Westinghouse Electric Corporation
Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

AW-97-1076

February 7, 1997

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

ATTENTION: MR. T. R. QUAY

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: WESTINGHOUSE RESPONSES TO NRC REQUESTS FOR ADDITIONAL INFORMATION ON THE AP600

Dear Mr. Quay:

The application for withholding is submitted by Westinghouse Electric Corporation ("Westinghouse") pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10CFR Section 2.790, Affidavit AW-97-1076 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-97-1076 and should be addressed to the undersigned.

Very truly yours,

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

/jml

cc: Kevin Bohrer NRC OWFN - MS 12E20

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PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) contained within parentheses located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Section (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR2.790(b)(1).

AW-97-1076

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Brian A. McIntyre, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Brian A. McIntyre

Brian A. McIntyre, Manager
Advanced Plant Safety and Licensing

Sworn to and subscribed
before me this 10th day
of February, 1997

Janet A. Schwab

Notary Public

Notarial Seal
Janet A. Schwab, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires May 22, 2000
Member, Pennsylvania Association of Notaries

3077A

(1) I am Manager, Advanced Plant Safety And Licensing, in the Advanced Technology Business Area, of the Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Energy Systems Business Unit.

(2) I am making this Affidavit in conformance with the provisions of 10CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Energy Systems Business Unit in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) Enclosed is Letter NSD-NRC-97-4977, February 7, 1997 being transmitted by Westinghouse Electric Corporation (W) letter and Application for Withholding Proprietary Information from Public Disclosure, Brian A. McIntyre (W), to Mr. T. R. Quay, Office of NRR. The proprietary information as submitted for use by Westinghouse Electric Corporation is in response to questions concerning the AP600 plant and the associated design certification application and is expected to be applicable in other licensee submittals in response to certain NRC requirements for

justification of licensing advanced nuclear power plant designs.

This information is part of that which will enable Westinghouse to:

(a) Demonstrate the design and safety of the AP600 Passive Safety Systems.

(b) Establish applicable verification testing methods.

(c) Design Advanced Nuclear Power Plants that meet NRC requirements.

(d) Establish technical and licensing approaches for the AP600 that will ultimately result in a certified design.

(e) Assist customers in obtaining NRC approval for future plants.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for advanced plant licenses.

(b) Westinghouse can sell support and defense of the technology to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar advanced nuclear power designs and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for developing analytical methods and receiving NRC approval for those methods.

Further the deponent sayeth not.

Enclosure 2

Contains Non-Proprietary Information

NSD-NRC-97-4977

Question 280.10

Re: STAFF FOLLOW ON QUESTIONS AND REVIEW STATUS, SSAR SECTION 9.5.1 - FIRE PROTECTION (June 24, 1996 letter), 9.5.1.3.2 Fire-Protection Water-Supply System

In Section 9.5.1.2.1.3 of the SSAR, Westinghouse states that, "The fire water supply system is designed in accordance with the BTP 9.5-1 and the applicable NFPA standards."

In Table 9.5.1-1 of the SSAR, under BTP CMEB 9.5-1 Guidelines 121-144, Westinghouse commits to follow the BTP CMEB 9.5-1 Guidelines, but notes that, because of conflicting design considerations, there may be a need to take exception to specific guidance. For example, the water in the passive containment cooling system (PCS) storage tank is not dedicated for fire protection. The staff does not consider the present design a conservative approach to providing water for fire protection and views the present design as a departure from current operating reactors. Westinghouse agreed to provide a detailed explanation in the SSAR of why the water in the PCS tank will not be needed for fire protection simultaneously with PCS operation. The staff will evaluate Westinghouse response and determine whether the proposed design is acceptable. This is RAI #280.10 and designated as Open Item 9.5.1.3-5.

Response

As stated in the Standard Review Plan, section 9.5.1, from a licensing perspective "the purpose of the fire protection program (FPP) is to provide assurance, through a defense-in depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment in accordance with General Design Criteria 3 and 5." AP600, unlike current operating reactors, has relatively few "safe shutdown functions" that can be affected by a fire. All of them that can be affected by a fire will have been completed by the time the PCS level is below that necessary to support the seismic standpipe fire protection system.

SSAR subsection 9.5.1.2.1, Revision 8, states that in the unlikely event that water supply from the passive containment cooling system (PCS) is unavailable or insufficient, the seismic standpipe system can be supplied from the fire main by opening the normally closed cross-connect valve with the plant fire main. Note that the equipment and spaces protected by the seismic standpipe system are seismically qualified. Thus, the probability of there being a need for PCS operation AND a fire in the areas serviced by the seismic standpipe system AND the plant fire main being out of service AND no temporary fire water supply (pumper type fire truck) available is very small. . For events such as loss of all AC power, the heat capacity of the IRWST allows for fire fighting to proceed for approximately four hours with the seismic standpipe system before water is required for containment cooling. This capability is considered adequate.

SSAR Revision: NONE

Question 410.244

Provide the following information regarding the radwaste building HVAC system (VRS):

a. Provide data for the men's and women's locker room exhaust fans, and revise Table 9.4.8-1 of the SSAR accordingly.

b. Figure 9.4.8-1 of the SSAR shows both high efficiency filters and low efficiency filters, while Table 9.4.8-1, Sheet 2 of 2, only discusses prefilters. Provide efficiency data for both high and low efficiency filters in Tables 9.4.8-1, Sheet 2 of 2 for the VRS. Also, show all HEPA filters and prefilters (18 of each type as indicated in Table 9.4.8-1, Sheet 2 of 2) on Figure 9.4.8-1, Sheets 1 and 2 of 2.

Response

The radwaste building HVAC system serves no safety-related function and therefore has no nuclear safety design basis. The level of design information presented in the SSAR for this system has been reduced from that presented in Revision 2 of the SSAR. Consequently, Table 9.4.8-1 and Figure 9.4.8-1 are no longer presented in the SSAR.

The following information addresses the above requests:

a. Men's and women's lockers have been deleted from the Radwaste Building; therefore, the associated exhaust fans are also deleted.

b. The revised Radwaste Building and radwaste process no longer require local filtration; therefore, these local filters have been deleted. Exhaust filters have been deleted. HEPA filters are not required to meet offsite dose releases.

SSAR Revision: NONE

Question 410.263

Branch Technical Position ASB 10-2 provides design guidance to meet GDC 4 on dynamic effects associated with possible water hammers in the feedwater piping. Specifically, the feedwater system should be designed to (a) prevent or delay water draining from the feedring following a drop in steam generator water level, (b) minimize the volume of feedwater piping external to the steam generator which could pocket steam using the shortest horizontal run (less than 7 feet), (c) perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur and provide the procedures for these tests for approval, and (d) implement pipe refill flow limits where practical. Address the AP600 feedwater system design against these guidelines.

Response:

As indicated in SSAR Section 10.4, the Startup Feedwater system and the condensate and feedwater systems include many features to minimize the potential for unacceptable water hammers in the feedwater piping. These features are further delineated in SSAR subsection 3B.2.3. These features include:

(a) The design of the feedwater system includes a number of features specifically intended to minimize water hammers. For example, the startup feedwater system is entirely separate from the main feedwater system, including its entry point into the steam generators. This allows the system flowrates and pipe sizes to be more consistent with their intended function than if startup feed flows were directed through main feed headers and feedrings. In addition, the main feedring itself is designed to minimize the potential for water hammer. The spray tubes are located on the top of the feedring so that the feedring does not drain when steam generator levels drop below the feedring level. The thermal sleeve is welded which also prevents drainage when steam generator levels fall.

(b) The main feedwater line is continuously sloped upward to the steam generator nozzle. The horizontal run from the steam generator to the feedwater elbow is minimized.

(c) Tests have been performed on many Westinghouse feedring type steam generators in the United States. These tests verify the effectiveness of Westinghouse feedring designs like that described above for AP600 in preventing water hammer. Westinghouse does not consider that further design testing is required.

(d) Pipe flow limits, especially on startup feed, are not required for AP600 because the startup feed path is separate from the main feed path, including its entry into the steam generator. In addition, unlike other PWRs, AP600 has no auxiliary feedwater system and associated lines and operations.

In addition, Chapter 14 of the SSAR includes a preoperational test (subsection 14.2.9.1.7) to verify unacceptable dynamic effects do not occur for expected startup and restart conditions. The changes below explicitly reinforce that plant dynamic effects testing includes feedwater related tests.

SSAR Revision:

Section 3.9.2.1, second paragraph, second sentence:

The pre-operational testing programs are outlined in subsection 14.2.8.9.

Section 14.2.9.1.7, subsection "Purpose":

The purpose of the expansion, vibration and dynamic effects testing is to verify that the safety-related, high energy piping and components are properly installed and supported such that expected movement due to thermal expansion during normal heatup and cooldown, and as a result of transients; thermal stratification and thermal cycling; as well as vibrations caused by steady-state or dynamic effects ~~during steady-state and transients~~ do not result in excessive stress or fatigue to safety-related plant systems and equipment, as described in Section 3.9.

Section 14.2.9.1.7, subsection "General Test Method and Acceptance Criteria", subparagraphs b) and c):

b) Vibration testing is performed on safety-related and high-energy system piping and components during both cold and hot conditions to demonstrate that steady-state vibrations are within acceptable limits. See Subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to steady-state vibration. This testing includes visual observation and local and remote monitoring in critical steady-state operating modes. Results are acceptable when visual observations show no signs of excessive vibration and when measured vibration amplitudes are within acceptable levels.

c) Testing for significant vibrations caused by dynamic ~~events~~ effects is conducted during hot functional testing and may be performed as part of other specified preoperational tests. This testing is conducted to verify the stress analyses of safety-related and high-energy system piping under transient conditions are acceptable. See Subsection 3.9.2.1.1 for the acceptable standard for alternating stress intensity due to dynamic effects vibration. These tests are performed to verify that the dynamic effects ~~are within expected values~~ caused by ~~during~~ transients such as pump starts and stops, valve stroking, and significant process flow changes are within expected values. These tests include anticipated normal operating evolutions with system differential temperatures, such as, startup, which could induce dynamic effects.

Question 440.197

How much condensate is expected in the containment following an MSLB and feedwater line break, respectively? Can this lead to boron dilution if the scenario goes into ADS actuation?

Response:

As discussed in subsection 15.1.5, at no time during the analyzed steam line break or feedwater line break event does the core makeup tank level approach the setpoint for actuation of the automatic depressurization system. Recirculation flow from the reactor coolant system cold legs into the core makeup tanks maintains a relatively large core makeup tank water inventory throughout the event. However, should it be postulated that an ADS actuation were to occur during these events, the condensate from the shell will be delivered primarily to the IRWST.

To determine the potential for dilution of the IRWST from a secondary side break, the mass of secondary side fluid from a 30% power main steam line break, for example, can be considered for potential dilution of the mass of borated water in the IRWST post accident. The minimum boron concentration in the IRWST is 2400 ppm and based on a minimum IRWST mass, the potential dilution from all of the secondary side fluid would result in a concentration of 2293 ppm boron in the IRWST. This new IRWST boron concentration is above a level that would be considered to be a potential dilution problem. The ADS discharge to the IRWST will serve to maintain the contents of the tank mixed to limit stratification.

Should the condensate be assumed to flow to the containment sump, then the injection of the sump fluid would not occur until the IRWST had been drained either through the RCS and relieved through ADS or via opening of the isolation valves in the sump discharge lines and draining the IRWST to the sump. In either case, there is no mechanism for stratification of unborated water. The condensate return will all be heated limiting potential thermal stratification in the sump. In the event of draining the IRWST directly to the sump the draining process will result in mixing action.

SSAR Revision: NONE

Question 440.395
Re: Open Item 2698

The logic changes for ADS actuation on CMT level that were made early in 1994 are not reflected in the third paragraph of Section 6.0 (p. 6-1). The staff is aware that these changes may, in fact, not have been implemented in the OSU operating procedures. Nevertheless, since the publication of the final version of this report is after the design/logic changes were made, those changes should be acknowledged. Provide this information.

Response:

Section 6.0 of the Reference 440.395-1 describes the performance of the AP600 Passive Safety Injection Systems, including the design changes made in February, 1994 as described in reference 440.395-2. However, in paragraph 3, the setpoint for the initiation of ADS based on CMT level was incorrectly stated as 25% of the CMT volume. The correct value is approximately 33% of the CMT volume.

This correction will be made and included in the errata to be issued for the OSU Facility Scaling Report.

In addition, all design changes described in Reference 440.395-2 were assessed for impact on the OSU test facility and those judged to be important were incorporated in the test facility prior to the start of matrix testing in June, 1994. These changes included the logic for ADS actuation which initiates stage 1 at 67% of CMT volume, stages 2 and 3 on timers following stage 1 initiation, and stage 4 at 20% of CMT volume.

Design changes delineated in Reference 440.395-2 that were incorporated in the OSU tests will be added to Section 6.0 as clarification.

References

440.395-1 WCAP-14270, "AP600 Low-Pressure Integral Systems Test at Oregon State University Facility Scaling Report,", January, 1995

440.395-2 "AP600 Design Change Description Report,", February 15, 1994

SSAR Revision: NONE

Question 440.570

Re: OSU FDR and TAR

Recent staff evaluation of OSU data indicates that a check valve failure in the normal RHR line opened a flow path between the two DVI lines. The largest potential impact of this occurrence appears to be in the DVI-line break tests, in which flow from the intact DVI line could be diverted to the broken line. Explain how this condition affects reliability of the test data from OSU for these events, and how the flow path is accounted for in analyzing the data. The test's ability to meet Westinghouse's established acceptance criteria should also be discussed.

Response:

For the OSU test facility, flow meter FMM-206 records the total DVI-2 line flow delivered to the reactor vessel. The NRHR line junction with the DVI-2 line is upstream of flow meter FMM-206. Therefore, should flow be diverted from the intact line through the NRHR line, flow meter FMM-206 would record the actual flow delivered to the reactor vessel through the intact DVI-2 line. Also, the bypass flow that enters DVI-1 will be measured by FMM-205 before the flow exits the break.

The analysis of the OSU data performed in Reference 440.570-1 uses a total flow as measured by a single instrument to calculate mass balances to minimize uncertainty in analysis results. Accordingly, the flow as measured with FMM-206 was used in calculating mass delivery into the reactor vessel for the DVI tests. Thus, the possible bypass flow from the intact DVI-2 line to the broken DVI-1 line through the NRHR line due to a failure of check valve RCS-826 to fully seat does not negate the test from satisfying established test acceptance criteria since the bypass flow is accounted for in these measurements.

In addition, sufficient instrumentation is available to determine if bypass flow occurred in the NRHR line. Flow meters measure the total DVI line flow (FMM-205 and FMM-206) as well as the individual flow components from the CMT (FMM-501 and FMM-504), accumulator (FMM-401 and FMM-402), IRWST injection (FMM-701 and FMM-702) and the containment recirculation lines (FMM-901 and FMM-902). Also, fluid thermocouples (TF-813 and TF-814) are located in the NRHR line which can detect changes in the fluid temperature in the NRHR line near the check valves. Note that since valves RCS-801 and RCS-802 are open during the test, these thermocouples may be exposed to fluid from the DVI lines whether or not bypass flow occurs in the NRHR line.

For example, during the IRWST injection phase of the test, the flows measured in each DVI line (FMM-205 and FMM-206) can be compared to the corresponding IRWST flows (FMM-701 and FMM-702 respectively) entering the DVI lines. If the DVI and IRWST flows in each injection line equate, by deduction, there is no bypass flow. However, if there are flow differences, then flow may have been diverted through the NRHR line due to the pressure differential between the DVI lines. The flow differential in DVI-2 (FMM-702 minus FMM-206) would be measured in DVI-1 (FMM-205), which in the case of a double ended guillotine (DEG) DVI break is the break flow from the PXS side of the break.

A comparison of the DVI and IRWST flows was made for two DEG DVI line break tests, SB12 and SB28, using the methodology described above. SB12 and SB28 are both breaks of DVI-1 with different failures. SB12 had only a single failure of one of the two lines of ADS-1 and ADS-3. SB28 had multiple failures including ADS 4-1 and

ADS 4-2 and ACC-1 and no PRHR actuation. A comparison of the two tests in Reference 440.570-2 shows the timing behavior of these two tests are very similar until primary sump injection occurs, well into the conduct of the tests.

For test SB12, performed on July 21, 1994, a comparison of the DVI and IRWST flows during the IRWST injection period (Figures 440.570-1 and 440.570-2) indicate the DVI flows and IRWST flows in each line are equal. Therefore, it is assumed no bypass flow occurred during this period of the test.

Examination of the NRHR fluid thermocouples in test SB12 (Figure 440.570-3) show some heatup at the beginning of the transient and cooling thereafter. The initial heatup may be due to the injection of the hot CMT fluid into the DVI lines, followed by the cooler accumulator and IRWST injection fluids.

Test SB28, performed on September 7, 1994, shows that flow differences exist between the DVI lines and their corresponding IRWST injection line during the IRWST injection phase of the test (Figures 440.570-4 and 440.570-5). When the flow differential for DVI-2 and IRWST-2 (Figure 440.570-6) is added to the flow measured in IRWST-1 and compared with the flow in DVI-1 (Figure 440.570-7), they are equal throughout the IRWST injection period. This indicates bypass flow may have occurred in the NRHR line for this test.

Examination of the fluid temperatures in the NRHR line for test SB28 (Figure 440.570-8) shows initial heatup, then alternating cooling and heating, followed by slowly increasing temperatures throughout the transient. Maximum temperatures for fluid thermocouple TF-814 correspond to the times when the flow differentials shown in Figure 440.570-6 are zero, which may indicate a lack of cooling of the NRHR line when no bypass flow occurs. The heatup trend throughout the remainder of the transient matches the rise in fluid temperature in the IRWST for the same time period, which again indicates bypass flow in the NRHR line.

Based on these comparisons, check valve RCS-286 appears to be seated during test SB12 and was not fully seated for test SB28. However, sufficient instrumentation exists to determine the apparent bypass flow in the NRHR line during test SB28. Examination of the fluid temperatures in the NRHR line for these tests support these conclusions. The reliability of the test data is not affected as evidenced by the similar performance of these tests as reported in Reference 440.570-2. Also, the test facility provides for a total flow rate measurement in the DVI line downstream of the NRHR/DVI line junction. Thus, the data from these tests remains valid and can be used for code validation purposes.

Reference 440.570-1 WCAP-14292, Revision 1, "AP600 Low Pressure Integral Systems Test at Oregon State University: Test Analysis Report," Proprietary [LTCT-T2R-600], September 1995

Reference 440.570-2 WCAP-14252, "AP600 Low Pressure Integral Systems Test at Oregon State University: Final Data Report," Proprietary [LTCT-T2R-100], May 1995

SSAR Revision: None

Question 440.571

Re: OSU FDR and TAR

Over a range of several thousand seconds in many tests, the PRHR HX wide range level appears to drift upwards considerably. Explain what is occurring: is this an actual change in level, or is an instrumentation problem?

Response:

A review of the data from test SB18 suggests that the apparent upward drift of the wide range level transducer output is an indication of an actual change in level until about 2250 seconds into the test. The interpretation of the data leading to this conclusion is as follows:

- For this test, the indicated level in the PRHR goes to zero at about 700 seconds. Coincidentally, the rate of change of the fluid temperature in the PRHR inlet header is observed to at first stop, and then continue to decrease at a very small rate to about 900 seconds. These data are interpreted as indicating that the pressure head on the hot leg side has decreased sufficiently that a slight suction is applied to the residual water in the PRHR discharge line through the heat exchanger tubes that supports a 5.0 inch head of water in the tubes.

- At about 930 seconds, 4th stage ADS is actuated. As the PRHR feed line and the 4th stage ADS line for loop 2 share a common takeoff from hot leg 2, the pressure in the PRHR feedline is further reduced, causing the water level in heat exchanger tubes to rise further. This rise in heat exchanger water level is coincident with a rapid decrease in the temperature of the fluid in the PRHR heat exchanger inlet header to about saturation temperature at 1 atmosphere. After that time the fluid temperature in the PRHR inlet header becomes and remains subcooled.

- From about 1200 seconds until about 1600 seconds, the water level in the tubes remains constant and the PRHR heat exchanger inlet header fluid temperature increases to near saturation at 1 atmosphere. This is taken to be the result of steam drawn into the header by condensation.

- Between about 1600 seconds and 2250 seconds, the indicated water level in the tubes again increases slightly, and the inlet header fluid temperature is observed to again decrease. This is consistent with the time period for which liquid flow is again measured as being discharged from the PRHR. Thus the indicated increase in water level, from about 5 inches to about 12 inches of water in the PRHR from about 1600 seconds to about 2250 seconds is interpreted as resulting from condensate forming on the inside surfaces of the PRHR tubes.

- After about 2250 seconds, the level measurements for the PRHR heat exchanger indicates that the water level in the PRHR tubes again increases. As the measured outlet flow remains at about zero, this increase in water level is evaluated as an indication of flashing of the reference leg rather than a build up of water in the exchanger tubes.

Thus, for test SB18, the level indication for the PRHR heat exchanger is interpreted as an actual increase in the water level in the PRHR heat exchanger tubes that occur when the hot leg pressure on the PRHR heat exchanger drops below the backpressure in the outlet plenum of SG2, and again when ADS 4 actuates. Later in time when the PRHR flow measurement drops to zero, the increase in indicated PRHR water level is interpreted as indication flashing of the levels transducer reference leg. This explanation is expected to apply to all tests for which a similar behavior was observed.

SSAR Revision: NONE

Question 440.572

Re: Presentation of data for Test SB18 in Section 5.1.2 of the OSU FDR.

The explanation for the holdup of CMT-2 injection is that the accumulator injection closed the CMT outlet check valve, thus preventing CMT-2 from draining. However, Figure 5.1.2-6 shows CMT-2 level hanging up between around 125 and 350 seconds, while Figure 5.1.2-16 does not show significant accumulator injection until about 400 seconds, by which time CMT-2 is draining at roughly the same rate as CMT-1. Is there another possible explanation for this behavior, such as condensation at the top of CMT-2?

Response:

The break location for this test is at the bottom of cold leg 3 (CL-3), the same cold leg as the CLBL for CMT-01. The effect of the break location is to allow the CLBL for CMT1 to drain earlier during the test than does the CLBL line for CMT-02. The difference in time of draining for the two CLBLs is readily noted in Figure 5.1.2-6.

From Figure 5.1.2-6, CMT-01 CLBL is noted to drain at about 125 seconds. With CMT-01 CLBL line drained, steam may enter CMT-01 and draining of that tank may begin. CMT-01 is noted to begin draining coincident with low level in its associated CLBL.

Similarly, CMT-02 remains full and in recirculation until its CLBL drains. Again referring to Figure 5.1.2-6, CMT-02 CLBL drain down is noted to be completed at about 350 seconds and, coincidentally, CMT-02 also begins to drain. Westinghouse concludes that the data from this test suggests that the draining of the CMTs is strongly dependant upon the time the CLBLs drain and not condensation.

SSAR Revision: NONE

Question 440.573

Re: Presentation of data for Test SB18 in Section 5.1.2 of the OSU FDR.

In comparing Test SB19 to SB18:

a. Why is the transition from recirculation from draining in the CMTs later in SB19 than in SB18?

b. Is there a systematic explanation for differences in core levels and timing of events during the initial depressurization phase?

c. Why are the break flows higher in SB19 for the first 400 seconds?

If the differences are ascribed to the simulation of an elevated containment backpressure in SB19, provide a detailed explanation of the ways in which the containment pressure influences early-phase reactor/safety systems performance. It is not clear how the containment pressure is "felt" by the RCS, since, for instance, critical flow out the break and ADS valves should be insensitive to the ambient pressure. In addition, the discussion should address possible influences of the BAMS upon RCS response; i.e., if the behavior noted in the OSU facility is in part related to that aspect of the loop design.

Response:

The difference in observed system performance is attributed to simulating an increase in containment backpressure. The increase in simulated containment backpressure results in the following;

a. One affect of the increased containment backpressure was to delay the draining of the CMT CLBLs in Test SB19 relative to their behavior observed for SB18. Thus, the CMTs did not begin to drain as early for test SB19 as for test SB18. From Figure 5.1.3-9, it is noted that the general behavior of the draining remains the same between the two tests; that is, the CLBL nearest the break drains earliest, allowing its associated CMT to begin to also drain.

b. The observed differences in core levels and timing of events between test SB19 and SB18 are the result of differences in the local pressure in the core region that are driven by the effects of a higher containment backpressure simulated for Test SB19. Specifically the saturation temperature for SB19 is higher than that for test SB18 and the energy input to the working fluid from the core simulation is the same for tests SB19 and SB18. Thus, after the initial blowdown, it is expected that phenomena driven by boiling of core coolant will occur later in test SB19 than in test SB18.

c. The higher containment backpressure simulated in test SB19 is applied to the break separator. The quality of break flow is dependant upon its discharge backpressure. As the pressure in the break separator is higher for test SB19 than for test SB18, the quality of the break flow for SB19 is lower that for SB18. Thus, for the same pressure drop across the break, a larger amount of liquid is discharged from the break early in test SB19 than for test SB18.

SSAR Revision: NONE

Question 440.574

Re: Presentation of data for Test SB18 in Section 5.1.2 of the OSU FDR.

For test SB12, explain why ADS1-3 flow becomes negative (Figure 5.4.1-18).

Response:

The indicated negative flow of liquid through the loop seal of the ADS 1-3 separator is evaluated as being the result of a "sloshing" effect of liquid in the ADS 1-3 loop seal at that time. Note that the data of Figure 5.4.1-18 suggests oscillations in the flow measurements. The definitive calculation of ADS1-3 flow is presented in the OSU TAR and takes into account change in stored fluid (vapor and liquid) mass of the ADS1-3 separator and vapor flow as well as the liquid flow shown in Figure 5.4.1-18.

Note that the magnetic flow meters are not calibrated for reverse or back flow. The negative sign associated with the meter output is indicative of the direction of flow only; the magnitude of the signal is not a valid indication of the volume of flow in the reverse direction.

SSAR Revision: NONE

Question 440.576

Re: OSU TAR

Please provide a discussion, supported by quantitative analysis, demonstrating that the uncertainty in OSU data is bounded by instrumentation uncertainties/errors, as appears to be implied by the instrument error analysis in the FDR; or, if this is not the case, to establish the bounds of those uncertainties. This can be looked upon as determining the "error bars" that would be placed on the quantities plotted in the FDR and TAR.

The discussion should focus particularly on derived quantities; i.e., those using the output of several instruments (e.g., adjusting level readings using density corrections derived from temperature data at discrete locations), or those in which assumptions or models must be used (e.g., "filtering" or "smoothing" of instrument readings, use of fluid temperatures to represent wall temperatures, assumption of adiabatic conditions at certain wall boundaries, use of empirical models to derive some two-phase flow parameters) to infer elements of system response. The effects of component failures and/or system interactions should also be considered, such as the impact of the failed RHR valve on flow through the nominally intact line during DVI break tests.

Response:

Westinghouse will provide a discussion, supported by quantitative analysis, of the uncertainty in OSU data that accounts for instrumentation uncertainties/errors. The discussion will focus on derived quantities, that is;

- Those quantities whose calculation uses the output of several instruments (e.g., adjusting level readings using density corrections derived from temperature data at discrete locations), or,
- Those in which assumptions or models must be used (e.g., "filtering" or "smoothing" of instrument readings, use of fluid temperatures to represent wall temperatures, assumption of adiabatic conditions at certain wall boundaries, use of empirical models to derive some two-phase flow parameters) to infer elements of system response.

The effects of component failures and/or system interactions will be considered only if they increase the uncertainty associated with the determination of the overall system mass or energy balance. The uncertainties evaluated to support this discussion will be presented in the form of "error bars" placed on the calculated quantities plotted in the TAR.

Westinghouse proposes to provide this discussion by Febuary 17, 1997. This schedule accounts for calculations to be performed in support of developing the requested discussion.

SSAR Revision: NONE

Question 440.577

Re: OSU TAR

The original version of the OSU TAR contained only two tests SB01 and SB18, with SB01 represented as the "baseline" test for OSU in both the FDR and the TAR. During recent discussions with the staff, Westinghouse has stated that SB01 is now considered "atypical" due to the absence of the ADS discharge line vacuum breaker, and that SB18 ought to be used as the "benchmark" for the OSU facility. Westinghouse should make the necessary changes to reflect this position in the FDR and TAR (e.g., revise Tables 5.1.2-3 and 5.1.3-3).

Response:

Westinghouse will issue errata sheets for the FDR and TAR.

SSAR Revision: NONE

Question 440.575

Re: OSU TAR

Two "issues" are identified concerning the core fluid thermocouples in Section 4.11, but no subsequent analysis or explanation of the issues is provided. Specifically,

a. How did the fluid temperature histories at the center and perimeter differ?

b. The "best average core temperature" is asserted to be represented by the center-rod temperatures, without quantitative justification. Why is this procedure preferable to a weighted average of the core and perimeter rods?

Response:

a. Thermocouples at the periphery or perimeter of the heater rod bundle were lower than those measured at the center of the rod bundle by about 10-15 degrees as the test progressed into the long-term cooling portion of the transient.

The attached two time history plots of heater rod thermocouple data from test SB-18 illustrate the difference between measurements taken at the bundle center and the bundle periphery. Rod B1-103 is near the bundle center and Rod B2-503 is at the bundle periphery. Two thermocouples were chosen for this illustration, both are at the 46.13 inch elevation.

The first plot shows that during the initial cooldown of the transient, the two thermocouples behave in a very similar manner. After about 600 seconds into the transient, a temperature difference between the center and the periphery quickly develops and maintains itself over the transient.

The second plot is an expanded view of data from the same two thermocouples from 500 seconds to 1000 seconds. From about 600 seconds after initiation of the transient and beyond, the data of this plot suggest that a temperature difference of about 10 to 15 degrees generally exists between the two thermocouples.

b. The fluid temperature measurement taken at the center of the rod bundle was used as representative of the core for the following reasons;

1) The radial power profile was flat; that is, each rod received the same power.

2) The power-to-flow ration across the heater rod bundle, except those at it's periphery, was essentially uniform.

Thus, for a specific elevation, given a flat radial power profile, a constant power to flow ration and the open lattice structure of the heater rod bundle, a radially uniform fluid temperature is expected. Although the effect is small, including the peripheral rods provides for conservatively high steam generation rates.

SSAR Revision: NONE

Question 440.580

Re: OSU TAR (OITS 3471)

In Figure 5.4.2-33 of the OSU TAR, curve "C" shows integrated PRHR heat removal. The curve peaks at around 600-800 seconds, after which it begins to decrease. If this is an integrated curve, a decrease would seem to indicate heat transfer from the IRWST to the primary system, which does not seem to be physically plausible. Please explain what this curve shows and the reason for its shape.

Response:

As stated in the text of page 5.4.2-1, the thermal energy transfer from the PRHR was calculated from the change in the IRWST fluid energy, allowing for the contributions of energy carried into the IRWST by the ADS 1-3 inflow and out of the IRWST by the IRWST injection flow. Prior to actuation of ADS 1-3, the energy increase of the IRWST inventory was attributed solely to operation of the PRHR. ADS 1-3 was actuated at about 520 seconds into the transient. Once ADS 1-3 was actuated, circulation through the primary loops was quickly broken, heat rejection through the PRHR quickly dropped to a minimal level, and the predominant heat rejection path from the primary system to the IRWST was by way of ADS 1-3. Furthermore, at about 1240 seconds after initiation of the transient, injection from the IRWST was initiated; this provided for a mass and energy loss from the IRWST.

From Figure 5.4.2-33, the effectiveness of the PRHR to reject heat is calculated to terminate at about 540 seconds into the transient. Although the PRHR rejects minimal heat to the IRWST after this time, the energy balance equation solved for the IRWST accounts for heat transfer from both the PRHR and ADS 1-3 throughout the entire duration of the test. After about 540 seconds, the total amount of energy transferred from the PRHR should be expected to remain either nearly constant or increase slightly over time. However, systemic errors associated with the calculation of energy carried by the ADS flow into the IRWST and with energy removal associated with IRWST injection flows may affect the inferred integrated energy transfer from the PRHR.

The calculated decrease in total energy transferred by the PRHR begins at about 800 seconds into the transient. As this decrease occurs at about 250 seconds following actuation of ADS 1-3, this decrease is evaluated as resulting from systemic errors in the calculation of ADS 1-3 flows and energy levels. Furthermore, in terms of total thermal energy of the system, the energy transfer associated with the PRHR is small. Thus, errors associated with PRHR thermal energy have a minimal affect on the overall energy balance of the [illegible]

SSAR Revision: NONE

Question 440.581

Re: OSU TAR (OITS 3472)

There appears to be a slight offset in time on Figure 5.4.2-67 of the OSU TAR; the break flow begins to rise before time "zero". Please explain this behavior.

Response:

The plot of Figure 5.4.2-67 does show an apparent vapor mass flow prior to the initiation of break simulation (time t = 0.0 seconds). This behavior was likely the result of an unintentional shift in the time scale when the plot was generated. Further, it is noted that:

1) the steam mass flow is small; ~ 0.8 lbm/sec, and,

2) the time shift is small, ≤ 10 sec.

Thus, the shift has negligible affect on the conclusions reached or use of the data.

The data will be reviewed to assess if an amended plot should be provided with the errata for the OSU TAR.

SSAR Revision: NONE

Question 440.582

Re: OSU TAR (OITS 3473)

In Section 5.4.2.2 of the OSU TAR, in the subsection entitled "Energy Transport via the Break and Automatic Depressurization System," Westinghouse's primary concern is with the integrated behavior of the system as a function of time. However, the description of the behavior of the flow, (see introductory paragraph of the subsection), does not capture the highly dynamic nature of the system's response, and does not appear to consider whether any important phenomena are involved in driving that response. The staff believes that an understanding of the local phenomena driving system behavior should be reflected in the text. Specifically:

a. It is unclear why a "reference value" of 4 lbm/sec was chosen for the break liquid flow rate. The break flow peaks at a value well above 4 lbm/sec, drops briefly to that value, then appears to oscillate about a mean value less than 4 lbm/sec, and finally increases gradually for about 200 seconds (for reasons explained in Section 5.4.2.2), before oscillating briefly again and dropping to near zero, presumably coincident with the opening of the ADS valves. Please explain how the reference value was chosen and the phenomena governing system response.

b. Please explain the sharp downspikes (negative flow) and general oscillatory behavior occurring between about 100 and 200 seconds. The staff understands that negative indicated flows may not be accurate due to instrumentation limitations, but it appears that system-wide oscillations occurred during this period (as indicated by plots of other parameters), which may have influenced the break flow. Further, how did opening the CMT discharge valves affect break flow?

c. The increase in break flow is attributed to a decreased enthalpy at the break, however, there is no explanation of why the enthalpy decreased. Please address.

Response:

a. Reference to approximate average values was often used in describing the overall response of the observed phenomena; describing all transient observations in detail would have been prohibitive in terms of both schedule and volume. The "reference value" of 4 lbm/sec was chosen as an approximate average value over the time period of interest.

   The oscillatory behavior of the break flow is due to the unsteady nature of two-phase flow and to the break flow measurement process.

b. The break flow is measured by separating the flow into its vapor and liquid components and measuring each component separately. The vapor flow is then exhausted via the BAMS to atmosphere, and the liquid flows into the sump simulation. The gooseneck-like drain for the liquid drained from the break separator also holds the liquid flow meter.

   The observed oscillations in break flow are due to the intermittent draining of liquid through the gooseneck-like drain.

c. The decrease in enthalpy referenced in the text of Section 5.4.2.2 is due to the injection of accumulator and CMT inventory.

SSAR Revision: NONE