provides conformance with the regulatory guidance in Regulatory Guide 1.45 that leak detection should be provided following seismic events that do not require plant shutdown.

5.2.5.3.3 Containment Atmosphere Radioactivity Monitor

Leakage from the reactor coolant pressure boundary will result in an increase in the radioactivity levels inside containment. The containment atmosphere is continuously monitored for airborne gaseous radioactivity. Air flow through the monitor is provided by the suction created by a vacuum pump. Gaseous and N_{13}/F_{18} concentration monitors indicate radiation concentrations in the containment atmosphere.

s are and F_{18} are

The gas channel can respond ~~rapidly~~ ~~to reactor coolant pressure boundary leakage~~. N_{13} ~~is a~~ neutron activation product which ~~is~~ proportional to power levels. ~~Additionally, N_{13} has a~~ relatively short half life and consequently ~~will reach equilibrium rapidly~~. An increase in activity inside containment would therefore indicate a leakage from the reactor coolant pressure boundary. Based on the concentration of N_{13}/F_{18} and the power level, reactor coolant pressure boundary leakage can be estimated.

will detect a 0.5 GPM leak $(N_{13}/F_{18}$

The N_{13}/F_{18} monitoring system ~~has a high sensitivity~~ when the reactor is operating at a power range higher than 20 percent. The N_{13} monitor is seismic Category I. Conformance with the guidance that leak detection should be provided following seismic events that do not require plant shutdown is provided by the seismic Category I classification. Safety-related Class 1E power is not required since loss of power to the radiation monitor is not consistent with continuing operation following an earthquake. ~~Above 20 percent power level, in one hour, a leak less than 0.5 gpm can be detected.~~ Operating experience has indicated the average long-term leakage (from sampling losses, collected leakoffs, and unidentified leakage to the containment) from the reactor coolant system ranges between 0.1 and 0.3 gpm. The N_{13} concentration will increase by at least 25 percent above an existing 0.1 gpm leakage background and almost 10 percent for an existing 0.3 gpm leakage. Both increases are well within the sensitivity of the N_{13}/F_{18} monitor capabilities. 2

Radioactivity concentration indication and alarms for loss of sample flow, high radiation, and loss of indication are provided. Sample collection connections permit sample collection for laboratory analysis. The radiation monitor can be calibrated during power operation.

5.2.5.3.4 Containment Pressure, Temperature and Humidity Monitors

Reactor coolant pressure boundary leakage increases containment pressure, temperature, and humidity, values available to the operator through the plant control system.

At full power, the minimum detectable leak is 0.082 GPM when the radionuclide concentration in containment reaches equilibrium.

The N_{13}/F_{18} monitor can detect a 0.5 GPM leak when the plant is above 20% power and the concentration of radiogas in containment is at equilibrium.

FAX to DINO SCALETTI

March 10, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Don Lindgren
Robin Nydes
Ed Carlin
Earl Novendstern
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR LOFTRAN

This is a background package for the remaining open items for LOFTRAN for your information. LOFTRAN is of interest because by our joint NRC/W schedule, the FSER for this section should be turned into Projects by the middle of April. There are 2 Open Items with NRC Status of Action W. Both of these items still require some Westinghouse action. Thank you.

Jim

Jim Winters
412-374-5290

Selection: [nrc st code]='Action W' And [resp eng] like 'loft*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
3134	NRR/SRXB	21.6.1.7-2	DSER-OI	21.6.1.7-2 Westinghouse needs to identify the information provided in RAI response that will be incorporated into the LOFTRAN final verification and validation (V&V) document (WCAP-14307) or the code applicability document (WCAP-14234). Westinghouse letter NSD-NRC-96-4814, dated 9/5/96, explains the W-action to revise WCAPs 14307 and 14234 to incorporate information from RAI responses. Expected completion is 12/20/96. rkn 11/15/96	LOFTRAN/Novendstern	Action W	Action W		
3222	NRR/SCSB	21.6.5-27	DSER-OI	21.6.5-27 Westinghouse needs to demonstrate the acceptability of LOFTRAN for the calculation of the MSLB mass and energy release for the AP600 DBA evaluation model. The response is being written and is expected by the end of Nov. rkn 11/15/96	Loftran/Carlin,Ed	Action W	Action W		

2 of 2

Jarret

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	3/10/97	NAME:	Steve Kerch
TO:	Jim Bongarra	LOCATION:	Monroeville, PA.
PHONE:	301-415-1046	PHONE:	412-374-5104
COMPANY:	NRC		
LOCATION:	Rockville, Md.	FAX:	(412) 374-5099

Cover + Pages = 1 + 16 17 total

- REMOVE ALL STAPLES
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- PLEASE MAKE COPIES OF TWO-SIDED PAGES

Comments:

Jim,

Attached are the following documents: (1) Human Factors Engineering design description and ITAAC, pages 3.2-1 through 3.2-8; (2) markup of the Minimum inventory design description and ITAAC for the main control room, 4 pages; and (3) markup of the minimum inventory design description and ITAAC for the remote shutdown room, 4 pages. These have been reviewed and approved by my management and are forwarded in advance of the formal copies. We have not yet decided where to place the remote shutdown room ITAAC. We may place it with the Data Display and Processing System design description and ITAAC. You will also notice that a few minor changes have been made to the ITAAC on task analysis as compared to the draft that I faxed you on December 19, 1996. If you have any questions or comments following your review, please call me at 412-374-5104.

Thank You,
Steve Kerch

Phone Number of Receiving Equipment: Jim Bongarra 301-415-2222

3.2 Human Factors Engineering

Design Description

~~The main control room (MCR) provides a facility and resources for the safe control and operation of the plant.~~The AP600 human-system interface (HSI) will be developed and evaluated based upon a human factors engineering (HFE) program. The HSI scope includes the main control room (MCR) and the remote shutdown room (RSR). The HSI scope provides the displays, controls, procedures, and alarms required for normal, abnormal and emergency plant operations. Implementation of the HFE program involves the completion of the following human factors engineering analyses and plans.

1. ~~2.~~ ~~The MCR includes two reactor operator workstations, one senior reactor operator workstation, safety related displays, and safety related controls.~~The integration of human reliability with human factors engineering design is performed in accordance with the implementation plan. Critical human actions (if any) and risk important tasks are identified and used as an input to the task analysis activities.

2. ~~1.~~ ~~The MCR provides a suitable workspace environment for use by MCR operators.~~Task analysis is performed in accordance with the task analysis implementation plan. Task analysis identifies the information and control requirements for the operators to execute the tasks allocated to them.

3. ~~The human system interface (HSI) resources available to the MCR operators include the alarm system, plant information system, computerized procedure system, safety related displays, wall panel information system, and controls (soft and dedicated).~~The HSI design is performed in accordance with the HSI design implementation plan. The HSI design includes the functional design of the operation and control centers and the HSI resources, the specification of design guidelines, the detailed HSI resource design specifications, and the man-in-the-loop concept testing.

4. ~~The MCR and the available HSI permit execution of MCR tasks by MCR operators to operate the plant and maintain plant safety.~~An HFE program verification and validation implementation plan is developed. The plan establishes methods for conducting evaluations of the HSI design.

5. The HFE program verification and validation is performed in accordance with the HFE verification and validation plan and includes implementation of the following activities:

   a. Task support verification
   b. HFE design verification
   c. Integrated system validation
   d. Issue resolution verification
   e. Plant HFE/HSI verification

Inspection, Test, Analyses, and Acceptance Criteria

Table 3.2-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the MCR.

AP600

Table 3.2-1 Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Test, Analyses**	**Acceptance Criteria**
1. ~~2.~~ ~~The MCR includes two reactor operator workstations, one senior reactor operator workstation, safety related displays, and safety related controls.~~The integration of human reliability analysis with human factors engineering design is performed in accordance with the implementation plan.	~~An inspection of the MCR workstations and control panels will be performed.~~An inspection of the documentation associated with the integration of human reliability analysis with human factors engineering design will be performed.	~~The MCR includes two reactor operator workstations, one senior reactor operator workstation, safety related displays, and safety related controls.~~ A report exists and concludes that critical human actions (if any) and risk important tasks were identified and examined by task analysis, and used as input to the HSI design, and procedure development.
2. ~~1.~~ ~~The MCR provides a suitable workspace environment for use by the MCR operators.~~Task analysis is performed in accordance with the task analysis implementation plan.	~~i) See Certified Design Material, subsection 2.7.1, Nuclear Island Non-radioactive Ventilation System.~~ ~~ii) See Certified Design Material, subsection 2.2.5, MCR Emergency Habitability System.~~ ~~iii) See Certified Design Material, subsection 2.6.3, Class 1E dc and UPS System.~~An inspection of the task analysis documentation will be performed	~~i) See Certified Design Material, subsection 2.7.1, Nuclear Island Non-radioactive Ventilation System.~~A report exists and concludes that function based task analyses were conducted in conformance with the task analysis implementation plan and include the following functions: Control reactivity; control RCS boron concentration; control fuel and clad temperature; control RCS coolant temperature, pressure, and inventory; provide RCS flow; control main steam pressure; control SG inventory; control containment pressure and temperature; provide control of main turbine.

Table 3.2-1 (cont) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Test, Analyses**	**Acceptance Criteria**
		~~ii) See Certified Design Material, subsection 2.2.5, MCR Emergency Habitability System~~ A report exists and concludes that operational sequence analyses (OSAs) were conducted in conformance with the task analysis implementation plan. OSAs performed include the following: - plant heatup and startup from post refueling to 100% power; - reactor trip, turbine trip, and safety injection; - natural circulation cooldown (startup feedwater with SG); - loss of reactor or secondary coolant; - post LOCA cooldown and depressurization; - loss of RCS inventory during shutdown; - loss of RNS during shutdown; - manual ADS actuation; - manual reactor trip via PMS, via DAS; - ADS valve testing during mode 1

Table 3.2-1 (cont)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Test, Analyses	Acceptance Criteria
3. ~~The HSI resources available to the MCR operators include the alarm system, plant information system, computerized procedure system, safety related displays, wall panel information system, and controls (soft and dedicated).~~ The HSI design is performed in accordance with the HSI design implementation plan.	~~An inspection of the HSI resources available in the MCR for the MCR operators will be performed.~~ An inspection of the HSI design documentation will be performed.	~~The as-built HSI includes an alarm system, plant information system, computerized procedure system, safety related displays, wall panel information system, and controls (soft and dedicated).~~ A report exists and concludes that the HSI design was conducted in conformance with the implementation plan and includes the following documents: - Operation and Control Centers System Specification Document - Functional requirements and design basis documents for the alarm system, plant information system, computerized procedure system, wall panel information system, soft controls, and the qualified data processing system. - Design guideline documents for the alarm system, plant information system displays, computerized procedure system, and soft control displays. - Design specifications for the alarm system, plant information system displays, computerized procedure system, qualified data processing system displays, wall panel information system displays, and controls (soft and dedicated). - Man-in-the-loop concept test reports.

Table 3.2-1 (cont) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	Inspections, Test, Analyses	Acceptance Criteria
~~4. The MCR and the available HSI permit execution of MCR tasks by MCR operators to operate the plant and maintain plant safety.~~ An HFE program verification and validation implementation plan is developed.	~~Tests and analyses of the following plant evolutions and transients, using a facility that physically represents the MCR configuration and dynamically represents the MCR HSI and the operating characteristics and responses of the AP600 design, will be performed:~~ ~~i) Normal plant heatup and startup to 100% power~~ ~~ii) Normal plant shutdown and cooldown to cold shutdown~~ ~~iii) Transients: reactor trip and turbine trip~~ ~~iv) Accidents:~~ ~~- small break loss of coolant accident~~ ~~- large break loss of coolant accident~~ ~~- steam line break~~ ~~- feedwater line break~~ ~~- steam generator tube rupture~~. An inspection of the HFE verification and validation plan will be performed.	~~The test and analysis results demonstrate that the MCR operators can perform the following:~~ ~~i) Heat up and start up the plant to 100% power~~ ~~ii) Shut down and cool down the plant to cold shutdown~~ ~~iii) Bring the plant to safe shutdown following the specified transients~~ ~~iv) Bring the plant to a safe, stable state following the specified accidents~~A report exists and concludes that the HFE verification and validation plan was developed and includes plans for the following activities: - Task support verification - HFE design verification - Integrated system validation - Issue resolution verification - Plant HFE/HSI verification

Table 3.2-1 (cont) Inspections, Tests, Analyses, and Acceptance Criteria		
Design Commitment	**Inspections, Test, Analyses**	**Acceptance Criteria**
5. The HFE program verification and validation is performed in accordance with the HFE verification and validation plan and includes implementation of the following activities: a. Task support verification b. HFE design verification c. Integrated system validation d. Issue resolution verification e. Plant HFE/HSI verification		A report exists and concludes that:
	a. An inspection of the documentation for the task support verification will be performed.	a. Task support verification was conducted in conformance with the implementation plan and includes verification that the information and controls provided by the HSI matches the display and control requirements generated by the function based task analyses and the operational sequence analyses.
	b. An inspection of the documentation for the HFE design verification will be performed.	b. HFE design verification was conducted in conformance with the implementation plan and includes verification that the HSI design is consistent with the AP600 specific design guidelines developed for each HSI resource.
	c. Tests and analyses of the following plant evolutions and transients, using a facility that physically represents the MCR configuration and dynamically represents the MCR HSI and the operating characteristics and responses of the AP600 design, will be performed: i) Normal plant heatup and startup to 100% power ii) Normal plant shutdown and cooldown to cold shutdown iii) Transients: reactor trip and turbine trip	c. The test and analysis results demonstrate that the MCR operators can perform the following: i) Heat up and start up the plant to 100% power ii) Shut down and cool down the plant to cold shutdown iii) Bring the plant to safe shutdown following the specified transients iv) Bring the plant to a safe, stable state following the specified accidents

Table 3.2-1 (cont)
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Test, Analyses	Acceptance Criteria
	iv) Accidents: - small-break loss-of-coolant accident - large-break loss-of-coolant accident - steam line break - feedwater line break - steam generator tube rupture	
	d. An inspection of the documentation for the HFE design issue resolution verification will be performed.	d. HFE design issue resolution verification was conducted in conformance with the implementation plan and includes verification that human factors issues documented in the design issues tracking system have been addressed in the final design.
	e. An inspection of the plant HFE/HSI design verification documentation will be performed.	e. The plant HFE/HSI is consistent with the HFE/HSI verified in 5a. through 5d.

The controls, displays, and visual alerts listed in Table ______ are retrievable from the remote shutdown workstation.

Table ~~2.5.2-5~~ 9
Minimum Inventory of Displays and ~~Fixed Position~~ Controls

	Description	Control	Display	Alert
insert ① and ②	Neutron Flux	-	Yes	yes
	Reactor Coolant System (RCS) Pressure	-	Yes	yes
	Wide-Range Hot Leg Temperature	-	Yes	
	Wide-Range Cold Leg Temperature	-	Yes	Yes
Insert ③, ④ and ⑤	Containment Water Level	-	Yes	Yes
	Containment Pressure	-	Yes	yes
	Pressurizer Water Level	-	Yes	Yes
Insert ⑥	Pressurizer Reference Leg Temperature	-	Yes	
Insert ⑦	Pressurizer Pressure	-	Yes	
	Core Exit Temperature	-	Yes	Yes
	RCS Subcooling	-	Yes	Yes
Insert ⑧	In-Containment Refueling Water Storage Tank (IRWST) Water Level	-	Yes	Yes
	Passive Residual Heat Removal (PRHR) Flow	-	Yes	Yes
	PRHR Outlet Temperature	-	Yes	Yes
	Passive Containment Cooling System (PCS) Storage Tank Water Level	-	Yes	
	PCS Cooling Flow	-	Yes	
	IRWST to Normal Residual Heat Removal System (RNS) Suction Valve Status	-	Yes	yes
Remotely Operated	Containment Isolation Valve Status(2)	-	Yes	
	Containment Area High-Range Radiation Level	-	Yes	Yes
	Containment Pressure (Extended Range)	-	Yes	[redacted]
	Containment Hydrogen Concentration	-	Yes	
Insert ⑨	Manual Reactor Trip	Yes	-	
	Manual Safeguards Actuation	Yes	-	
	Manual Core Makeup Tank Actuation	Yes	-	

Note: Dash (-) indicates not applicable.

		Control	Display	Alarm(1)
Insert 1:	Neutron Flux Doubling			yes
Insert 2:	Startup Rate		yes	yes
Insert 3:	RCS Cooldown Rate Compared to the Limit Based on RCS Pressure		yes	yes
Insert 4:	Change of RCS temperature by more than 5°F in the last 10 minutes			yes
Insert 6:	Pressurizer Water Level Trend		yes	
Insert 7:	Reactor Vessel - Hot Leg Water Level		yes	yes
Insert 8:	RCS Cold Overpressure Limit		yes	yes
Insert 9:	CMT Level		yes	
Insert 5	Wide-Range Cold Leg Temperature Compared to the Limit Based on RCS Pressure		yes	yes

Table 2.5.2-5 (cont) Minimum Inventory of Displays and Fixed Position Controls			
Manual Description	Control	Display	Alert (1)
Manual Automatic Depressurization System (ADS) Stages 1, 2, and 3 Initiation	Yes	-	
Manual ADS Stage 4 Initiation	Yes	-	
Manual PRHR Actuation	Yes	-	
Manual Containment Cooling Actuation	Yes	-	
Manual IRWST Injection Actuation	Yes	-	
Manual Containment Recirculation Actuation	Yes	-	
Manual Containment Isolation ~~(Selected)~~	Yes	-	
Manual Main Steam Line Isolation	Yes	-	
Manual Feedwater Isolation	Yes	-	
Manual Containment Hydrogen Igniter (Nonsafety-Related)	Yes	-	

Note: Dash (-) indicates not applicable.

(1) These parameters are used to generate visual alerts that identify challenges to the critical safety functions.

(2) These instruments are not required after 24 hours.

7. The PMS, in conjunction with the operator workstations, provides the following functions:

   a) The PMS provides for the minimum inventory of displays, visual alerts, and fixed position controls, as identified in Table 2.5.2-5, in the main control room (MCR).

   b) The PMS provides for the transfer of control capability from the MCR to the remote shutdown room (RSR).

   ~~c) The PMS provides for the minimum inventory of displays and controls, as identified in Table 2.5.2-5, in the RSR. The controls in the RSR do not need to be fixed position.~~

8. a) The PMS automatically removes blocks of reactor trip and engineered safety features actuation when the plant approaches conditions for which the associated function is designed to provide protection. These blocks are identified in Table 2.5.2-6.

   b) The PMS automatically produces a reactor trip or engineered safety feature initiation upon an attempt to bypass more than two channels of a function that uses two-out-of-four initiation logic.

   c) The PMS provides the interlock functions identified in Table 2.5.2-7.

9. Setpoints are determined using a methodology which accounts for loop inaccuracies, response testing, and maintenance or replacement of instrumentation.

10. The PMS hardware and software are verified and validated through a program that provides confirmation that system functional requirements are properly and correctly implemented in the delivered hardware and software.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.5.2-8 specifies the inspections, tests, analyses, and associated acceptance criteria for the PMS.

Table 2.5.2-5 Minimum Inventory of Displays and ~~Fixed Position~~ Controls [9]			
Description	Control	Display	Alert [1]
insert ① and ② Neutron Flux	-	Yes	yes
Reactor Coolant System (RCS) Pressure	-	Yes	yes
Wide-Range Hot Leg Temperature	-	Yes	
Wide-Range Cold Leg Temperature	-	Yes	Yes
Insert ③, ④ and ⑤ Containment Water Level	-	Yes	Yes
Containment Pressure	-	Yes	Yes
Pressurizer Water Level	-	Yes	Yes
Insert ⑥ Pressurizer Reference Leg Temperature	-	Yes	
Insert ⑦ Pressurizer Pressure	-	Yes	
Core Exit Temperature	-	Yes	Yes
RCS Subcooling	-	Yes	Yes
Insert ⑧ In-Containment Refueling Water Storage Tank (IRWST) Water Level	-	Yes	Yes
Passive Residual Heat Removal (PRHR) Flow	-	Yes	Yes
PRHR Outlet Temperature	-	Yes	Yes
Passive Containment Cooling System (PCS) Storage Tank Water Level	-	Yes	
PCS Cooling Flow	-	Yes	
IRWST to Normal Residual Heat Removal System (RNS) Suction Valve Status	-	Yes	yes
Remotely Operated Containment Isolation Valve Status (2)	-	Yes	
Containment Area High-Range Radiation Level	-	Yes	Yes
Containment Pressure (Extended Range)	-	Yes	
Containment Hydrogen Concentration	-	Yes	
Insert ⑨ Manual Reactor Trip	Yes	-	
Manual Safeguards Actuation	Yes	-	
Manual Core Makeup Tank Actuation	Yes	-	

Insert ⑩

Note: Dash (-) indicates not applicable.

		Control	Display	Alarm[1]
Insert ①:	Neutron Flux Doubling			yes
Insert ②:	Startup Rate		yes	yes
Insert ③:	RCS Cooldown Rate Compared to the Limit Based on RCS Pressure		yes	yes
Insert ④:	Change of RCS temperature by more than 5°F in the last 10 minutes			yes
Insert ⑥:	Pressurizer Water Level Trend		yes	
Insert ⑦:	Reactor Vessel - Hot Leg Water Level		yes	yes
Insert ⑧:	RCS Cold Overpressure Limit		yes	yes
Insert ⑨:	CMT Level		yes	
Insert ⑩:	Manual main control room emergency habitability system actuation	yes		
Insert ⑤	Wide-Range Cold Leg Temperature Compared to the Limit Based on RCS Pressure		yes	yes

Table 2.5.2-5 (cont) Minimum Inventory of Displays and Fixed Position Controls			(1)
Description	**Control**	**Display**	Alert
Manual Automatic Depressurization System (ADS) Stages 1, 2, and 3 Initiation	Yes	-	
Manual ADS Stage 4 Initiation	Yes	-	
Manual PRHR Actuation	Yes	-	
Manual Containment Cooling Actuation	Yes	-	
Manual IRWST Injection Actuation	Yes	-	
Manual Containment Recirculation Actuation	Yes	-	
Manual Containment Isolation ~~(Selected)~~ 2	Yes	-	
Manual Main Steam Line Isolation	Yes	-	
Manual Feedwater Isolation	Yes	-	
Manual Containment Hydrogen Igniter (Nonsafety-Related)	Yes	-	

Note: Dash (-) indicates not applicable.

(1) These parameters are used to generate visual alerts that identify challenges to the critical safety functions.

(2) These instruments are not required after 24 hours.

** TX CONFIRMATION REPORT **

AS OF MAR 10 '97 09:03 PAGE.01

WETSO/RM 468 EC EAST

	DATE	TIME	TO/FROM	MODE	MIN/SEC	PGS	CMD#	STATUS
02	03/10	08:53	301 504 2222	G3--S	09'28"	017		OK

FAX to DINO SCALETTI

March 11, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Diane Jackson
Ted Quay

Robin Nydes
Chip Suggs
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #172 (M5.2.5-29)

In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items from the smallest item number on. The relevant information from OITS related to Open Item #172 (M5.2.5-29) is attached. We provided the original comparison to STS with NSD-NRC-96-4833 on October 11, 1996. We then provided probability risk assessment information related to the differences from STS with NSD-NRC-97-4939 on January 14, 1997. This was reiterated in the RAI responses provided by NSD-NRC-97-4972 of February 6, 1997. This item (#172) was asked by a technical branch other than the Tech Spec branch and requests justification specific to a single TechSpec section. The letters identified above were in response to questions asked by the Tech Spec branch and provide general justification for Action Times. Included in the general justifications are specific entries for TS 3.4.9, the subject of this item #172. Please help us provide the branch to branch coordination required to obtain proper review of this information. We believe that the letters identified above resolve the concerns of item #172. We requested your action to change the NRC Status of this item on February 14, 1997. Since NRC should be responsible to review information submitted by Westinghouse, it seems a reasonable request that NRC acknowledge receipt of the information. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

1 of 2

AP600 Open Item Tracking System Database: Executive Summary

Date: 3/11/97

Selection: [item no] between 172 And 172 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
172	NRR/SPLB	5.2.5	MTG-OI	M5.2.5-29 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) STS 3.4.15 states that, should the containment air cooler condensate flow rate monitor become inoperable, a channel check should be performed on the containment atmosphere radioactivity monitor once per 8 hours. The AP600 TS 3.4.9 states that a grab sample should be performed once per 24 hours. Westinghouse should provide justification regarding the acceptability of the alternate action. Action: submit T.S. 3.4.9 with June 96 rev. rkn 3/28 Closed - With issuance of the Tech Specs in SSAR Rev. 9. Action W - Need an explanation of Action Times as they relate to STS. Closed - Applicable information provided in NSD-NRC-96-4833 of 10/11/96, NSD-NRC-97-4939 of 1/14/97 and NSD-NRC-97-4972 of 2/6/97.	TECHSPEC/Suggs, C.	Closed	Action W		

2of2

 Total Records: 1

FAX to DINO SCALETTI

March 11, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Don Lindgren
Mike Corletti
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #164 (M5.2.5-20)

This item should now be becoming an embarrassment. In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items from the smallest item number on. The relevant documentation related to Open Item #164 (M5.2.5-20) is attached. We provided this FAX response on January 10, 1997. We resent the FAX with a request for NRC Status change on February 12, 1997. We believed that this list of references resolved the concerns of item #164 and subsequent telephone conversations. We believe that it is an NRC responsibility to review Westinghouse submittals and it seems a reasonable request that NRC acknowledge receipt of the information provided on references. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 2/12/97

Selection: [item no] between 164 And 164 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
164	NRR/SPLB	5.2.5	MTG-OI		Corletti,M	Closed	Action W		

M5.2.5-20 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) Identify each system connected to the reactor coolant system (RCS) that could experience intersystem leakage and provide a discussion of the leak detection method, including protective features to ensure that the system does not overpressurize.

Closed - Westinghouse has completed necessary submittals to support staff review. See the response for RAI 440.132 for a discussion of this issue.

Action W - per 12/2 telecon, Westinghouse to provide explicit references to where we covered the systems connected to the RCS in the SSAR or other document.
Action N - FAX to Huffman on 1/10/97 provided explicit references.

2 of 3

FAX to BILL HUFFMAN

January 10, 1997

CC: Don Lindgren
Mike Corletti
Brian McIntyre

ADDITIONAL INFORMATION FOR OPEN ITEM 164

This is in response to the 12/2/96 request to provide, explicitly, where we covered leakage from each system connected to RCS in the SSAR or other document. We explicitly cover intersystem leakage from the RCS in WCAP-14425, the ISLOCA report. This WCAP is referenced in the SSAR in a number of places. The most relevant are in section B-63 of SSAR section 1.9, and in subsection 1.9.5.1.7. We believe this completes Westinghouse actions required for Open Item 164 and request NRC direction to change its "NRC Status". We recommend "Action N".

Jim

Jim Winters
412-374-5290

3 of 3

FAX to DINO SCALETTI

March 11, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Don Lindgren
Bruce Rarig
Bob Osterrieder
Earl Novendstern
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEMS FOR NOTRUMP

This is a background package for the remaining open items for NOTRUMP for your information. NOTRUMP is of interest because by our joint NRC/W schedule, the FSER for this section should be turned into Projects by the middle of April. There are 37 Open Items with NRC Status of Action W. Twenty nine (29) of these items still require some Westinghouse action. The remaining 8 (3144, 3147, 3148, 3149, 3150, 3157, 3158 and 3159) have been answered by information in either NSD-NRC-96-4851 of 10/18/96 or NSD-NRC-96-4863 of 10/28/96 (over 4 months ago). It seems a reasonable request that NRC acknowledge receipt of this information. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of these items. We recommend "Action N". Thank you.Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 3/11/97

Selection: [nrc st code]='Action W' And [resp eng] like '*trum*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
2608	NRR/SRXB	15.	RAI-OI	WCAP-14206 (NOTRUMP CAD) 440.335 Page 4-11, item no. 2, "Frictional pressure drops", justification for using constant friction factors, particularly at low flow, low pressure conditions are needed. Please demonstrate that the use of constant friction factors are adequate for simulation of AP600 at low pressure, low flow conditions.	TRUMP/Novendstern/O	Action W	Action W		
2609	NRR/SRXB	15.	RAI-OI	WCAP-14206 (NOTRUMP CAD) 440.336 Page 4-11, item no. 3, "Momentum Equation," momentum flux has been shown for conventional plants to be a second order effect and has been excluded in many small break LOCA analyses. Please describe if momentum flux is included in the AP600 analyses. If not, justification for its omission is also needed.	TRUMP/Novendstern/O	Action W	Action W		
2610	NRR/SRXB	15.	RAI-OI	WCAP-14206 (NOTRUMP CAD) 440.337 Page 4-12, item 4e, "CHF correlation," the use of the Macbeth correlation needs to be justified for low pressure, low pressure conditions. Please demonstrate that the Macbeth correlation is adequate for the low flow and pressure conditions expected for AP600.	TRUMP/Novendstern/O	Action W	Action W		
2611	NRR/SRXB	15.	RAI-OI	WCAP-14206 (NOTRUMP CAD) 440.338 Page 4-13, item 6, "Pump modeling," please demonstrate that the NOTRUMP pump model can predict the AP600 pump coastdown. Also describe and justify the use of the two-phase pump degradation curves for use in AP600 analyses.	TRUMP/Novendstern/O	Action W	Action W		
2612	NRR/SRXB	15.	RAI-OI	WCAP-14206 (NOTRUMP CAD) 440.339 On Page 4-16, item 2, it is stated that since no change to the numerical scheme has been made to NOTRUMP that no noding nor time step studies are needed. The INEL disagrees with this statement. Since the successful performance of the passive safety systems depend on the accurate modeling of the small pressure differences that characterize AP600 phenomenological behavior, node and time step size can affect the magnitude of these small pressure differences driving the flow in the system. Please provide time step and nodalization studies to justify the AP600 nodalization.	TRUMP/Novendstern/O	Action W	Action W		
2615	NRR/SRXB	15.	RAI-OI	WCAP-14206 (NOTRUMP CAD) 440.342 It is customary to provide a calculative methods document separate from the documents describing the code physical model changes, benchmarks, and break spectrum analyses. Since many codes are used to perform a single analysis and in view of the fact that the long term cooling code and containment models have yet to be described, it would be useful to provide a calculative methods report detailing how all of the various codes are interfaced and used to produce the break spectrum analyses. This document should describe the initial conditions and provide sensitivity studies along with justification for the nodalization, models, and assumptions pertaining to AP-600. It should also contain the small break LCOA spectrum analysis. In summary, the NOTRUMP Applicability Document should contain the following information: o A NOTRUMP code section describing all changes to the code to accommodate AP-600 along with model benchmarks, o A description of the containment modeling approach with calculations justifying the model, o A description of the "Long Term Cooling Code" describing the methods, use, and code benchmarks. Also present results of the analyses and describe how the code is interfaced with NOTRUMP, o A section presenting the calculative methods including sensitivity calculations justifying each of the codes comprising the AP-600 small break LOCA analysis package and the full break spectrum analysis, and o A test matrix listing the pertinent separate and integral tests used to benchmark the AP-600 small break LOCA code package.	TRUMP/Novendstern/O	Action W	Action W		

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Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
2922	NRR/SRXB	15.	RAI-OI	NOTRUMP CMT PVR (MT01-GSR-011) 440.441 Were wall temperatures measured in the facility in the CMT and piping? If so, how does the NOTRUMP code compare to these data. Please provide the comparisons and discuss the results.	TRUMP/Novendstern/O	Action W	Action W		
2923	NRR/SRXB	15.	RAI-OI	NOTRUMP CMT PVR (MT01-GSR-011) 440.442 Were wall heat structures modeled in the piping and reservoir? If not, please justify the omission? If so, please describe the model and mesh structure in the all walls where wall heat was simulated.	TRUMP/Novendstern/O	Action W	Action W		
2924	NRR/SRXB	15.	RAI-OI	NOTRUMP CMT PVR (MT01-GSR-011) 440.443 Please confirm and justify the reservoir nodalization? Fig. 3-1 indicates that a single node was modeled. Please justify the nodalization and explain the effects of thermal stratification and mixing, or lack thereof, in the S/W reservoir on the NOTRUMP results.	TRUMP/Novendstern/O	Action W	Action W		
2925	NRR/SRXB	15.	RAI-OI	NOTRUMP CMT PVR (MT01-GSR-011) 440.444 Was a time step study performed for these tests? What time steps were used to simulate these tests? Please discuss and show that the time steps used do not contribute to the error in the NOTRUMP predictions. Are the time steps consistent with those used in the plant model?	TRUMP/Novendstern/O	Action W	Action W		
2927 W	NRR/SRXB	15.	RAI-OI	NOTRUMP CMT PVR (MT01-GSR-011) 440.446 As mentioned in Section 5.0, please summarize the referenced report and briefly explain why the inlet flow uncertainty is higher than the outlet flow uncertainty measurement for the tests. Please explain this uncertainty in light of the NOTRUMP inlet flow rate predictions.	TRUMP/Novendstern/O	Action W	Action W		
3140	NRR/SRXB	21.6.2.2-1	DSER-OI	21.6.2.2-1 Westinghouse needs to identify which information from the NOTRUMP-related RAI response will be formally incorporated into NOTRUMP-related documentation such as the final verification and validation report, the code applicability document (WCAP-14206), or the SSAR.	TRUMP/Novendstern/O	Action W	Action W		
3141	NRR/SRXB	21.6.2.2-2	DSER-OI	21.6.2.2-2 Westinghouse needs to submit the final verification and validation report.	TRUMP/Novendstern/O	Action W	Action W		
3142	NRR/SRXB	21.6.2.4-1	DSER-OI	21.6.2.4-1 Westinghouse needs to explain what provision will be used to ensure that volumetric-based momentum equations will be used for all AP600 calculations.	TRUMP/Novendstern/O	Action W	Action W		
3143	NRR/SRXB	21.6.2.4-2	DSER-OI	21.6.2.4-2 Westinghouse needs to submit the NOTRUMP assessment cases to demonstrate the adequacy of the re-casting of the momentum equation and drift flux equations in net volumetric form.	TRUMP/Novendstern/O	Action W	Action W		

Selection: [nrc st code]='Action W' And [resp eng] like '*trum*' Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
3144	NRR/SRXB	21.6.2.4-3	DSER-OI	21.6.2.4-3 Westinghouse needs to submit the assessment cases to demonstrate the acceptability of modifications to the transient terms in the momentum equation of NOTRUMP. Closed - Response provided via Westinghouse letter NSD-NRC-96-4851 dated October 18, 1996.	TRUMP/Novendstern/M	Closed	Action W		
3145	NRR/SRXB	21.6.2.4-4	DSER-OI	21.6.2.4-4 Westinghouse needs to explain what provision will be used in NOTRUMP to ensure that options to override the default flow partitioning will be used for all AP600 calculations.	TRUMP/Novendstern/O	Action W	Action W		
3146	NRR/SRXB	21.6.2.4-5	DSER-OI	21.6.2.4-5 Westinghouse needs to complete all benchmark and assessment calculations (to be included in the FV&V report) to demonstrate the acceptability of the logic modifications for application of the NOTRUMP code to the AP600 SBLOCA.	TRUMP/Novendstern/O	Action W	Action W		
3147	NRR/SRXB	21.6.2.4-6	DSER-OI	21.6.2.4-6 Westinghouse needs to determine whether to use additional separate effects level swell tests to support the Qualification of the NOTRUMP code to address the code's capability to ppredict two-phase level swell and system mass inventory (see Open Item 21.6.2.6-2). Closed - Response provided via Westinghouse letter NSD-NRC-96-4860 dated October 25, 1996.	TRUMP/Novendstern/O	Closed	Action W		
3148	NRR/SRXB	21.6.2.4-7	DSER-OI	21.6.2.4-7 Westinghouse needs to submit benchmark calculations to demonstrate that the modified pump model is reasonable for application of NOTRUMP to the AP600 SBLOCA. Closed - Response provided via Westinghouse letter NSD-NRC-96-4851 dated October 18, 1996.	TRUMP/Novendstern/O	Closed	Action W		
3149	NRR/SRXB	21.6.2.4-8	DSER-OI	21.6.2.4-8 Westinghouse needs to submit benchmark calculations to demonstrate the acceptability of the changes made to the NOTRUMP gravitational head term, and applicability to the AP600 SBLOCA. Closed - Response provided via Westinghouse letter NSD-NRC-96-4851 dated October 18, 1996.	TRUMP/Novendstern/O	Closed	Action W		
3150	NRR/SRXB	21.6.2.4-9	DSER-OI	21.6.2.4-9 Westinghouse needs to submit benchmark calculations (to be included in the FV&V) to dempnstrate the acceptability of the model changes and additions. Closed - Response provided via Westinghouse letter NSD-NRC-96-4851 dated October 18, 1996.	TRUMP/Novendstern/O	Closed	Action W		
3151	NRR/SRXB	21.6.2.4-10	DSER-OI	21.6.2.4-10 Westinghouse needs to submit benchmark calculations to demonstrate the acceptability of the adequacy of the NOTRUMP birthing logic, and its applicability to the AP600 SBLOCA.	TRUMP/Novendstern/M	Action W	Action W		

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Selection: [nrc st code]='Action W' And [resp eng] like '*trum*' Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
3152	NRR/SRXB	21.6.2.4-11	DSER-OI	21.6.2.4-11 The NOTRUMP FV&V report and assessment calculations need to demonstrate the acceptability of the Zuber critical heat flux correlation for AP600 SBLOCA analysis.	TRUMP/Novendstern,H	Action W	Action W		
3153	NRR/SRXB	21.6.2.4-12	DSER-OI	21.6.2.4-12 The NOTRUMP FV&V report needs to demonstrate the acceptability of the smoothing logic.	TRUMP/Novendstern/M	Action W	Action W		
3154	NRR/SRXB	21.6.2.4-13	DSER-OI	21.6.2.4-13 Westinghouse needs to submit the assessment calculations to demonstrate acceptable logic operation and logic interactions during the FV&V of the AP600 NOTRUMP code.	TRUMP/Novendstern/O	Action W	Action W		
3155	NRR/SRXB	21.6.2.5-1	DSER-OI	21.6.2.5-1 Westinghouse needs to address the models affecting the fluid entering the ADS piping, particularly for the hot legs and pressurizer in the FV&V report.	TRUMP/Novendstern/O	Action W	Action W		
3156	NRR/SRXB	21.6.2.5-2	DSER-OI	21.6.2.5-2 Westinghouse needs to investigate the NOTRUMP code's inability to properly characterize CMT thermal stratification and to better explain some of the differences in CMT discharge flow comparisons.	TRUMP/Novendstern/H	Action W	Action W		
3157	NRR/SRXB	21.6.2.5-3	DSER-OI	21.6.2.5-3 Westinghouse needs to submit its reanalysis of previously analyzed component seperate-effects tests that are listed in Table 21.7 to demonstrate the acceptability of these tests. Closed - Analyses provided via Westinghouse letter NSD-NRC-96-4863, dated October 28, 1996.	TRUMP/Novendstern/O	Closed	Action W		
3158	NRR/SRXB	21.6.2.6-1	DSER-OI	21.6.2.6-1 Westinghouse needs to submit benchmark calculations to demonstrate the acceptability of the NOTRUMP midel changes and additions for which these benchmark calculations are to be performed. Closed - Response provided via Westinghouse letter NSD-NRC-96-4851 dated October 18, 1996.	TRUMP/Novendstern/O	Closed	Action W		
3159	NRR/SRXB	21.6.2.6-2	DSER-OI	21.6.2.6-2 Westinghouse needs to demonstrate the overall adequacy of the seperate effects testing relating to level swell and void fraction distribution in the NOTRUMP code (see Open Item 21.6.2.4-6). Closed - Response provided via Westinghouse letter NSD-NRC-96-4860 dated October 25, 1996.	TRUMP/Novendstern/O	Closed	Action W		
3160	NRR/SRXB	21.6.2.6-3	DSER-OI	21.6.2.6-3 Westinghouse needs to submit reanalysis of the integral systems tests listed in Table 21.10.	TRUMP/Novendstern/O	Action W	Action W		

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Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
3161	NRR/SRXB	21.6.2.7-1	DSER-OI	21.6.2.7-1 Westinghouse needs to submit PRHR primary side heat transfer comparisons between NOTRUMP and OSU/SPEC-2 data in the FV&V report.	TRUMP/Novendstern/K	Action W	Action W		
3162	NRR/SRXB	21.6.2.7-2	DSER-OI	21.6.2.7-2 The NOTRUMP FV&V report needs to address the effects of noncondensible gases on PRHR heat transfer.	TRUMP/Novendstern/H	Action W	Action W		
3229	NRR/SRXB	21.6.2.4-3	DSER-CN	21.6.2.4-3 Westinghouse needs to verify that the NOTRUMP code does not use the Bjornard and Griffith modification.	Trump	Action W	Action W		
3230	NRR/SRXB	21.6.2.4-4	DSER-CN	21.6.2.4-4 Westinghouse needs to verify that heat link methodology for transition boiling is not used in AP600 NOTRUMP calculations.	TRUMP/Novendstern/O	Action W	Action W		
3232	NRR/SRXB	21.6.2.7-1	DSER-CN	21.6.2.7-1 Comparisons of the NOTRUMP code simulations to the OSU and SPES-2 test data in the FV&V report should confirm the applicability or insensitivity of the NOTRUMP flow regimes models to the key system response parameters.	TRUMP/Novendstern/O	Action W	Action W		

96

To: Tom Kenyon

Interlocks

81 - 301 - 415 - 2002 (fax)

Provided below is the draft of a new SSAR section, 7.6.2.3, as a result of our telecon yesterday. Please let me know if this is acceptable to you, Hulbert, et. al. and I'll put it in a letter, and OITS, with a commitment to include it in the next SSAR revision (if W mgt approves this).

Robin Nydes 3/11

7.6.2.3 Interlocks for the Accumulator Isolation Valve and IRWST Discharge Valve

The accumulator isolation and IRWST injection isolation valve operators are nonsafety-related since the valves are not required to change position to mitigate an accident. The SSAR Chapter 15 safety analyses assume that these valves are not subject to valve mispositioning (prior to an accident) or spurious closure (during an accident). Valve mispositioning and spurious closure are prevented by the following:

(dot) The AP600 Technical Specifications, SSAR Section 16.1, require these valves to be open and power locked out whenever these injection paths are required to be available. The accumulators are required to be available when the RCS pressure is above 1000 psia. Both IRWST injection lines are required to be available in Modes 1, 2, 3. One IRWST injection line is required to be available in Mode 4, 5, and in Mode 6 with the reactor upper internals not removed and the refueling cavity not filled.

(dot) The AP600 Technical Specifications, SSAR Section 16.1, require verification that the MOVs are open every 24 hours. They also require verification that power is removed every 31 days.

(dot) With power locked out, redundant (nonsafety-related) valve position indication is provided in the main control room and remote shutdown workstation. Valve position indication and alarm are provided to alert the operator if these valves mispositioned. These indications are powered by different nonsafety-related power supplies.

In addition, the valves have a confirmatory open signal during an accident (S-signal for accumulator MOVs and ADS stage 4 signal for IRWST MOVs). The valves also have an automatic open signal when their close permissive clears during plant startup. The confirmatory open and the automatic open control signals are provided to the valve operator by the nonsafety-related plant control system.

end

FAX to DINO SCALETTI

March 6, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Don Lindgren
Mike Corletti
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #157 (M5.2.5-13)

In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items from the smallest item number on. The relevant documentation related to Open Item #157 (M5.2.5-13) is attached. We provided the attached FAX response on January 9, 1997 (two months ago). We believed that this list of references resolved the concerns of item #157 and subsequent telephone conversations. It seems a reasonable request that NRC acknowledge receipt of the change. We requested this acknowledgement on February 12, 1997 (almost a month ago). We understand that NRC must determine if the information provided is adequate, but this determination itself is an NRC action. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Project Management Summary

Date: 3/6/97

Selection: [item no] between 157 And 157 Sorted by Item #

Item No.	Branch	DSER Section/ Question	Type	Coord/Resp Engineer / Title/Description / Issue Closure Path / Status Detail	Res Est (hrs)	(W) Status	NRC Status	Schedule ICP	Draft	Review	Transmit
157	NRR/SPLB	5.2.5	MTG-OI	Lindgren, D. / Corletti,M.	1	Closed	Action W	1/20/95 A		5/24/95 S	

M5.2.5-13 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) Identify each system that's susceptible to intersystem leakage, discuss the method of leak detection, and protective features.

See the response for RAI 440.132 for a discussion of this issue
DISCUSSED AT 1/25/95 MEETING BETWEEN WESTINGHOUSE AND NRC PLANT SYSTEMS BRANCH
NRC to review RAI 440.132, 210.61, Section 5.4.7, and Section 1.9.5.

Closed - Westinghouse has completed necessary submittals to support staff review.
Action N - NRC to review RAI 440.132, 210.61, Section 5.4.7, and Section 1.9.5.
Action W - per 12/2 telecon, Westinghouse to provide explicit references to where we covered the CVS portion of ISLOCA in the SSAR or other document.
Action N - FAX to Huffman on 1/9/97 provided explicit references.

2 of 3

 Total Records: 1

FAX to BILL HUFFMAN

January 9, 1997

CC: Don Lindgren
Mike Corletti
Brian McIntyre

ADDITIONAL INFORMATION FOR OPEN ITEM 157

This is in response to the 12/2/96 request to provide, explicitly, where we covered the CVS portion of ISLOCA in the SSAR or other document. We explicitly cover the CVS portion of ISLOCA in WCAP-14425, the ISLOCA report. This WCAP is referenced in the SSAR in a number of places. The most relevant to the CVS potion of ISLOCA are in section B-63 of SSAR section 1.9, and in subsection 1.9.5.1.7. We believe this completes Westinghouse actions required for Open Item 157 and request NRC direction to change its "NRC Status". We recommend "Action N".

Jim

Jim Winters
412-374-5290

Westinghouse

FAX COVER SHEET

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	3-12-97	NAME:	Cindy Haag
TO:	Joe Sebrosky / Bill Huffman	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-4277
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 1 + 4

The following pages are being sent from the Westinghouse Energy Center, East Tower, Monroeville, PA. If any problems occur during this transmission, please call:

WIN: 284-5125 (Janice) or Outside: (412)374-5125.

COMMENTS:

Joe / Bill -

Attached are markup page changes to AP600 SSAR subsections 6.3.2.1.2, 6.3.2.2.7.7, and 6.3.2.2.7.9. We are adding a statement to each subsection about valve diversity. This is being done to address PRA insights and providing the place for the disposition within the SSAR. These changes will be included in SSAR Rev. 12 unless we hear differently from you.

Cindy

cc: J. Winters
C. Haag
J. Evans
T. Schulz
D. Lindgren

Although gas accumulation is not expected, there is a vertical pipe stub on the top of the inlet piping high point that serves as a gas collection chamber. Level detectors indicate when gases have collected in this area. There are provisions to allow the operators to open manual valves to locally vent these gases to the in-containment refueling water storage tank.

The passive residual heat removal heat exchanger, in conjunction with the passive containment cooling system, can provide core cooling for an indefinite period of time. After the in-containment refueling water storage tank water reaches its saturation temperature (in about 2 hours), the process of steaming to the containment initiates.

Condensation occurs on the steel containment vessel, which is cooled by the passive containment cooling system. The condensate is collected in a nonsafety-related gutter arrangement located at the operating deck level which returns the condensate to the in-containment refueling water storage tank. The gutter normally drains to the containment sump, but when the passive residual heat removal heat exchanger actuates, a nonsafety-related isolation valve in each of the two gutter drain lines shut and the gutter overflow returns directly to the in-containment refueling water storage tank.

Recovery of the condensate maintains the passive residual heat removal heat exchanger heat sink for a very long time. Without recovery of the condensate, the in-containment refueling water storage tank inventory is sufficient to provide passive residual heat removal heat exchanger operation for 72 hours.

The passive residual heat removal heat exchanger is also used to maintain a safe shutdown condition. It removes decay heat and sensible heat from the reactor coolant system to the in-containment refueling water storage tank, the containment atmosphere, the containment vessel, and finally to the ultimate heat sink -- the atmosphere outside of containment. This occurs after in-containment refueling water storage tank saturation is reached and steaming to containment initiates.

6.3.2.1.2 Reactor Coolant System Emergency Makeup and Boration

The core makeup tanks provide reactor coolant system makeup and boration during events not involving loss of coolant when the normal makeup system is unavailable or insufficient. There are two core makeup tanks located inside the containment at an elevation slightly above the reactor coolant loops. During normal operation, the core makeup tanks are completely full of cold, borated water. The boration capability of these tanks provides adequate core shutdown margin following a steam line break.

The core makeup tanks are connected to the reactor coolant system through a discharge injection line and an inlet pressure balance line connected to a cold leg. The discharge line is blocked by two normally closed, parallel air-operated isolation valves that open on a loss of air pressure or electrical power, or on control signal actuation. Insert (A) here.

- The gravity injection line flow paths from the in-containment refueling water storage tank

- The containment recirculation lines that connect to the gravity injection lines

The check valves selected for these applications incorporate a simple swing-check design with a stainless steel body and hardened valve seats. The passive core cooling system check valves are safety-related, designed with their operating parts contained within the body, and with a low pressure drop across each valve. The valve internals are exposed to low temperature reactor coolant or borated refueling water.

During normal plant operation, these check valves are closed, with essentially no differential pressure across them. Confidence in the check valve operability is provided by operation at no differential pressure clean/cold fluid environment, the simple valve design, and the specified seat materials.

The check valves normally remain closed, except for testing or when called upon to open following an event to initiate passive core cooling system operation. The valves are not subject to the degradation from flow operation or impact loads caused by sudden flow reversal and seating, and they do not experience significant wear of the moving parts.

These check valves are periodically tested during shutdown conditions to demonstrate valve operation. These check valves are equipped with nonintrusive position sensors to indicate when the valves are open or closed.

In current plants, there are many applications of simple swing-check valves that have similar operating conditions to those in the passive core cooling system. The extensive operational history and experience derived from similar check valves used in the safety injection systems of current pressurized water reactors indicate that the design is reliable. Check valve failure to open and common mode failures have not been significant problems.

6.3.2.2.7.7 Accumulator Check Valves

The accumulator check valve design is similar to the accumulator check valves in current pressurized water reactor applications. It is also similar to the low differential pressure opening check valve design described in subsection 6.3.2.2.7.6. Insert ⑤ here.

During normal operation, the check valves are in the closed position with a nominal differential pressure across the disc of about 1550 psid. The valves remain in this position, except for testing or when called upon to open following an event. They are not subject to the degradation from flow operation or impact loads caused by sudden flow reversal and seating. They do not experience significant wear of the moving parts and they are expected to function with minimal backleakage.

In the incontainment refueling water storage tank injection lines, the squib valves are in series with normally closed check valves. In the containment recirculation lines, the squib valves are in series with normally closed check valves in two lines and with normally closed motor operated valves in the other two lines. As a result, inadvertent opening of these squib valves will not result in loss of reactor coolant or in draining of the incontainment refueling water storage tank.

The type of squib valve used in these applications provides zero leakage in both directions. It also allows flow in both directions. A valve open position sensor is provided for these valves. Insert (c) here.

Squib valves are also used to isolate the fourth stage automatic depressurization system lines. These squib valves are in series with normally open motor operated gate valves. Redundant-series controllers are provided to prevent spuriously opening of these squib valves. The type of squib valve used in this application provides zero leakage of reactor coolant out of the reactor coolant system. The reactor coolant pressure acts to open the valve. A valve open position sensor is provided for these valves.

6.3.2.3 Applicable Codes and Classifications

Sections 5.2 and 3.2 list the equipment ASME Code and seismic classification for the passive core cooling system. Most of the piping and components of the passive core cooling system within containment are AP600 Equipment Class A, B, or C and are designed to meet seismic Category I requirements. Some system piping and components that do not perform safety-related functions are nonsafety-related.

The requirements for the control, actuation, and Class 1E devices are presented in Chapters 7 and 8.

6.3.2.4 Material Specifications and Compatibility

Materials used for engineered safety feature components are given in Section 6.1. Materials for passive core cooling system components are selected to the meet the applicable material requirements of the codes in Section 5.2, as well as the following additional requirements:

- Parts of components in contact with borated water are fabricated of, or clad with, austenitic stainless steel or an equivalent corrosion-resistant material.
- Internal parts of components in contact with containment emergency sump solution during recirculation are fabricated of austenitic stainless steel or an equivalent corrosion resistant material.
- Valve seating surfaces are hard-faced to prevent failure and to reduce wear.

INSERT A

The core makeup tank discharge isolation valves are diverse from the passive residual heat removal heat exchanger outlet isolation valves because they use different globe valve body styles and different air operator types.

INSERT B

The accumulator check valves are diverse from the core makeup tank valves because they use different check valve types.

INSERT C

The IRWST injection squib valves are diverse from the containment recirculation squib valves because they are designed to different design pressures.

FAX to DINO SCALETTI

March 12, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Robin Nydes
Chip Suggs
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #173 (M5.2.5-30)

This item is similar to item #172 (yesterday's FAX). The relevant documentation related to Open Item #173 (M5.2.5-30) is attached. We provided the original comparison to STS with NSD-NRC-96-4833 on October 11, 1996. We then provided probability risk assessment information related to the differences from STS with NSD-NRC-97-4939 on January 14, 1997. This was reiterated in the RAI responses provided by NSD-NRC-97-4972 of February 6, 1997. We then asked for a new NRC Status, with a package like this one, on February 17, 1997. This item (#173) was asked by a technical branch other than the Tech Spec branch. The letters identified above were in response to questions asked by the Tech Spec branch and provide general justification for Action Times. Included in the general justifications are specific entries for TS 3.4.5, the subject of this item #173. Please help us provide the branch to branch coordination required to obtain proper review of this information. We believe that the letters identified above resolve the concerns of item #173. Since NRC should be responsible to review information submitted by Westinghouse, it seems a reasonable request that NRC acknowledge receipt of the information related to Open Item #173. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 3/12/97

Selection: [item no] between 173 And 173 Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
173	NRR/SPLB	5.2.5	MTG-OI		TECHSPEC/Suggs, C.	Closed	Action W		

M5.2.5-30 (REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE) STS 3.4.15 includes SR 3.4.15.2, which states that a channel operational test (COT) should be performed on the containment atmosphere radioactivity monitor every 31 days. AP600 TS 3.4.5 includes SR 3.4.9.2 which states that the COT should be performed every 92 days. Westinghouse should provide justification for the deviation from STSs.

Action W: justification of differences between AP600 TS and STS will be provided with TS. rkn 3/29
Closed - With issuance of the Tech Specs in SSAR Rev. 9.
Action W - Need an explanation of Action Times as they relate to STS.
Closed - With issuance of letters NSD-NRC-96-4833 (10-11-96) which explains differences between STS and AP600 TS, NSD-NRC-97-4939 (1-14-97) which provides the response to RAI 630.10 for PRA support of TS, and NSD-NRC-97-4972 (2-6-97) which responds to RAIs 630.11-14 regarding the basis for completion times and surveillance frequencies. rkn 2/24/97

287

Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-96-4833
DCP/NRC0616
Docket No.: STN-52-003

October 11, 1996

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

ATTENTION: T. R. QUAY

SUBJECT: CLOSING THE LAST DSER OPEN ITEM FOR AP600 SSAR SECTION 16.1, TECHNICAL SPECIFICATIONS (TS)

Dear Mr. Quay:

This letter is written to close the last DSER open item for AP600 SSAR Section 16.1, Technical Specifications (TS). Westinghouse committed to provide written explanation of technical differences between the AP600 TS and those presented in NUREG-1431, the Standard TS (STS). Attached are:

1. A roadmap which identifies the sections comprising the STS versus those included in the AP600 TS. For any TS that are included in the STS but not in the AP600 TS, an explanation is provided. For any TS that are included in the AP600 TS but not in the STS, those sections are shaded in the roadmap and explained. Explanations are also provided for other content differences between the STS and AP600 TS.

2. A description of general or overall changes whose explanations apply to multiple TS.

3. A list of technical differences between the STS and AP600 TS. The TS and BASES are grouped by section and an explanation of each difference is provided.

4. A table of and explanation for those LCOs whose endpoint is defined as MODE 4 for the AP600, rather than MODE 5 or "Go to LCO 3.0.3" per the STS.

Discussions regarding ties between the AP600 PRA and the Technical Specifications will be provided in the response to RAI 630.10.

2943A

3 of 7

This submittal closes Open Item Tracking System (OITS) item 2353, which is the final open item for the AP600 Technical Specifications. If you have any questions regarding this transmittal, please contact Robin K. Nydes at (412) 374-4125.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/nja

Attachment

cc: W. Huffman, NRC
A. Chu, NRC
C. Grimes, NRC
N. Liparulo, Westinghouse (w/o Attachments)

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-4939
DCP/NRC0705
Docket No.: STN-52-003

January 14, 1997

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

ATTENTION: T. R. QUAY

SUBJECT: WESTINGHOUSE RESPONSE TO RAI 630.10

Dear Mr. Quay:

Enclosed are three copies of the Westinghouse response to RAI 630.10 regarding AP600 Technical Specification deviations from NUREG-1431 based on probability risk assessment. The NRC technical staff should review this response as part of their review of the AP600 Technical Specifications. This closes DSER open item tracking system item #3054. If there are any questions regarding this transmittal, please contact Robin K. Nydes at (412) 374-4125.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/jml

enclosure

cc: Angela Chu, NRC - (w/enclosure)
W. C. Huffman, NRC - (w/enclosure)
Nicholas Liparulo, Westinghouse - (w/o enclosure)

3047A

sd7

AP600

Question 630.10. Provide a list of proposed AP600 Technical Specification requirements that deviate from NUREG-1431 based either totally or partially on probabilistic risk assessment (PRA) or PRA insights.

Response: The deviations from NUREG-1431 are explained in Reference 1. There are no AP600 Technical Specifications which deviate from NUREG-1431 with the PRA as the basis.

However, selection of a standardized Completion Time or Surveillance Frequency considers available PRA results as described in Reference 2. Per NRC request, attached is a list comparing the NUREG-1431 Standardized Technical Specification (STS) completion times and surveillance frequencies to the AP600 TSs. Deviations from STS times which are less restrictive than STS times are highlighted and any PRA relationship is given in the comment column.

SEE ATTACHED LIST

SSAR Revision: NONE

References:

1. NSD-NRC-96-4833, Closing the Last DSER Open Item for AP600 SSAR Section 16.1, Technical Specifications (TS), 10/11/96.

2. NSD-NRC-96-4699, Westinghouse AP600 Technical Specifications Approach, 5/3/96.

Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

NSD-NRC-97-4972
DCP/NRC0732
Docket No.: STN-52-003

February 6, 1997

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

TO: T. R. QUAY

SUBJECT: RESPONSE TO RAIs 630.11 THROUGH 630.14

REFERENCE: LETTER FROM NRC TO WESTINGHOUSE (HUFFMAN TO LIPARULO), "REQUEST FOR ADDITIONAL INFORMATION ON WESTINGHOUSE AP600 TECHNICAL SPECIFICATIONS OPTIMIZATION METHODOLOGY", DATED DECEMBER 12, 1996.

Enclosed for NRC review are the Westinghouse responses to the following Technical Specification RAIs, provided by the above Reference.

630.11	Completion Time Anchor Point
630.12	Surveillance Frequency Baseline
630.13	Request for Response to RAI 630.10
630.14	Differences Between the Proposed Tech Specs Approach and Tech Specs Rev. 2

This completes Westinghouse activity for Open Item Tracking System items 4224 through 4227, a report for which is attached. Please advise as to the NRC status for these items. If you have any questions regarding this transmittal, please contact Robin K. Nydes (412) 374-4125.

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/jml
enclosure
attachment

cc: W. Huffman, NRC (w/enclosure/attachment)
A. Chu, NRC (w/enclosure/attachment)

3075A

7d1

FAX to DINO SCALETTI

March 12, 1997

CC: Sharon or Dino, please make copies for: Ted Quay
Bill Huffman
Diane Jackson
Tom Kenyon
Joe Sebrosky

Cindy Haag
Don Lindgren
Robin Nydes
Brian McIntyre
Ed Cummins
Bob Vijuk

This is a reminder list of the Open Items where we have recently provided background documentation showing the difference between "W Status" and "NRC Status". In all cases, we believe the next action is with NRC and await your definitization of a Westinghouse action or your direction to change the "NRC Status" to something other than "Action W". Note that we have received no information from NRC on items on this list for over a week. **Note that submittal dates over a year old and request dates over a month old are in bold type.**

Open Item Number	Westinghouse Submittal	Request for Status Change
142 (M3.11-9)	**2/29/96**	**2/3/97** 3/4/97
157 (M5.2.5-13)	1/9/97	**2/12/97** 3/6/97
164 (M5.2.5-20)	1/10/97	**2/12/97** 3/11/97
172 (M5.2.5-29)	1/14/97	2/14/97 3/11/97
173 (M5.2.5-30)	1/14/97	2/17/97 3/12/97
182 (M5.4.11-5)	1/10/97	2/20/97
184 (M5.4.11-7)	1/13/97	2/20/97
405	7/8/96	**2/11/97**
556 (DSER 2.5.4.8-1)	12/20/96	3/10/97
681 (DSER 3.8.2.4-3)	2/11/97	2/17/97
706 (DSER 3.8.2.4-28)	2/11/97	2/17/97

Open Item Number	Westinghouse Submittal	Request for Status Change
710 (DSER 3 8.3.1-1)	1/16/97	2/18/97
716 (DSER 3.8.3.2-5)	1/16/97	2/18/97
717 (DSER 3.8.3.3-1)	1/16/97	2/18/97
718 (DSER 3.8.3.3-2)	1/16/97	2/18/97
722 (DSER 3.8.3.4-3)	1/16/97	2/18/97
724 (DSER 3.8.3.4-5)	1/16/97	2/18/97
729 (DSER 3.8.3.4-10)	1/16/97	2/18/97
731 (DSER 3.8.3.4-12)	1/16/97	2/18/97
740 (DSER 3.8.4.1-3)	1/16/97	2/20/97
754 (DSER 3.8.4.4-6)	1/16/97	2/20/97
757 (DSER 3.8.4.5-1)	1/16/97	2/20/97
758 (DSER 3.8.4.5-2)	1/16/97	2/20/97
782 (DSER 3.9.2.3-1)	**6/30/95**	2/28/97
783 (DSER 3.9.2.3-2)	2/19/97	2/28/97
786 (DSER 3.9.3.1-1)	**2/29/96**	2/28/97
793 (DSER 3.9.3.3-2)	2/19/97	2/28/97
801 (DSER 3.9.6.2-4)	2/19/97	2/28/97
802 (DSER 3.9.6.2-5)	2/19/97	2/28/97
805 (DSER 3.9.6.2-8)	2/19/97	2/28/97
807 (DSER 3.9.6.3-1)	2/19/97	2/28/97
809 (DSER 3.9.6.4-2)	2/19/97	2/28/97
854 (DSER 4.2.8-1)	4/12/96	3/4/97
1172 (DSER 11.2-6)	2/21/97	3/4/97
1210 (DSER 12.4.2-2)	4/30/96	**2/6/97**
1227	7/8/96	**2/11/97**
1228	7/8/96	**2/11/97**
1231	7/8/96	**2/11/97**
1232	7/8/96	**2/11/97**
1716	12/17/96	2/28/97

Open Item Number	Westinghouse Submittal	Request for Status Change
1727	12/17/96	2/28/97
1730	2/19/97	2/28/97
1731	2/19/97	2/28/97
1736	2/19/97	2/28/97
1740	10/11/96	2/28/97
1742	12/17/96	2/28/97
1745	12/17/96	2/28/97
1747	12/17/96	2/28/97
1753	12/17/96	2/28/97
1760	12/17/96	2/28/97
1792 (DSER-CN 3.9.2.1-4)	10/23/96	2/28/97
1793 (DSER-CN 3.9.2.3-1)	10/23/96	2/28/97
1797 (DSER-CN 3.9.2.4-4)	10/14/96	2/28/97
1802 (DSER-CN 3.9.3.3-3)	9/5/96	2/28/97
1803 (DSER-CN 3.9.3.3-4)	9/5/96	2/28/97
1807 (DSER-CN 3.9.7-1)	6/19/96	2/28/97
1888 (DSER-COL 3.8.2.4-1)	2/11/97	2/17/97
2034 (DSER-OI50 13.)	7/8/96	**2/11/97**
2066	12/17/96	2/28/97
2347	1/16/97	2/18/97
2348	1/16/97	2/18/97
2349	1/16/97	2/18/97
3057	5/30/96	2/18/97
3144 (DSER 21.6.2.4-3)	10/18/96	3/11/97
3147 (DSER 21.6.2.4-6)	10/25/96	3/11/97
3148 (DSER 21.6.2.4-7)	10/18/96	3/11/97
3149 (DSER 21.6.2.4-8)	10/18/96	3/11/97
3150 (DSER 21.6.2.4-9)	10/18/96	3/11/97
3157 (DSER 21.6.2.5-3)	10/25/96	3/11/97

Open Item Number	Westinghouse Submittal	Request for Status Change
3158 (DSER 21.6.2.6-1)	10/18/96	3/11/97
3159 (DSER 21.6.2.6-2)	10/25/96	3/11/97
3247 (RAI 230.98)	4/30/96	2/18/97
3372 (RAI 210.213)	1/8/97	2/28/97
4123 (RAI 480.440)	12/13/96	3/10/97
4124 (RAI 480.441)	12/13/96	3/10/97
4125 (RAI 480.442)	12/13/96	3/10/97
4126 (RAI 480.443)	12/13/96	3/10/97
4127 (RAI 480.444)	12/13/96	3/10/97
4128 (RAI 480.445)	12/13/96	3/10/97
4129 (RAI 480.446)	12/13/96	3/10/97
4130 (RAI 480.447)	12/13/96	3/10/97
4131 (RAI 480.448)	12/13/96	3/10/97
4132 (RAI 480.449)	12/13/96	3/10/97
4133 (RAI 480.450)	12/13/96	3/10/97
4134 (RAI 480.451)	12/13/96	3/10/97
4135 (RAI 480.452)	12/13/96	3/10/97
4136 (RAI 480.453)	12/13/96	3/10/97
4137 (RAI 480.454)	12/13/96	3/10/97
4138 (RAI 480.455)	12/13/96	3/10/97
4139 (RAI 480.456)	12/13/96	3/10/97
4140 (RAI 480.457)	12/13/96	3/10/97
4141 (RAI 480.458)	12/13/96	3/10/97
4142 (RAI 480.459)	12/13/96	3/10/97
4143 (RAI 480.460)	12/13/96	3/10/97
4998	2/19/97	2/28/97
4999	2/19/97	2/28/97
5001	2/19/97	2/28/97
5002	2/19/97	2/28/97

Open Item Number	Westinghouse Submittal	Request for Status Change

Note that the status was changed for a large number of items so they have been removed from the table.

Thanks for your help.

Jim

Jim Winters

Westinghouse

RECIPIENT INFORMATION		SENDER INFORMATION	
DATE:	MARCH 13, 1997	NAME:	JIM WINTERS
TO:	TOM KENYON	LOCATION:	ENERGY CENTER - EAST
PHONE:	FACSIMILE:	PHONE:	Office: 412-374-5290
COMPANY:	USNRC	Facsimile:	win: 284-4887 outside: (412)374-4887
LOCATION:			

Cover + Pages 4

The following pages are being sent from the Westinghouse Energy Center, East Tower, Monroeville, PA. If any problems occur during this transmission, please call:

WIN: 284-5125 (Janice) or Outside: (412)374-5125.

COMMENTS:

TOM

HERE IS REWRITE OF RESPONSE FOR RAI 471.23 (OITS #20). IT WILL BE FORMALLY ISSUED UNLESS WE HEAR FROM YOU. THE SSAR REVISIONS INCLUDED HERE WILL BE IN REVISION 12. CALL WITH QUESTIONS.

Jim Winters

CC: LINDGREN
WINTERS
ISRAELSON
SEJVAR
MCINTYRE
JEANNE EVANS
TUPPER

Question 471.23 Revision 1

Verify that the airborne radiation monitors described in Section 11.5.2.3.2 of Chapter 11 of the SSAR will be sensitive enough to detect 10 DAC-hrs in any area of the plant that can be accessed by plant personnel.

Response:

Five (5) airborne radiation monitors are described in Subsection 11.5.2.3.2. An additional six (6) monitors for areas than can be accessed by plant personnel are described in Subsection 11.5.2.3.1. These radiation monitors are part of the permanently installed AP600 radiation monitoring system and provide general area monitoring. These radiation monitors are supplemented by local portable continuous air monitors (CAMs). CAM use is directed by the Health Physics staff during maintenance operations with a high potential for airborne radioactivity levels.

The eleven (11) radiation monitors mentioned above monitor selected areas of the plant that can be accessed by plant personnel. These selected areas are as follows:

1) Fuel Handling Area
2) Auxiliary Building
3) Annex Building
4) Main Control Room and Technical Support Center
5) Containment
6) Health Physics and Hot Machine Shop
7) Radwaste Building

Areas 1, 2, 3, 6, and 7 are monitored by measuring the concentration of radioactive materials in the exhaust air from each area.

Area 4 is monitored by measuring the concentration of radioactive materials in the supply air.

Area 5 is monitored by three separate airborne process monitors:

1) The Containment Air Filtration Exhaust Radiation Monitor measures the concentration of radioactive materials in the containment purge exhaust air.

2) The Containment Atmosphere Radiation Radiogas Monitor measures the radiation from the radioactive gases in the containment atmosphere.

3) The Containment Atmosphere Radiation N^{13}/F^{18} Monitor measures the concentration of radioactive airborne gaseous contamination inside the containment as an indication of reactor coolant pressure boundary leakage.

These eleven (11) monitors are sensitive enough to detect 10 DAC-hours as discussed below.

SSAR Table 11.5-1 provides a listing of each detector, the principal isotope(s) it monitors, and the detector's nominal range. The lower value of the detector's nominal range corresponds to the detector's minimum detectable level. These minimum detectable levels are achieved with a 95% confidence level at standard operating conditions. The following table summarizes Table 11.5-1 and includes the DAC occupational values from Table 1, Column 3, of Appendix B (Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage) of 10 CFR 20.

Airborne Process Radiation Monitor	Isotope(s)	Detector Minimum Detectable Level	DAC Occupational Values - 10 CFR 20, Appendix B, Table 1
Fuel Handling Area Exhaust	Kr-85 Xe-133	1.0E-6 μCi/cc 1.0E-6 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc
Auxiliary Building Exhaust	Kr-85 Xe-133	1.0E-6 μCi/cc 1.0E-6 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc
Annex Building Exhaust	Kr-85 Xe-133	1.0E-6 μCi/cc 1.0E-6 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc
MCR Supply Air Duct Particulate	Sr-90 Cs-137	1.0E-12 μCi/cc 1.0E-12 μCi/cc	8.0E-9 μCi/cc 6.0E-8 μCi/cc
MCR Supply Air Duct Iodine	I-131	1.0E-11 μCi/cc	2.0E-8 μCi/cc
MCR Supply Air Duct Gas	Kr-85 Xe-133	1.0E-7 μCi/cc 1.0E-7 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc
Containment Air Filtration Exhaust	Kr-85 Xe-133	1.0E-6 μCi/cc 1.0E-6 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc
Health Physics and Hot Machine Shop Exhaust	Sr-90 Cs-137	1.0E-13 μCi/cc 1.0E-13 μCi/cc	8.0E-9 μCi/cc 6.0E-8 μCi/cc
Radwaste Building Exhaust	Sr-90 Cs-137	1.0E-13 μCi/cc 1.0E-13 μCi/cc	8.0E-9 μCi/cc 6.0E-8 μCi/cc
Containment Atmosphere N^{13}/F^{18}	N-13 F-18	1.0E-7 μCi/cc 1.0E-7 μCi/cc	N/A 3.0E-5 μCi/cc
Containment Atmosphere Gas	Kr-85 Xe-133	1.0E-7 μCi/cc 1.0E-7 μCi/cc	1.0E-4 μCi/cc 1.0E-4 μCi/cc

The above table shows that for each principal isotope, the minimum detectable level for each monitors detector(s) is almost two (2) to almost five (5) orders of magnitude below the corresponding 10 CFR 20 DAC occupational value.

These radiation monitors utilize two basic types of detectors, as described in Section 11.5.2.3.2. The particulate (Sr-90/Cs-137) and iodine detectors use shielded fixed filters, located in the sample stream, that are viewed by beta and gamma sensitive scintillators, respectively. The radiogas detectors use beta sensitive scintillators with their sensitive volumes directly exposed to the process or sample stream.

The response time for each fixed filter detector depends upon background radiation levels, airborne radioactivity levels, sample flow rate, and system configuration. When the detectors have achieved statistically accurate operating conditions, the detector response times are as follows:

1) Step change in radioactivity levels above the ALERT setpoint - < 4 seconds, not including sample transport time.

2) Gradually increasing radioactivity levels above the ALERT setpoint - < 2 seconds, not including sample transport time.

The step change requires a longer response time to assure that the change is not a spurious radioactivity spike. The time to achieve statistical accuracy (95% confidence level) can vary from ten minutes to one hour, depending upon radioactivity concentrations. The only time the detectors will not be operating under statistically accurate conditions will be the time following a filter change or a system shutdown for maintenance. Sample transport times are minimized by locating the detectors as close as practicable to the process sample point.

The response time for the in-line detectors is less than ten seconds. These detectors are provided with dynamic background radiation compensation.

Combining the minimum detectable levels shown in the table above with the detector response times discussed above, it has been shown that each monitor is sensitive enough to detect 10 DAC-hours.

SSAR Revision:

In Table 11.5-1, "Radiation Monitor Detector Parameters", add the following:

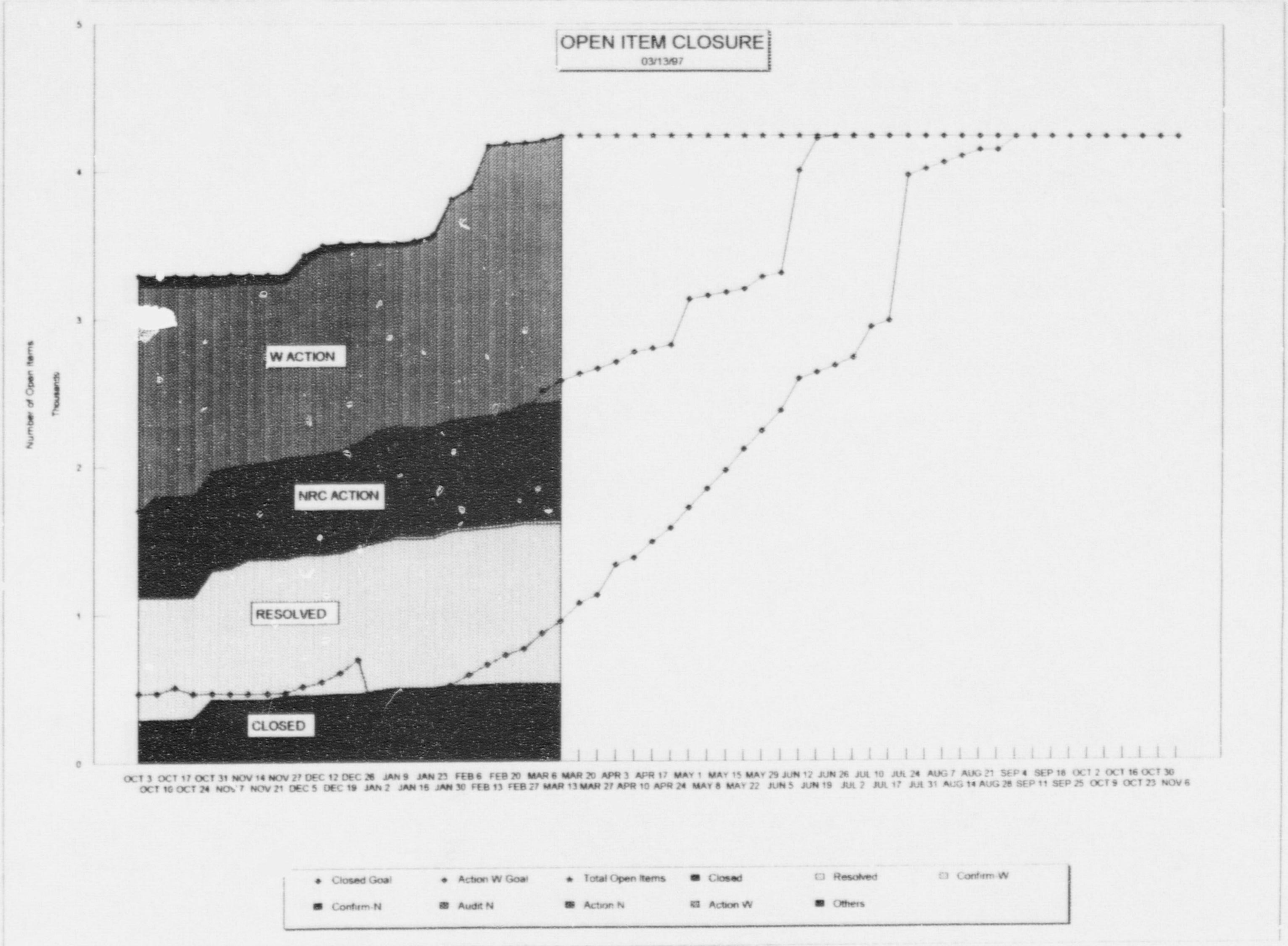
- Add "(Note 5)" in the "Service column for:

  Containment Atmosphere Gas
  Containment Atmosphere N^{13}/F^{18}
  Fuel Handling Area Exhaust
  Auxiliary Building Exhaust

Annex Building Exhaust
Main Control Room Supply Air Duct (Particulate) - both entries
MCR Supply Air Duct (Iodine) - both entries
MCR Supply Air Duct (Gas) - both entries
Containment Air Filtration Exhaust
H.P. & Hot Machine Shop Exhaust
Radwaste Building Exhaust

- In "Notes:" add:

"5. Monitor is sensitive enough to detect 10 Derived Air Concentration (DAC)-hours."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03/13/97
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Thousands
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2
3
4
5
W ACTION
NRC ACTION
RESOLVED
CLOSED
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OCT 10 OCT 24 NOV 7 NOV 21 DEC 5 DEC 19 JAN 2 JAN 16 JAN 30 FEB 13 FEB 27 MAR 13 MAR 27 APR 10 APR 24 MAY 8 MAY 22 JUN 5 JUN 19 JUL 2 JUL 17 JUL 31 AUG 14 AUG 28 SEP 11 SEP 25 OCT 9 OCT 23 NOV 6
Closed Goal
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Audit N
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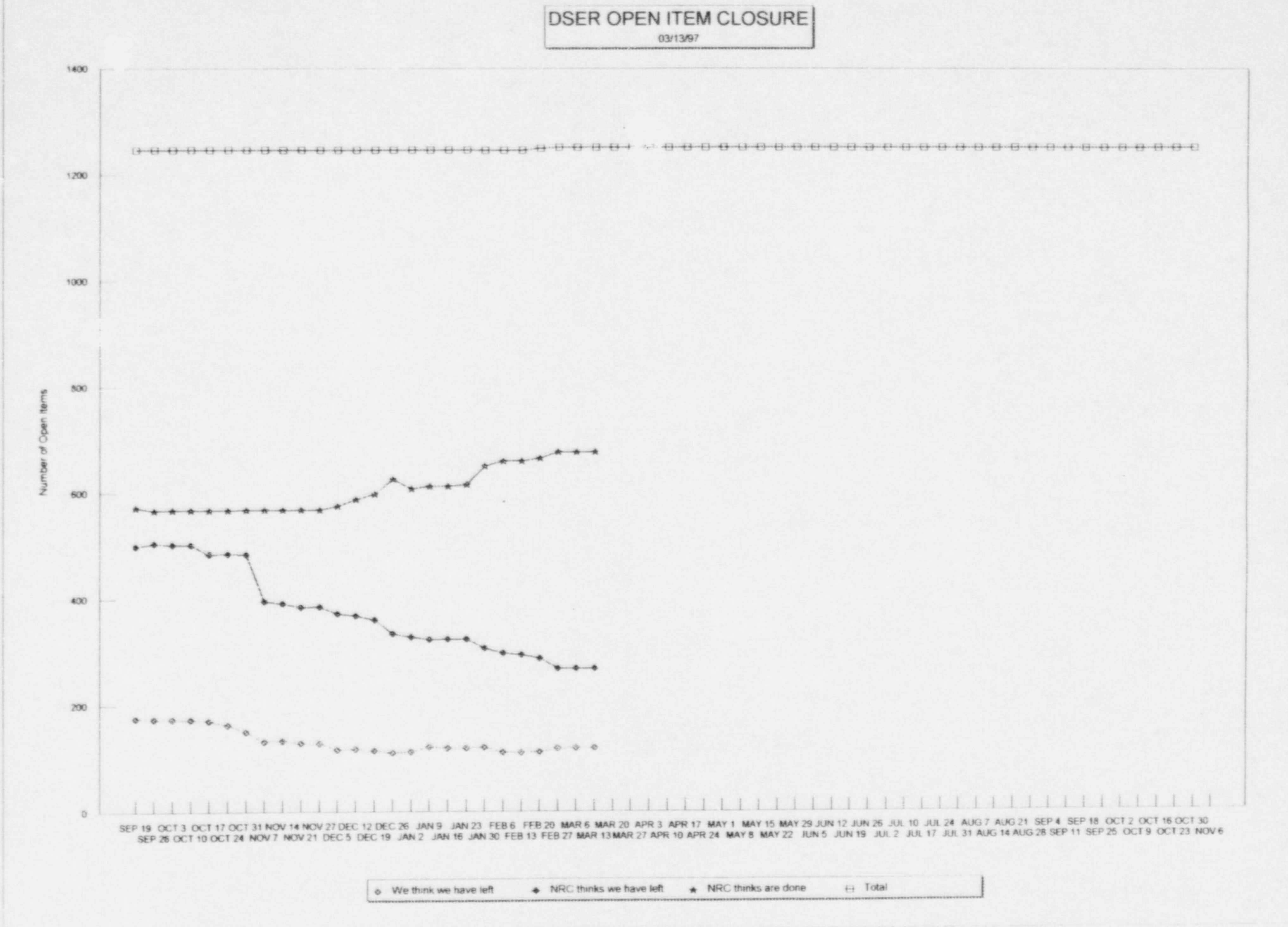
Action W
Others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Number of Open Items
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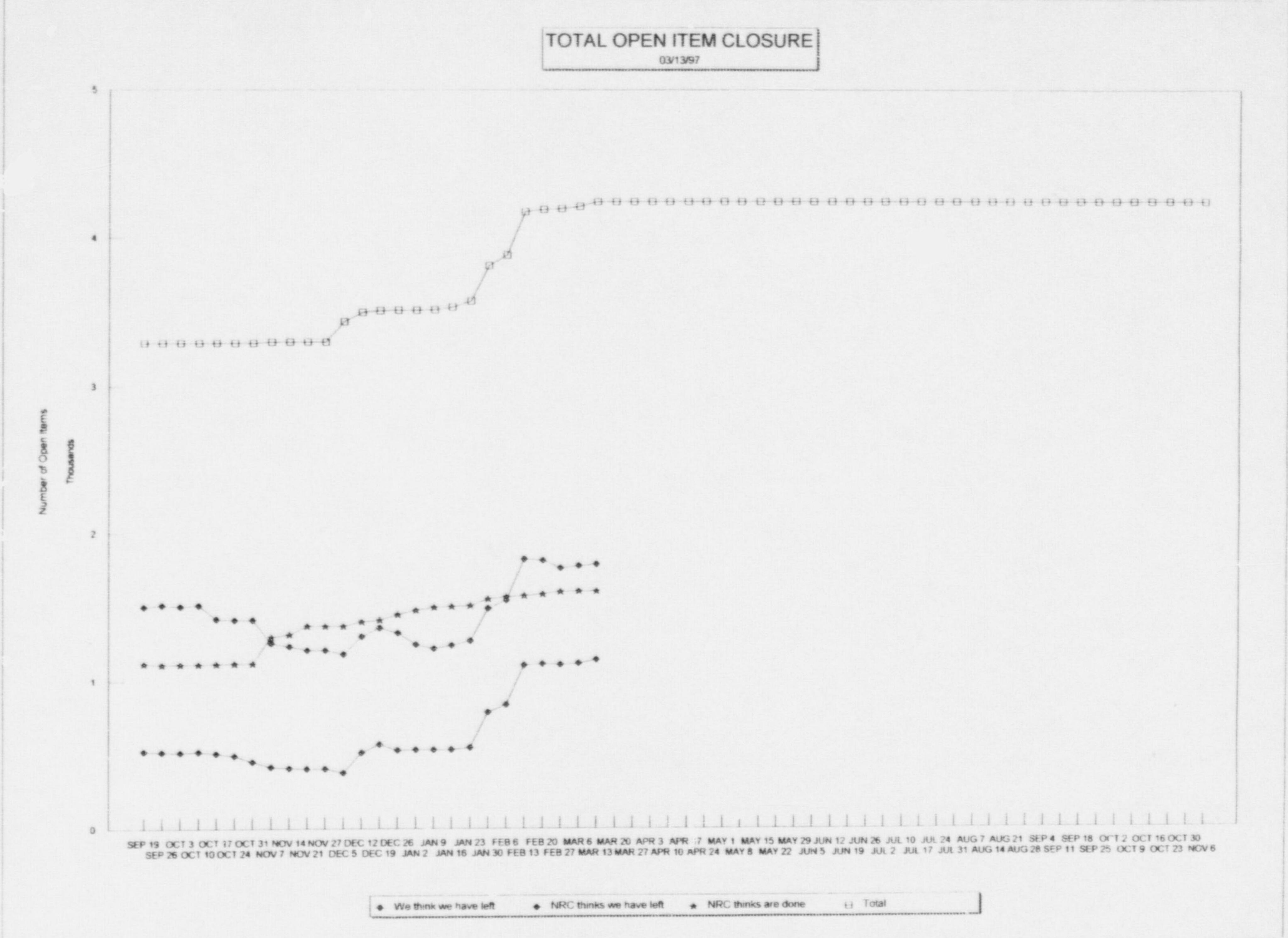
We think we have left
NRC thinks we have left
NRC thinks are done
Total


TOTAL OPEN ITEM CLOSURE
03/13/97
Number of Open Items
Thousands
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1
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SEP 19 OCT 3 OCT 17 OCT 31 NOV 14 NOV 27 DEC 12 DEC 26 JAN 9 JAN 23 FEB 6 FEB 20 MAR 6 MAR 20 APR 3 APR 17 MAY 1 MAY 15 MAY 29 JUN 12 JUN 26 JUL 10 JUL 24 AUG 7 AUG 21 SEP 4 SEP 18 OCT 2 OCT 16 OCT 30
SEP 26 OCT 10 OCT 24 NOV 7 NOV 21 DEC 5 DEC 19 JAN 2 JAN 16 JAN 30 FEB 13 FEB 27 MAR 13 MAR 27 APR 10 APR 24 MAY 8 MAY 22 JUN 5 JUN 19 JUL 2 JUL 17 JUL 31 AUG 14 AUG 28 SEP 11 SEP 25 OCT 9 OCT 23 NOV 6
We think we have left
NRC thinks we have left
NRC thinks are done
Total

Winters, James

From: Dino Scaletti
To: Winters, James
Subject: AP600 OITS
Date: Wednesday, February 26, 1997 3:38PM

Jim,

Some of the following may have been already addressed as action NRC.

Move Items 1809, 1810 and 1811 to "Action N"
Move Items 3264, 3265, 3266, 3267, and 3268 to "Action N"
Move Items 3269, 3270, and 3271 to "Action N"
Move the 15 Items referenced in your 2/14/97 fax to "Action N"
Move #5 (RAI 410.263) to "Action N"
Move 1883, 2430, 2431, 2432, and 3518 to "Action N"

Dino

FAX to DINO SCALETTI

March 14, 1997

CC: Sharon or Dino, please make copies for: Bill Huffman
Ted Quay

Don Lindgren
Ron Vijuk
Terry Schulz
Mike Corletti
Ed Cummins
Bob Vijuk
Brian McIntyre

OPEN ITEM #182 (M5.4.11-5)

In my quest to make sure we have provided NRC with everything needed to prepare an FSER, I am researching open items from the smallest item number on. The relevant documentation related to Open Item #182 (M5.4.11-5) is attached. We provided the original fax of a markup on January 10, 1997 (over two months ago). We requested an NRC Status change ad resent the markup on February 20, 1997 (almost a month ago). We included the changes indicated on the January fax in Revision 11 of the SSAR on February 28, 1997 (two weeks ago). Although we understand that NRC may need to review the IRWST design for ADS actuation, we believe that the information provided is sufficient to resolve the concerns of item #182. It seems a reasonable request that NRC acknowledge receipt of the information. We request that NRC provide a definitive action for Westinghouse or provide direction to change the status of this item. We recommend "Action N". Thank you.

Jim

Jim Winters
412-374-5290

AP600 Open Item Tracking System Database: Executive Summary

Date: 3/14/97

Selection: [item no] between 182 And 182 Sorted by Item #

Item No	Branch	DSER Section/ Question	Type	Title/Description Detail Status	Resp Engineer	(W) Status	NRC Status	Letter No. /	Date
182	NRR/SPLB	5.4.1?	MTG-OI		Corletti,M.	Closed	Action W		

M5.4.11-5 (PRESSURIZER RELIEF DISCHARGE) Section 5.4.11.3 states that the IRWST is sized based on the heat load and steam volume following an actuation of the ADS. Does this include steam, water, and noncondensable gases from all three ADS stages? Provide the analysis.

Closed - See Section 6.3 for a discussion of the IRWST during accuation of the automatic depressurization system. DISCUSSED AT 1/25/95 MEETING BETWEEN WESTINGHOUSE AND NRC PLANT SYSTEMS BRANCH

Closed - NRC to review following Westinghouse providing of specific SSAR reference. Specific reference provided by fax markup of SSAR on January 10, 1997. Specific reference included in SSAR Revision 11 of 2/28/97. jww

2.85

 Total Records: 1

5.4.10.3 Design Evaluation

An evaluation verifies the design adequacy and structural integrity of the reactor coolant loop and the primary equipment supports system. This evaluation compares the analytical results with established criteria for acceptability. Structural analyses demonstrate design adequacy for safety and reliability of the plant in case of a seismic disturbance, and/or loss of coolant accident conditions. Loads that the system is expected to encounter during its lifetime (thermal, weight, and pressure) are applied, and stresses are compared to allowable values. Subsection 3.9.3 discusses the modeling and analysis methods.

5.4.10.4 Tests and Inspections

Nondestructive examinations are performed according to the procedures of the ASME Code, Section V, except as modified by the ASME Code, Section III, Subsection NF.

5.4.11 Pressurizer Relief Discharge

The AP600 does not have a pressurizer relief discharge system. The AP600 has neither power operated pressurizer relief valves nor a pressurizer relief discharge tank. Some of the functions provided by the pressurizer relief discharge system in previous nuclear power plants are provided by portions of other systems in the AP600.

The safety valves connected to the top of the pressurizer provide for overpressure protection of the reactor coolant system. First-, second-, and third-stage automatic depressurization system valves provide for depressurization of the reactor coolant system and venting of noncondensable gases in the pressurizer following an accident. These functions are discussed in subsections 5.2.2, 5.4.12, and in Section 6.3. The AP600 does not have power operated relief valves connected to the pressurizer.

The discharge of the safety valves is directed through a rupture disk to containment atmosphere.

The discharge of the first-, second-, and third-stage automatic depressurization system valves is directed to the in-containment refueling water storage tank. For the automatic depressurization system valves, the following discussion considers only the gas venting function. Only the first stage automatic depressurization valves are used to vent non-condensible gases following an accident. The sizing considerations and design basis for the in-containment refueling water storage tank for the depressurization function are discussed throughout Section 6.3. The provisions to minimize the differential pressure between the containment atmosphere and the interior of the in-containment refueling water storage tank are also discussed in subsection 6.3.2.

The safety valve on the normal residual heat removal system, which provides low temperature overpressure protection, discharges into the in-containment refueling water storage tank. See subsection 5.4.7 for a discussion of the connections to and location of the safety valve in the normal residual heat removal system.

5.4.11.1 Design Bases

The containment has the capability to absorb the pressure increase and heat load resulting from the discharge of the safety valves to containment atmosphere. The in-containment refueling water storage tank has the capability to absorb the pressure increase and heat load from the discharge, including the water seal, steam and gases, from a first-stage automatic depressurization system valve when used to vent noncondensable gases from the pressurizer following an accident. The venting of noncondensable gases from the pressurizer following an accident is not a safety-related function.

5.4.11.2 System Description

Each safety valve discharge is directed to a rupture disk at the end of the discharge piping. A small pipe is connected to the discharge piping to drain away condensed steam leaking past the safety valve. The discharge is directed away from any safety related equipment, structures, or supports that could be damaged to the extent that emergency plant shutdown is prevented by such a discharge.

The discharge from each of two groups of automatic depressurization system valves is connected to a separate sparger below the water level in the in-containment refueling water storage tank. The piping and instrumentation diagram for the connection between the automatic depressurization system valves and the in-containment refueling water storage tank is shown in Figure 6.3-1. The in-containment refueling water storage tank is a stainless steel lined compartment integrated into the containment interior structure. The discharge of water, steam, and gases from the first-stage automatic depressurization system valves when used to vent noncondensable gases does not result in pressure in excess of the in-containment refueling water storage tank design pressure. Additionally, vents on the top of the tank protect the tank from overpressure, as described in subsection 6.3.2.

Overflow provisions prevent overfilling of the tank. The overflow is directed into the refueling cavity. The in-containment refueling water storage tank does not have a cover gas and does not require a connection to the waste gas processing system. The normal residual heat removal system provides nonsafety-related cooling of the in-containment refueling water storage tank.

5.4.11.3 Safety Evaluation

The design of the control for the reactor coolant system and the volume of the pressurizer is such that a discharge from the safety valves is not expected. The containment design pressure, which is based on loss of coolant accident considerations, is greatly in excess of the pressure that would result from the discharge of a pressurizer safety valve. The heat load resulting from a discharge of a pressurizer safety valve is considerably less than the capacity of the passive containment cooling system or the fan coolers. See Section 6.2.

Venting of noncondensable gases, including entrained steam and water from the loop seals in the lines to the automatic depressurizations system valves, from the pressurizer into spargers below the water line in the in-containment refueling water storage tank does not result in a significant increase in the pressure or water temperature. The in-containment refueling water storage tank is not susceptible to vacuum conditions resulting from the cooling of hot water in the tank, as described in subsection 6.3.2. The in-containment refueling water storage tank has capacity in excess of that required for venting of noncondensable gases from the pressurizer following an accident.

5.4.11.4 Instrumentation Requirements

The instrumentation for the safety valve discharge pipe, containment, and in-containment refueling water storage tank are discussed in subsections 5.2.5, 5.4.9, and in Sections 6.2 and 6.3, respectively. Separate instrumentation for the monitoring of the discharge of noncondensable gases in not required.

5.4.11.5 Inspection and Testing Requirements

Sections 6.2 and 6.3 discuss the requirements for inspection and testing of the containment and in-containment refueling water storage tank, including operational testing of the spargers. Separate testing is not required for the noncondensable gas venting function.

5.4.12 Reactor Coolant System High Point Vents

The requirements for high point vents are provided for the AP600 by the reactor vessel head vent valves and the automatic depressurization system valves. The primary function of the reactor vessel head vent is for use during plant startup to properly fill the reactor coolant system and vessel head. Both reactor vessel head vent valves and the automatic depressurization system valves may be activated and controlled from the main control room. The AP600 does not require use of a reactor vessel head vent to provide safety-related core cooling following a postulated accident.

The reactor vessel head vent valves (Figure 5.4-8) can remove noncondensable gases or steam from the reactor vessel head to mitigate a possible condition of inadequate core cooling or impaired natural circulation through the steam generators resulting from the accumulation of noncondensable gases in the reactor coolant system. The design of the reactor vessel head vent system is in accordance with the requirements of 10 CFR 50.34 (f)(2)(vi).

The first stage valves of the automatic depressurization system are attached to the pressurizer and provide the capability of removing noncondensable gases from the pressurizer steam space following an accident. Venting of noncondensable gases from the pressurizer steam space is not required to provide safety-related core cooling following a postulated accident. Gas accumulations are removed by remote manual operation of the first stage automatic depressurization system valves.

The discharge of the automatic depressurization system valves is directed to the in-containment refueling water storage tank. Subsection 5.4.6 and Section 6.3 discuss the automatic depressurization system valves and discharge system.