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

ATTENTION: R. W. BORCHARDT

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

Dear Mr. Borchardt:

Enclosed are three copies of the Westinghouse responses to NRC requests for additional information on the AP600 from your letters of August 18, 1994 and September 2, 1994. In addition, revisions of responses previously submitted are provided. A listing of the NRC requests for additional information responded to in this letter is contained in Attachment A.

These responses are also provided as electronic files in WordPerfect 5.1 format with Mr. Kenyon's copy.

If you have any questions on this material, please contact Mr. Brian A. McIntyre at 412-374-4334.

Brian A. McIntyre for

Nicholas J. Liparulo, Manager
Nuclear Safety Regulatory and Licensing Activities

/nja

Enclosure

cc: B. A. McIntyre - Westinghouse
T. Kenyon - NRR

NTD-NRC-95-4405
ATTACHMENT A
AP600 RAI RESPONSES
SUBMITTED FEBRUARY 7, 1995

RAI No.	Issue
210.100R01	Flow-induced vibrations of reactor internals
210.102R01	Preoperational vibration test program
210.118	ALARA exposure for RCP flywheel
210.127	RCP enciosure seal weld tensile loads
440.255	Fraction of core power deposited in relector
620.009R01	Lessons learned (nam-machine interface system)
620.015R01	M-MIS design team
620.072R01	Allocation decisions

Question 210.100

The response to Q210.16 dated January 1, 1993 indicates that preoperational test data from several operating plants and from scale-model flow tests are used for the assessment of flow-induced vibrations of the AP600 reactor internals. The response also indicates that the assessment has not yet been finalized and the effort was planned to be completed in the first quarter of 1994. When this assessment is complete, revise Section 3.9.2 of the SSAR to provide a more detailed summary of the assessment results used (a) for verifying your conclusions on the adequacy of the AP600 reactor internals design to withstand flow induced vibration, and (b) to provide the basis for classifying the first AP600 plant as Non-Prototype, Category II in accordance with Position C.1.5 of RG 1.20.

Response: (Revision 1)

a. Response Revision 1 to RAI 210.16 outlines the vibration assesment performed on the reactor internals for the AP600. The SSAR revisions to include the additional information for the assessment results are provided below.

b. The basis for classifying the first AP600 plant as Non-Prototype, Category II is provided below in paragraphs to be added as a SSAR revision. **Response revision 1 for this RAI provides additional basis for the classification in the revision for the third paragraph of Section 3.9.2.4 and an additional sentence in the revision to the sizteenth paragraph of Section 3.9.2.3.**

SSAR Revision:

Revise the tenth paragraph of Section 3.9.2.3 as follows:

These tests confirmed that the internals behaved as expected and that the vibration levels were within allowable values. The vibration testing for 17x17 fuel internals and inverted hat upper internals is reported in WCAP-8766 (Reference 4) and WCAP-8516-P (Reference 5). The vibration testing of three-loop XL type lower core support structure in DOEL 4 is reported in WCAP 10846 (Reference 6). The vibration evaluations of upper and lower internals assemblies for a four-loop XL plant, including reference to the test results in Paluel 1 (four-loop XL type without neutron pads), are reported in WCAP-10865 (Reference 7).

The results of the Doel 3, Doel 4 and Paluel 1 reactor internals vibration test programs are utilized to perform the vibration assessment of the AP600 reactor internals. The measured responses from Doel 3 and Doel 4 are adjusted to the lower AP600 flow rate to determine the expected upper internals and lower internals vibration levels respectively. The AP600 flow velocities are 85 per cent or less of the Doel 4 values at various locations in the reactor internals which results in lower responses in the AP600.

Revise the thirteenth paragraph of Section 3.9.2.3 as follows:

The vibrations of the upper internals components are well characterized by previous plant testing based on the following: The control assembly guide tubes and support column designs are similar to those in a previously tested plant. The outlet nozzle velocity is less than the outlet nozzle velocity of previously tested three-loop plants.

The AP600 upper internals design is substantially the same as that measured in the Doel 3 plant and 3XL scale model tests. The AP600 support column, guide tube and upper support assembly are nearly identical to the components in the test. There are slight differences in the numbers of guide tubes and support columns but little effect on the vibrational responses is expected since these components respond as individual beams. The corresponding AP600 responses were calculated to be less than the previous plant responses due to the lower flow velocities in the AP600.

Revise the fifteenth paragraph of Section 3.9.2.3 as follows:

The core barrel outside diameter and inside diameter and the reactor vessel inside diameter are the same as the tested three-loop plants. The core barrel length is one foot shorter. The coolant velocity in the downcomer annulus between the core barrel and the reactor vessel wall is lower in the AP600 design than in previous three-loop plants because the AP600 has no thermal shield or neutron pads in the annulus and the total reactor coolant flow rate is lower.

The vibrational response of the core barrel was measured during the Doel 4 reactor internals vibration measurement program. The diameter, length and thickness are nearly identical to the AP600 core barrel and both utilize the single combined lower core support plate. The cantilever beam mode frequency and amplitude of the AP600 core barrel are calculated to be similar to the measured Doel 4 responses. Comparison of the 4XL scale model to the Paluel plant test results indicate that the removal of the neutron panels has little effect on core barrel vibration.

Revise the sixteenth paragraph of Section 3.9.2.3 as follows:

The reflector is shorter than the core barrel, has a larger cross sectional area and a smaller diameter than the core barrel, and is more rigidly clamped at its axially supported end, so that its vibration is expected to occur primarily coupled to the core barrel and to have small vibration levels relative to the core barrel.

The replacement of the baffle-former structure with the radial reflector reduces the stiffness of the core barrel. The AP600 shell mode amplitudes are estimated to be higher than the standard 3 loop core barrel responses based on scaling the measured responses to the AP600 reduced core barrel stiffness. The AP600 shell mode amplitudes are acceptable.

The AP600 core barrel and reflector will be instrumented during the preoperational testing of the first plant to determine the shell mode frequencies and amplitudes.

Revise the third paragraph of Section 3.9.2.4 as follows:

With respect to reactor internals, the first AP600 plant is classified as a Non-Prototype Category II according to Regulatory Position C.1.5 in Regulatory Guide 1.20. The comparison in Subsection 3.9.2.3 of the AP600 reactor internals with previous Westinghouse designs supports this classification. The AP600 reactor vessel internals do not represent a first-of-a-kind or unique design based on the arrangement, design, size, or operating conditions. The units referenced in the Subsection 3.9.2.3 as supporting the AP600 reactor vessel internals design features and configuration have successfully completed vibration assessment programs including vibration measurement programs. These units have subsequently demonstrated extended satisfactory inservice operation.

The prototype (reference) plant for the AP600 is H. B. Robinson which has substantially the same size and operating conditions as the AP600. Structural differences include modifications resulting from the use of 17x17 fuel, the removal of the thermal shield and the change to the inverted top hat upper internals support assembly. These design changes were incorporated into the Doel 3 and Doel 4 reactor internals as well as the AP600.

The effects of these design evolutions from the reference plant were shown by instrumented preoperational testing at the Doel 3 (upper internals) and Doel 4 (lower internals) plants. The vibrational responses of the AP600 reactor internals are characterized by the Doel 3 and 4 vibration measurement programs.

Question 210.102

The response to Q210.18 dated January 14, 1993 indicates that the preoperational vibration test program for the initial AP600 plant remains to be developed. Thus, detailed information regarding the program, including types and locations of sensors to be installed, the bases used to establish expected and acceptable vibration levels, and the conditions at which data are to be acquired, is not available at this time. The staff's position is that such information is essential for ensuring design adequacy of reactor internals to withstand flow-induced vibrations under operational transients. Subsequent to the staff receiving an acceptable response to Q210.100, develop and provide such information in the SSAR for design certification review.

Response:

As noted in Revision 1 of the response to RAI 210.16 the reactor internals flow-induced vibration assessment program has been completed. As noted in Revision 1 of the response to RAI 210.18 a test plan for reactor internals vibration has been developed. This test will be similar to previous plant tests and will provide added confidence of the adequacy of the reactor internals design, in addition to the vibration assessment program. The following is a summary of the plan.

The reactor internals flow-induced vibration measurement program will be conducted during preoperational tests of the first AP600 plant. The major structural components of the first AP600 plant reactor lower internals will be instrumented during pre-operational testing. Transducers will be installed on the reactor vessel and the internals prior to the cold hydrostatic test. The integrity of these transducers and the operability of the data acquisition equipment will be verified during this test.

Some units used as reference design for the AP600 reactor vessel internals are located in foreign countries. As a result the NRC may have limited information on those vibration assessment programs. Because of this, the first plant will include vibration measurement and evaluation beyond that required for a Non-Prototype, Category II design. This additional measurement will consist of transducers positioned to address design features that have been validated, in part, by the vibration assessment programs of units located in foreign countries. The additional transducers will be located on guide tubes in the upper internals. This additional vibration measurement will provide the NRC staff an additional level of confidence in the design without the need to do an extensive review of the data from the vibration measurement programs of the foreign units.

The response of the reactor and the internals due to flow-induced vibration will be measured during the hot functional testing. As shown by the results of the vibration assessment program, the dominant vibration modes of the internals with no core present are similar to those with the core in place and their vibration amplitudes are expected to be more than 10% higher.

Data will be acquired at several temperatures from cold startup to hot standby (529°F) conditions. Data will be recorded for pump startup and shutdown transients as well as for all possible combinations of steady-state pump operation. In addition, data will be recorded with none of the pumps operating in order to determine the background noise level.

Transducer signals will be monitored as they are being recorded to insure the validity of the data. A spectrum analyzer will be used during the test as an additional check on transducer performance. The spectrum analyzer will also provide preliminary information on the natural frequencies and responses of the instrumented components. The majority of the data, however, will be analyzed from the magnetic tapes.

The leads for these internally mounted transducers will be routed through the top mounted instrumentation guide tube conduits. The combined in-core detectors/core exit thermocouples will not be installed during the hot functional test. Special fittings, designed to ASME Section III, Class 1 pressure boundary rules, will be used to seal the transducer leads during this test. These fittings will be removed following the test.

All transducers and associated hardware will be removed after the completion of the Hot Functional testing.

Location of Transducers

Transducer locations and their directions of sensitivity are listed in the table included in the SSAR Revision. The measurement objectives for the instrumented components are listed below:

1. Four radially sensitive accelerometers mounted near the top of the radial reflector. These transducers are to detect shell mode vibration of the radial reflector and provide additional information on the core barrel beam modes.

2. Six axially sensitive strain gages mounted just below the core barrel flange. These transducers will detect axial vibration of the lower internals and core barrel beam modes.

3. Two axially sensitive strain gages (one inside and one outside) mounted on the upper support assembly skirt to detect vertical motion of the upper support structure. Alternatively, this information may be obtained using axially sensitive accelerometers.

4. Four axially (2 inside and 2 outside) sensitive strain gages located on the core barrel to lower core plate weld. These strain gages will provide direct information on the stresses at this location. Alternatively, this information may be obtained using axially sensitive accelerometers.

5. Four axially sensitive strain gages mounted on two lower support columns that attach the vortex suppression plate to the lower support plate. These gages will be mounted at 90 degree separation on two different support columns such that lateral displacement of the vortex suppression plate assembly can be determined. Alternatively, four horizontally sensitive accelerometers will be considered to obtain this information.

6. Two axially sensitive strain gages located on the upper support column extension. These transducers will detect the lateral displacement of the extension.

7. Four vertically sensitive and two horizontally sensitive accelerometers mounted on the reactor head closure studs at 90° intervals. These transducers will detect motion of the reactor vessel and the upper and lower internals flanges.

8. Four radially sensitive accelerometers will be installed at the upper core plate elevation to determine the shell mode responses of the core barrel.

9. Four axially sensitive strain gages mounted on two guide tubes located near the exits of the hot legs to detect lateral defections.

The bases used to establish expected and acceptable vibration levels and expected natural frequencies are found in the vibration assessment program. The final values established for expected and acceptable levels will be established prior to the start of testing.

SSAR Revision:

Revise the last paragraph of Subsection 3.9.2.4 as follows:

The reactor internals flow-induced vibration measurement program will be conducted during preoperational tests of the first AP600. Transducers will be installed on the reactor vessel and the internals prior to the cold hydrostatic test. The response of the reactor and the internals due to flow-induced vibration will be measured during the hot functional test. Data will be acquired at several temperatures from cold startup to hot standby conditions. The location of the transducers is outlined in Table 3.9-4 The leads for the internally mounted transducers will be routed through the top mounted instrumentation guide tube conduits through special fitting that will be removed following the test.

The bases used to establish expected and acceptable vibration levels and expected natural frequencies are found in the vibration assessment program. The acceptance standards for the inspection of reactor internals before and after the hot functional testing are the same as required in the shop by the original design drawings and specifications.

(1) The table number in the SSAR will depend on the number of tables added for other RAI responses.

Add a table to Section 3.9 for the transducer locations for the first plant AP600 reactor internals vibration measurement program as follows:

Table 3.9-4
First Plant AP-600 Reactor Internals Vibration Measurement Program Transducer Locations

Instrumented Component	Number and Type of Transducers	Approximate Transducer Locations	Direction of Sensitivity
Radial Reflector (Inner Wall)	4 accelerometers	0°, 180°, 225°, 270°	Radial
Core Barrel Flange (Outer Wall)	4 strain gages	0°, 90° 180°, 270°	Axial
Core Barrel Flange (Inner Wall)	2 strain gages	180°, 270°	Axial
Core Barrel Mid-elevation	4 accelerometers	0°, 180°, 225°, 270°	Radial
Upper Support Skirt (Inside and Outside)	2 strain gages	180°	Axial
Lower Support Plate Weld (Inside and Outside)	4 strain gages	0°, 90°	Vertical
Vortex Suppression Plate Support Columns (2)	4 strain gages or	On column near lower core support plate	Axial
	4 accelerometers	or on vortex suppression ring	Horizontal
Reactor Vessel (Head Studs)	4 accelerometers	0°, 90°, 180°, 270°	Vertical
	2 accelerometers	0°, 90°	Horizontal
Support Column Extension	2 strain gages	0°, 90°	Axial
Guide Tube A-7	2 strain gages	90° apart	Axial
Guide Tube N-7	2 strain gages	90° apart	Axial

Question 210.118

Exposure that is as low as is reasonably achievable (ALARA) is a concern, especially for the longer 60-year life plant.

a. Compare ALARA exposure between the AP600 design and the present 40-year life design for the expected maintenance and inspection of the RCP flywheel.

b. Estimate the radiation exposure likely at the end of the 60-year life.

Response:

a. Due to the nature of the AP600 canned motor pump, no maintenance and inspection of the reactor coolant pump flywheel is planned or needed.

   For the present 40-year life design of the shaft seal pump, the only maintenance and inspection requirements for the flywheel is the need for the 3-year and 10-year inservice inspection. The estimated dose to perform the 3-year flywheel inspection is 0.1 man-rem per pump, or 1.3 man-rem in 40 years. The estimated dose to perform the 10-year inspection is 0.6 man-rem per pump, or 2.6 man-rem in 40 years. Thus, the total dose for the flywheel inspection is 3.9 man-rem per pump. The AP600 canned motor pump is then 3.9 man-rem less than a shaft seal pump.

b. As noted above, no maintenance or inspection of the flywheel is planned. However, a dose estimate has been made for the unlikely event of a need to disassemble the AP600 pump. The pump disassembly program includes removal of the pump and impeller from the pump casing, disassembly and dimensional inspection of rotating parts and parts that are subject to mechanical or fluid friction, and dye penetrant testing of the rotor journals. The expected dose to perform the inspection is 2.32 man-rem.

SSAR Revision: NONE

Revise the third paragraph of Subsection 12.4.1.4 as follows:

No special maintenance activities are forecast for the canned motor reactor coolant pumps. ~~However, for the first AP600 plant, additional inspection of the reactor coolant pumps is planned.~~

Delete Table 12.4-3, "Dose Estimate for Reactor Coolant Pump Inspection"

Revise Table 12.4-9 as follows:

Table 12.4-9

DOSE ESTIMATE FOR REACTOR COOLANT PUMP IN-SERVICE INSPECTION (10-YEAR INTERVAL)

Activity	Average Dose Rate (millirem/hr)	Crew Size (no. workers)	Time (hours)	Occupational Radiation Exposure (man-rem)
Perform actions to expose pump casing	55	2	6	0.660
Visual inspection of studs, nuts, and washers	5	2	2	0.020
~~UT scan and dye penetrant testing of studs~~	~~5~~	~~2~~	~~2~~	~~0.020~~
~~Remotely conduct visual inspection of casing internal surface~~	~~5~~	~~2~~	4	~~0.040~~
~~Install pump~~	~~25~~	~~2~~	~~6.25~~	~~0.313~~

Total RCP in-service inspection ORE = 0.68 ~~1.05~~ man-rem/10 years[(a)]
Annual RCP in-service inspection ORE = 0.7 ~~0.11~~ man-rem/year

(a) ~~The described in service inspection is not a scheduled activity but is performed when an RCP is removed for maintenance. This is expected to occur no more often then once in ten years. If the RCP is not removed for maintenance, in service inspection is limited to external visual inspection and UT scan.~~

Revise Table 12.4-10, by deleting the line and note related to reactor coolant pump inspection as follows:

Work Description	Annual Dose (man-rem)
~~RCP Inspection~~	~~0.8~~[(a)]

(a) ~~The 0.8 man rem associated with special RCP inspection activities is applicable only to the first operating AP600.~~

Question 210.127

Only "seal" welds are used to connect the component parts of the Inconel 600 flywheel enclosure. The seal welds are partial penetration butt welds with tensile loads at the root of the weld across the faying surfaces.

a. Demonstrate that the Inconel 600 enclosure weldments (of the enclosure's joint designs) have corrosion fatigue lives in reactor coolant at operating temperature for the 60-year design life.

b. The flex-foot design develops high tensile stresses at the bottom of the groove forming the foot. Accordingly, the design will also generate high tensile stresses at the root of the seal weld. Tensile stresses at the root of a fillet weld should be avoided. Although the Inconel 600 flywheel enclosure is not a core support and it could be argued that the weld design limitations of NG-3350 should not apply to this design, the acceptability of this design detail should be evaluated on the basis of meeting the goal of a 60-year life. If the requirements of NG-3350 were applied, this joint should be a Category C, Types I, II, III joint or Types IV or V with limitations. The design is such that the flex-foot design prevents the use of these type joints. These type joints prevent the development of high tensile stresses at the root of the weld with the discontinuity of the faying surfaces directly at the root. Address the staff's concern.

Response:

a. Analysis of the seal welds was performed and documented in Reference 210.127-1. Fatigue usage for specification design conditions is 0.24 with an allowable of 1.0. In aqueous environment, several authors have shown no significant effect on fatigue performance of Alloy 600. For example see Reference 210.127-2. The temperature of operation is 150°F where no environmental degradation has been observed for Alloy 600 in primary coolant environment. Short term temperature excursions above 150°F due to operational transients would not be expected to have a significant effect on fatigue performance of Alloy 600.

b. Westinghouse recognizes that tensile stresses at the root of a fillet weld should be avoided. The flex foot weld, however, is a partial penetration weld with a minimum throat thickness that exceeds the wall thickness of either of the two attaching members. The weld is sized per Figure NB-4244(b)-2(b) of the ASME Code, Section III.

   NG-3350 category type joints are required to carry primary loads, i.e., if excessive deformation were to occur, the joint would rupture. For the flex foot design, however, the stresses are self limiting.

   Use of the stress limits of ASME Code, Section III, Subsection NG, along with an appropriate fatigue strength reduction factor at the weld, will provide a 60-year design life.

Reference:

210.127-1 WCAP-13734, "Structural Analysis Summary for the AP600 Reactor Coolant Pump High Inertia Flywheel," (Proprietary), May, 1993.

210.127-2 Ogundele and King, "Corrosion Fatigue of Alloy N06600 Steam Generator Tubing", Fifth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems, August, 1991.

SSAR Revision: NONE

Question 440.255

What fraction of the core power is deposited in the reflector?

Response:

The fraction of core power absorbed in the radial reflector due to gamma ray absorption is approxiamately 0.0044.

SSAR Revision: NONE

Question 620.9

Describe how the results/findings of past experience/lessons learned are documented, and provide a sample of the documentation (Section 18.3, p. 18.3-2).

Response (Revision 1)

Reference 620.9-1 specifies a number of sources of operating experience including predecessor plant and systems, recognized industry HFE issues, related human-system interface (HSI) technology and operator interviews. The Westinghouse M-MIS design process draws on these sources of operating experience. The results of operating experience reviews are documented in a variety of ways ranging from trip reports that document feedback from operators on early versions of M-MIS designs, to formal reports focused on particular M-MIS issues that synthesize design recommendations based on examination of the human factors literature and experiences in the nuclear industry and in related industries.

~~Westinghouse has generated and submitted to the NRC WCAP 13559, "Operational Assessment for AP600", which summarizes the applicability of I.E. bulletins, generic letters, I.E. circulars, information notices, and AEOD reports. In addition, Westinghouse has prepared a conformance matrix to compare the AP600 to the ALWR URD requirements, representing the degree to which the AP600 complies with each requirement. Lessons learned for the project are incorporated through compliance with the URD requirements, which are based on industry experience and utility inputs, and by using NRC, INPO, and other industry documents summarizing lessons learned.~~

A part of the Westinghouse M-MIS design process is to review recognized HFE issues and related HSI technology as input to the M-MIS design bases and functional requirements. Recent examples include a review of issues associated with navigating through large display networks, a review of issues associated with design of soft controls, and a review of issues associated with the design of a Wall Panel Information System to support group situation awareness.

Westinghouse conducts operator interviews as part of the M-MIS design process. These tend to focus on specific human performance issues in support of the design and evaluation of M-MIS resources. Examples include an expert panel session on feedwater control during startup that was performed as part of the development of the automatic feedwater control system; interviews with operators that are currently being performed to elicit user input on the M-MIS products being developed for that customer; and interviews of fossil power plant operators that were conducted on issues and lessons learned associated with the use of soft controls. We will also elicit operations input to the AP600 M-MIS design from a standard user's group of utility operators who will provide feedback on proposed design concepts and participate in concept tests.

Westinghouse reviews industry and NRC reports documenting operating experience with predecessor plant and systems. Recent examples include a review of Reference 620.9-2. This INPO report reviews significant operating experience reports (SOERs) and significant event reports (SERs) to identify problems and lessons learned that should be addressed during the advanced light water reactors design process. As part of the Westinghouse effort to incorporate lessons learned into the AP600 design process, engineering design groups reviewed the relevant portions of the INPO document. In addition, the INPO report was reviewed by a human factors specialist to

identify generic issues raised that need to be considered in the design of the AP600 man-machine interface systems (M-MIS). Reference 620.9-3 was generated to document the results of the multi-disciplinary review of the INPO document.

Another example of a review of operating experience documents is a Westinghouse report that examined NRC documents on operator performance in actual events. Reference 620.9-4 examines the factors that appeared to contribute to human performance problems in these events and the implications for the design of M-MIS resources for the AP600. The NRC documents reviewed included References 620.9-5 through 620.9-8.

References:

620.9-1 O'Hara, J.M.; Higgins, J.C.; Stubler, W.F. (1994). *Human Factors Engineering Program Review Model for Advanced Nuclear Power Plants.* Prepared for NRC by Brookhaven National Laboratory

620.9-2 INPO 93-004, Revision 2, April 1994. *Operating Experience To Apply To Advanced Light Water Reactor Designs.*

620.9-3 WCAP-14115, Revision 0, July 1994. *Review of Nuclear Plant Operating Experience and the Application To the AP600 Design.*

620.9-4 WCAP-14114, Revision 0, 1994. Roth, E.M. *Human Performance in Operating Events: Lessons Learned for the AP600 M-MIS Design.*

620.9-5 Kauffman, J. V., G. F. Lanik, E. A. Trager, and R. A. Spence. *Operating Experience Feedback Report - Human Performance in Operating Events,* NUREG-1275, Office for Analysis and Evaluation of Operational Data, U. S. Nuclear Regulatory Commission, Washington, D. C., December, 1992.

620.9-6 NRC, NUREG-1455, *Transformer Failure and Common-Mode Loss of Instrument Power at Nine Mile Point Unit 2 on August 13, 1991.* U. S. Nuclear Regulatory Commission, Washington, DC 20555, October, 1991.

620.9-7 NRC, *NRC Augmented Inspection Team Exit Meeting Presentation for Salem Unit 1 Reactor Trip with Multiple Safety Injections*, handout, April 26, 1994.

620.9-8 Wreathall, J., Reason, J. and Dougherty, Jr. E. M. *Latent Failures and Human Performance in Significant Operating Events,* draft report prepared for Division of Systems Research Office of Nuclear Regulatory Research U. S. Nuclear Regulatory Commission, July, 1993.

SSAR Revision: NONE

Question 620.15

What are the tools, techniques, etc. used by the team to fulfill their responsibilities? Describe the program Westinghouse has to track and document HFE-related problems/concerns/issues and their solutions throughout the design program (Section 18.4).

Response (Revision 1)

The Westinghouse M-MIS design team uses UNIX and DOS-based software to design data bases ~~and to write design documentation. The design team will use these tools to create various data bases;~~ to structure the design of the alarms, displays, and procedures; and to write design specifications for implementation by software designers. The design is an interfacing process, since many of the constraints of other disciplines play a role in ~~honing~~ the M-MIS. ~~solution, and of course, the human factors designers have to work within the constraints of the overall AP600 project.~~ Other descriptions of tools used by designers, including part-task simulation, are discussed in SSAR Subsection 18.8.2.3.1.7 and throughout the V&V discussions in Chapter 18.

Westinghouse does not view the M-MIS design as a design with independent and separate design program operating procedures. The M-MIS design is considered a part of the overall plant design. It is controlled by the same policies and procedures as the overall plant design. Two methods are used to identify, track and resolve design concerns and issues, including human factors (HFE) issues. In this manner, HFE issues are addressed in the same way as the other disciplines. The two methods used are the Design Configuration Change Control process and the Design Review process.

Review packages are prepared for all proposed design changes that are more significant (cost impact, any impact on safety, SSAR, or PRA). These packages contain impacts of change from affected functional groups (including the Man-Machine design group). A configuration control board (CCB) reviews the proposed design changes. The CCB is a board of individuals drawn from various organizations and disciplines, including one member from the M-MIS design team. The DCPs are maintained in a computerized data base that permits the tracking of the status of each DCP.

The second method of identifying and tracking design issues is through the Design Review (DR) process. Formal DRs include relevant design disciplines. The M-MIS team is represented on each design review board. WCAP 9817 describes the DR process, required responsibilities of participants and required documentation. Part of the DR data package consists of a set of checklists used by the review committee to guide their efforts. One of these checklists is a Human Factors checklist. For each design issue identified through the use of any checklist, action items are identified and documented. The design review is not considered complete until all action items are closed. A DR report is prepared to provide formal documentation of the findings and recommendations of the DR committee. It provides documentation of the DR meetings and resolution of all action items.

~~Anyone on the M-MIS design team can create a Design Change Proposal (DCP) in accordance with the AP600 procedures. In addition, all DCPs, plant design as well as M-MIS design, are reviewed by at least one member~~

~~of the M-MIS design team, and the impact is reviewed by M-MIS management. In this way, human factors issues are addressed and resolved in the same way that other disciplines are. The DCPs are maintained in a computerized data base that permits the tracking of the status of each DCP. In addition, plant design data will be maintained in the IMS data base for the project.~~

~~Eventually, as the design progresses, all plant mechanical, electrical, and I&C design documentation will be reviewed by M-MIS design team members in order to incorporate the design details into the M-MIS design.~~

~~Formal design reviews will be held for all design disciplines, and an M-MIS team member will act on the design review board for all disciplines. A human factors check list is part of the Westinghouse design review process. Design reviews for the M-MIS will also be held in the required design stages to ensure that the design does not progress too far without an appropriate review cycle to keep the design on track.~~

SSAR Revision: NONE

Question 620.72

Section 18.8.2.1.2.4 of the SSAR indicates that tasks are initially allocated to human or automatic subsystems based upon considerations of relative capabilities of each (as described in Appendix B of NUREG-0700). In addition, several factors influencing the function allocation process are identified, such as general philosophy, cost, technology readiness, and control philosophy. (Section 18.8.2.1.2.4)

a. Explain how these factors combine to make allocation decisions. Is a "formal" process such as that described in IAEA-TECDOC-668 (The role of automation and humans in NPPs) or NUREG/CR-3331 (A methodology for allocation nuclear power plant control functions to human and automated control) used to guide and structure the allocation process? Describe the "formal" process used.

b. Explain why regulatory requirements (mandatory allocations) and operating experience are not included on the process.

c. How are allocation decisions and their technical basis documented so that technology and future research can be used to improve allocations limited by factors such as technology readiness?

Response: (Revision 1)

The formal method for function allocation described in IAEA-TECDOC-668, *The Role of Automation and Humans in Nuclear Power Plants*, serves as a model for the allocation of functions in the AP600.

The process begins with an identification of objectives for overall system performance. A critical element of this initial phase is to specify the role that humans are intended to serve in plant control. SSAR Subsection 18.6.6 (Role of the Operator in the AP600 Main Control Room) lays the foundation for the desired role of humans in the AP600 control room. Other major influences on system design and function allocation are identified by the consideration of "influencing factors." Here, operating experience, regulatory requirements (mandatory allocations), safety factors, economic factors, societal factors, and technological factors are brought into the allocation process.

The function assignment process begins by identifying the functions for which automation is mandatory--determined by regulation, policy, or an assessment of human performance limitations. Functions for which human control is mandatory are also identified. Functions for which there is a strong preference in assignment, though not a mandatory allocation, are identified. Finally, schemes for sharing functions between humans and automation are identified. Many functions, though automated to some degree, will still require some level of input from human controllers. As human involvement decreases and automation increases, control moves through the following stages (see Reference 620.72-2):

1. Direct manual control - Human has direct authority over function.

2. Assisted manual control - Human still has direct authority over function, but information is provided by displays to aid in control.

3. Shared control - Human has control but automated system may limit range of control. Advisory systems or limits on controls by an automated system guide and limit human control.

4. Management by delegation - Human determines what will be done and when, and then allows automation to carry out parts of it.

5. Management by consent - Automation handles most of the task, but human must consent or allow major state changes. Still may be manual execution of critical actions.

6. Management by exception - Automation handles all of the task, informs human of intent and actions. Human must consent to critical actions and can take manual control if desired.

7. Autonomous operation - Fully autonomous operation, and human is not usually informed. Human monitors only to detect faults or problems.

Some of these options represent forms of dynamic allocation.

The initial set of allocation decisions--whether human, automation, or shared--is documented as the hypothesized assignments to indicate the major factors and influences that led to the decision. Some of these initial automation assumptions are documented in the respective System Specification Documents (SSDs) for a given system. From this point, the allocation plan can be evaluated on several dimensions to determine how well overall system performance is supported. Initially, evaluation will focus on an assessment of the human's role in realistic tasks. Primarily, a determination will be made of the workload imposed on the human. A workload that is too small is almost as bad as an excessive workload. Also, the overall role of the human as defined by the allocation process can be compared to the philosophy of the human's role. The desired operator's role needs to be preserved through the allocation process.

Other evaluations will also take place through the man-in-the-loop test plan. Even more critical than initial allocation of functions, however, is the manner in which automated systems are designed, assuming that many functions will be shared. A number of researchers (see References 620.72-1 through 620.72-3) have identified problems with interactions between humans and automated systems in other domains, especially commercial aviation. The potential problem is one of communication or feedback between the human and the automated process. The author of Reference 620.72-2 has defined a set of characteristics that should be given to automated systems:

1. Accountable - The automated system must inform the operator of its actions and be able to explain them on request.

2. Subordinate - Except in predefined situations, the automated system should never assume command. In those situations, it must be able to be countermanded easily.

3. Predictable - The automated system must conform to expectations about its behavior to maintain trust.

4. Adaptable - The automated system should be configurable within a wide range of operator preferences and needs.

5. Comprehensible - The automated system should be intelligible and easy to understand.

6. Flexible - The automated system should be tractable and exhibit a ready ability to adapt to new, different, or changing requirements for human control inputs (level of automation can be set according to operator's skill level).

7. Dependable - It should do, dependably, what it is ordered to do. It should never do what it is ordered not to do. It must never make the situation worse.

8. Informative - It must keep the operator informed.

9. Error-resistant - It must keep operators from committing errors wherever that is possible.

10. Error-tolerant - Some errors will occur, even in a highly error-resistant system. Automation must detect and mitigate the effect of these errors.

For the AP600 plant, some allocation decisions have already been made (automated tasks as found in Chapter 7 of the SSAR) without the benefit of a function based task analysis and the task allocation process. It is acceptable to base the initial task allocations of a new design on a predecessor design (Reference: Element 3 of the Human Factors Engineering Program Review Model For Advanced Nuclear Power Plants, January 25, 1994). The task allocation step of the function based task analysis will verify the design where preliminary decisions of automation have been made and will allocate functions where they have not.

While this guidance is presented at a very high level, it expresses important goals for the design and development of automation. We will incorporate the lessons learned from these studies when appropriate in order to improve the design of automated systems.

As an evaluation is completed, function allocation will be addressed iteratively to correct problems, reduce the likelihood of error, and enhance overall performance. Each modification will be reflected in the justification and documentation section.

References:

620.72-1 Bainbridge, L. (1987). Ironies of automation. In J. Rasmussen, K. Duncan, and J. Leplat (eds.), *New technology and human error*. New York: Wiley.

620.72-2 Billings, C.E. (1991). *Human-centered automation: A concept and guidelines*. (NASA TM 103885). Moffett Field, CA: NASA Ames Research Center.

620.72-3 Norman, D.A. (1989). *The problem with automation: Inappropriate feedback and interaction, not "over-automation."* Paper presented at the Discussion Meeting, Human Factors in High-Risk Situations, The Royal Society.

620.72-4 International Atomic Energy Agency (IAEA), *The Role of Automation and Humans in Nuclear Power Plants*, IAEA-TECDOC-668, October, 1992.

620.72-5 O'Hara, J.M.; Higgins, J.C.; Stubler W.F. *Human Factors Engineering Program Review Model For Advanced Nuclear Power Plants*, January 25, 1994.

SSAR Revision: NONE