UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 23, 1998

The Honorable Shirley Ann Jackson
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Jackson:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE WESTINGHOUSE ELECTRIC COMPANY APPLICATION FOR CERTIFICATION OF THE AP600 PASSIVE PLANT DESIGN

During the 454th meeting of the Advisory Committee on Reactor Safeguards, July 8-10, 1998, we completed our safety review of the Westinghouse Electric Company application for certification of its AP600 passive plant design. This report is intended to fulfill the requirement of 10 CFR 52.53 that "the ACRS shall report on those portions of the application which concern safety." During our review, we had the benefit of discussions with representatives of Westinghouse and its consultants, and the NRC staff. We also had the benefit of the documents referenced.

AP600 Application

On June 26, 1992, Westinghouse tendered its application to the NRC for certification of the AP600 design. This application was submitted in accordance with Subpart B, "Standard Design Certifications," of 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," and Appendix O, "Standardization of Design: Staff Review of Standard Designs." The application was docketed on December 31, 1992, and assigned Docket Number 52-003.

The application consists of the AP600 Standard Safety Analysis Report (SSAR), the Tier 1 Material, and the probabilistic risk assessment (PRA). On June 26, 1992, Westinghouse submitted the SSAR and the PRA. In December 1992, Westinghouse submitted the Tier 1 Material, which contains inspections, tests, analyses, and acceptance criteria (ITAAC) and Tier 1 design descriptions. Design certification is sought for the power generation complex, excluding those elements and features considered site-specific. All safety-related structures, systems, and components (SSCs) are located on the nuclear island and are to be included in the design certification.

Three aspects of the plant design (i.e., instrumentation and control (I&C) systems, human factors engineering, and some piping) will be completed by the combined license (COL) applicant using the design processes described in the SSAR and ITAAC.

The staff issued a Draft Safety Evaluation Report (DSER) on November 30, 1994, a supplement to the DSER in April 1996, and an Advance Final Safety Evaluation Report on May 2, 1998. Our activities related to the review of the AP600 design are listed in the Attachment. As a result of our review, we issued three interim letters identifying several issues. The resolution proposed by Westinghouse to these issues is acceptable, pending staff review and approval.

AP600 Design Description

The AP600 plant is designed for use at either single-unit or multiple-unit sites. The scope of the design is complete except for site-specific elements. The AP600 design has a nuclear steam supply system rating of 1933 MWt, with an electrical output of at least 600 MWe. The plant has a design objective of 60 years without a planned replacement of the reactor vessel. The design does provide, however, for the replacement of other major components, including the steam generators.

The primary objective of the AP600 design is to meet safety requirements and goals defined for advanced light-water reactors with passive safety features as specified in the Electric Power Research Institute Utility Requirements Document. An additional objective is to provide a greatly simplified plant with respect to design, licensing, construction, operation, inspection, and maintenance.

The plant arrangement consists of five principal structures; the nuclear island, the turbine building, the annex building, the diesel generator building, and the radwaste building.

The nuclear island, which includes all safety-related or seismic Category I structures, is designed to withstand the effects of natural phenomena and postulated events. It consists of a containment building, a concrete shield building, and an auxiliary building, which are described below.

- The containment building consists of a free-standing steel containment vessel which has a design pressure of 45 psig and associated internal structures. The vessel performs the function of containing the release of radioactivity to the atmosphere following postulated design-basis accidents. The vessel is also part of the passive containment cooling system.

- The shield building comprises the structure and annulus area that surrounds the containment building. In the event of an accident, the passive containment cooling system releases water that runs down the outside of the containment vessel to enhance heat removal.

- The auxiliary building is designed to provide protection and separation for the seismic Category 1 mechanical and electrical equipment located outside the containment building. The building also provides protection for safety-related equipment against the consequences of internal or external events. The main control room, Class 1E I&C systems, Class 1E electrical systems, and reactor fuel handling area are contained in the auxiliary building.

The turbine building houses the main turbine generator and associated fluid and electrical systems. The annex building includes the health physics area, the technical support center, access control, and personnel facilities. The diesel generator building houses two diesel generators and their associated support systems. The radwaste building contains facilities for the handling, processing, and storing of radioactive wastes.

The overall plant arrangement utilizes building configurations and structural designs to minimize the building volumes and quantities of bulk materials (concrete, structural steel, rebar) consistent with safety, operational, maintenance, and structural needs. The plant arrangement provides separation between safety and nonsafety equipment and systems to preclude adverse interactions among them. Separation between redundant safety equipment and systems provides confidence that the safety functions can be performed. In general, this separation is provided by concrete walls.

The ITAAC program is intended to ensure that the plant, when built, conforms to the design parameters and assumptions that existed at the time of design certification. For example, the efficacy of the passive emergency core cooling system depends on the flow resistances of piping segments, relief valves, and other components. The flow resistances will be measured in the as-built plant to ensure that they conform with the values derived and validated by the test and analysis program.

Safety Enhancement Features

The AP600 design contains many features that are not found in current operating plants. For example, a variety of engineering and operational improvements provide additional safety margins and comply with the Commission's Severe Accident, Safety Goal, and Standardization Policy Statements. Unique features of the AP600 design include an improved reactor core design, a large reactor vessel, a large pressurizer, an in-containment refueling water storage tank (IRWST), an automatic depressurization system, a digital microprocessor-based I&C system, hermetically sealed canned rotor coolant pumps mounted to the steam generator, and increased battery capacity.

The AP600 design represents a significant departure from previous commercial nuclear reactor technology in that it places more dependence on passive systems for accident response. Passive systems depend on gravity, condensation, and small pressure differences to prevent or mitigate damage to the core and to ensure containment of radioactive fission products in the event of accidents. Active systems, on the other hand, employ flow loops and pumps that require electrical or other sources of motive power. The performance of active systems is, in general, better known because of existing test data and extensive operating experience. Passive systems, although not tested under full-scale conditions, are more likely to ensure safety functions, especially under conditions where external or emergency motive power could be compromised.

The AP600 I&C systems are significantly different from those in current operating plants. The primary differences result from using software-based digital systems with multiplexed and fiber optics data links in place of the analog systems. The use of digital systems with multiplex and fiber optics data links reduces the amount of cabling in the plant, thereby reducing configuration complexity and fire hazards.

The AP600 design does not require Class 1E electrical power except that provided by the Class 1E dc batteries and their inverters. This feature significantly reduces the complexity of the plant electrical systems and the reliance on safety-grade diesel generators.

The AP600 plant includes an innovative security plan which features the use of defensive capabilities at various vital area access points. This feature results in elimination of the protective area boundary and associated security attributes used at current operating nuclear power plants.

AP600 Test and Analysis Program

Westinghouse conducted an extensive test and analysis program, utilizing separate-effects and integral-system facilities both to investigate the behavior of the AP600 passive safety systems and to develop a database for validation of the computer codes used to perform accident and transient analyses. Key aspects of the test and analysis program include:

- Core Makeup Tank (CMT) Test Program to characterize the CMT over an extended range of thermal-hydraulic conditions.
- Automatic Depressurization System (ADS) Test Program, both to characterize the steam flow through the IRWST sparger and to test the thermal-hydraulic behavior of the ADS piping network.
- Passive Residual Heat Removal (PRHR) System Test Program to generate data for design and characterization of the AP600 PRHR heat exchanger.
- Oregon State University Advanced Plant Experiment (APEX) Test Program to obtain integral-systems data for code validation; emphasis was placed on low-pressure and long-term core cooling behavior for design-basis, small-break loss-of-coolant accidents (LOCAs).
- SPES-2 High-Pressure, Full-Height Integral-Systems Test Program to obtain integral-systems data for code validation; the particular focus was on accident progression from initiation to establishment of stable IRWST injection.
- Passive Containment Cooling System Test Program to obtain integral-systems test data on the thermal-hydraulic performance of this system to support code validation.

This extensive test and analysis program was necessary to validate the accident analysis codes applied to new, passive emergency core cooling systems for which there is not a significant experience base. The accident analysis codes used by Westinghouse included:

- LOFTRAN/LOFTTR2 for analyses of non-LOCA transients
- NOTRUMP for evaluation-model analyses of small-break LOCAs
- WCOBRA/TRAC for best-estimate analyses of large-break LOCAs
- WCOBRA/TRAC for analyses of long-term core cooling
- WGOTHIC for design-basis accident analyses of the containment

To ensure that the test and analysis program adequately addressed important phenomena with respect to the passive systems and that the results would scale to the prototype size, Westinghouse developed a phenomena identification and ranking table and performed a scaling analysis for both the primary coolant system and the containment.

In addition, the NRC staff performed confirmatory experimental and analytical programs in support of the AP600 design certification review. These programs included the integral-systems testing performed at the Japan Atomic Energy Research Institute ROSA-AP600 facility, and follow-on testing performed at the Oregon State University APEX facility. The NRC staff also performed confirmatory analyses utilizing the NRC codes RELAP-5 and CONTAIN. The results of the staff's programs significantly aided our review of the Westinghouse test and analysis program.

During the extensive reviews of the Westinghouse test and analysis program, we raised numerous issues. These issues have been documented in our interim letters and meeting minutes. Based on discussions with representatives of Westinghouse and the NRC staff, all of our issues pertaining to the Westinghouse test and analysis program have been adequately resolved.

There are, however, a number of issues that arose during our review that, while not directly affecting the acceptability of the AP600 test and analysis program, should be considered in the context of future design certification reviews. We plan to address these issues in a future letter pertaining to lessons learned from the AP600 design certification review.

Probabilistic Risk Assessment

The AP600 design certification application included a PRA, in accordance with regulatory requirements. This PRA was done well and rigorous methods were used to quantify risk metrics, including core damage frequency (CDF) and large, early release frequency (LERF). Point estimates of the risk metrics are:

CDF = 2×10^{-7} per reactor year
LERF = 2×10^{-8} per reactor year

These risk metrics are low compared to those estimated for existing nuclear power plants. The PRA was an integral part of the design process. This contributed significantly to design modifications, which resulted in the low CDF and LERF.

The PRA addressed passive safety systems and software-based digital I&C systems. Qualitative analyses and extensive sensitivity studies were used to compensate for incomplete modeling of these important features of the plant. In addition, the concept of the "focused" PRA was introduced to reduce uncertainties in the estimated performance of passive systems. The objective of the "focused" PRA was to determine whether the goals for CDF and LERF could be met without the support of the nonsafety-related systems. The regulatory treatment of nonsafety systems (RTNSS) process was used to impose special requirements on some nonsafety systems to ensure, with high confidence, that they would be available when needed. For example, Westinghouse used the RTNSS process to impose administrative controls on the availability of the engineered safety feature actuation function of the diverse actuation system in order to reduce uncertainties associated with the digital system software. The RTNSS process is an excellent example of a good risk-informed and performance-based approach.

We applaud the use of the "focused" PRA and the RTNSS process in developing defense-in-depth measures. But, we caution against establishing the practice of comparing the results of "focused" PRAs with Safety Goals. These Goals apply to a plant as it is designed and operated. Comparison of these Goals with results of analyses, restricted to include only safety systems, would amount to the imposition of a new goal that does not appear in the Commission's Safety Goal Policy Statement.

Additional Observations

Westinghouse's approach for quantifying digital systems software in the PRA is consistent with the guidance in Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems in Nuclear Power Plants." This approach provides a method for identifying and assessing design strengths and weaknesses.

The AP600 plant will use passive autocatalytic recombiners to maintain hydrogen concentrations below the flammability limit within the containment following design-basis accidents. We agree, in principle, that these devices are improvements over hydrogen recombiners used in existing plants. The COL applicant is responsible for qualifying passive autocatalytic recombiners. The present regulatory requirements for qualifying mechanical equipment are insufficient to ensure continued passive autocatalytic recombiner operation for the expected duty cycle.

The AP600 reactor containment is a steel shell. It has been designed to meet Service Level C of the ASME Boiler and Pressure Vessel Code. The containment meets all regulatory requirements. Testing has shown that steel shell containments are susceptible to catastrophic failure when overpressurized. For the AP600 design, however, under the peak pressure calculated in the Level 2 PRA for severe accident conditions, the probability of failure of the containment is estimated to be approximately 0.01. Deformation of the pressurized containment vessel and its interaction with the shield building could also induce leakage and further reduce the likelihood of failure. In any event, we have not been able to identify significant risks associated with possible catastrophic failure modes of the AP600 containment.

Westinghouse has concluded that external reactor vessel cooling will prevent core debris from penetrating the reactor vessel. This conclusion is based on a scenario for degradation of the core that avoids consideration of direct contact by metallic core debris with the reactor vessel. The NRC staff has concluded that reactor vessel failure is not precluded and has required that Westinghouse consider ex-vessel core debris interactions. Westinghouse performed these evaluations and found that the AP600 containment performs satisfactorily under these severe conditions.

ACRS Conclusion Concerning AP600 Design

Based on our review of those portions of the AP600 application which concern safety, we believe that acceptable bases and requirements have been established to ensure that the AP600 design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public.

Dr. Thomas S. Kress did not participate in the Committee's deliberation regarding external reactor vessel cooling.

Dr. Dana A. Powers did not participate in the Committee's deliberation regarding the AP600 source term or the results of Sandia National Laboratories tests on containment structural integrity and on environmental qualification of passive autocatalytic recombiners.

Dr. George Apostolakis did not participate in the Committee's deliberation regarding the AP600 passive system reliability assessment or the analyses performed by the Idaho Engineering and Environmental Laboratory concerning the use of the WCOBRA/TRAC code and external reactor vessel cooling.

Sincerely,

R. L. Seale

R. L. Seale
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Advance Final Safety Evaluation Report Related to the Certification of the AP600 Design," May 1998 (Predecisional Information).
2. U.S. Department of Energy Report DE-AC03-90SF18495, dated June 26, 1992, prepared by Westinghouse Electric Corporation, "AP600 Standard Safety Analysis Report," updated through Revision 23 (issued May 18, 1998).
3. U.S. Department of Energy Report DE-AC03-90SF18495, dated June 26, 1992, prepared by Westinghouse Electric Company, "AP600 Probabilistic Risk Assessment," updated through Revision 11 (issued March 1998).
4. U.S. Department of Energy Report DE-AC03-90SF18495, December 1992, "AP600 Tier 1 Material," updated through Revision 5 (issued May 7, 1998).
5. Letter dated February 19, 1998, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: Interim Letter on the Safety

Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Plant Design.

6. Letter dated April 9, 1998, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: The Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Plant Design-Interim Letter 2.
7. Letter dated June 15, 1998, from R. L. Seale, Chairman, ACRS, to L. Joseph Callan, Executive Director for Operations, NRC, Subject: The Safety Aspects of the Westinghouse Electric Company Application for Certification of the AP600 Plant Design-Interim Letter 3.
8. U.S. Nuclear Regulatory Commission, Draft NUREG-1512, "Draft Safety Evaluation Report Related to the Certification of the AP600 Design," November 1994.
9. U.S. Nuclear Regulatory Commission, Supplement to the "Draft Safety Evaluation Report Related to the Certification of the AP600 Design," April 1996.
10. Westinghouse Electric Corporation, WCAP-14812, Revision 2, "Accident Specification and Phenomena Evaluation for AP600 Passive Containment Cooling System," April 1998 (Proprietary).
11. Westinghouse Electric Corporation, WCAP-14845, Revision 3, "Scaling Analysis for AP600 Containment Pressure During Design Basis Accidents," March 1998 (Proprietary).
12. Westinghouse Electric Corporation, WCAP-14326, Revision 3, "Experimental Basis for the AP600 Containment Vessel Heat and Mass Transfer Correlations," May 1998 (Proprietary).
13. Westinghouse Electric Corporation, WCAP-14135, Revision 1, "Final Data Report for PCS Large-Scale Tests, Phase 2 and Phase 3," April 1997 (Proprietary).
14. Westinghouse Electric Corporation, WCAP-14382, Revision 0, "WGOTHIC Code Description and Validation," May 1995 (Proprietary).
15. Westinghouse Electric Corporation, WCAP-14407, Revision 3, "WGOTHIC Application to AP600," April 1998 (Proprietary).
16. Office of Nuclear Regulatory Research (RES) Report, RPSB-98-04, "Phenomenology Observed in the AP600 Integral Systems Test Programs Conducted in the ROSA-AP600, APEX, and the SPES-2 Facilities," D. Bessette, RES, M. DiMarzo, University of Maryland, P. Griffith, Massachusetts Institute of Technology, April 1998.
17. Letter dated July 1, 1998, from B. McIntyre, Westinghouse, to NRC: Attention: J. Larkins, ACRS, transmitting Response to ACRS Request for NOTRUMP Break Area Sensitivity Study.
18. Letter dated July 2, 1998, from B. McIntyre, Westinghouse, to NRC: Attention: J. Larkins, ACRS, Subject: Responses to ACRS Reactor Coolant System Issues.
19. Letter dated June 25, 1998, from B. McIntyre, Westinghouse, to NRC: Attention: J. Larkins, ACRS, Subject: Closure of ACRS Thermal Hydraulic Subcommittee Items for June 11-12, 1998 Meeting.

Attachment: Chronology of the ACRS Review of the Westinghouse Application for the AP600 Passive Plant Design Certification

ATTACHMENT

CHRONOLOGY OF THE ACRS REVIEW OF THE WESTINGHOUSE APPLICATION
FOR THE
AP600 PASSIVE PLANT DESIGN CERTIFICATION

The extensive ACRS review of the AP600 design and its interactions with representatives of the NRC staff and Westinghouse are discussed in the minutes of the following ACRS meetings. The questions raised by ACRS members during meetings which were not formally documented in ACRS reports and letters were answered during subsequent discussions.

ACRS MEETING/DATES	SUBJECT
Thermal Hydraulic Phenomena 12/17/91	Proposed Commission Paper on Need for Full-Height, Full-Pressure Integral System Testing of AP600 Design
Thermal Hydraulic Phenomena 3/3/92	Integral System Testing Requirements for AP600 Design
Thermal Hydraulic Phenomena 6/23-24/92	Integral System Testing Requirements for AP600 Design
Thermal Hydraulic Phenomena 3/4-5/93	Office of Nuclear Regulatory Research (RES) RELAP5/MOD3 Code
Thermal Hydraulic Phenomena 7/22-23/93	Westinghouse Test and Analysis Program (TAP)
Thermal Hydraulic Phenomena 9/21/93	TAP - Oregon State University APEX Test Facility
Thermal Hydraulic Phenomena 10/28/93	RES - ROSA-V (ROSA-AP600) Confirmatory Test Program
Thermal Hydraulic Phenomena 1/4-5/94	RES - RELAP5/MOD3 Code
Thermal Hydraulic Phenomena 3/15-16/94	TAP - Core Makeup Tank Test Facility, Passive Containment Cooling System
Thermal Hydraulic Phenomena 5/18-19/94	TAP - WCOBRA/TRAC Code

Thermal Hydraulic Phenomena 8/25-26/94	RES - Confirmatory Test Programs
W Standard Plants Designs 1/11/95	Overview and General Description of the AP600 Plant Design
Thermal Hydraulic Phenomena 2/15-16/95	TAP - WCOBRA/TRAC Code
Thermal Hydraulic Phenomena 3/27-28/95	RES - Phenomena Identification and Ranking Table (PIRT) for RELAP5 Code
Thermal Hydraulic Phenomena 3/29-30/95	TAP - Passive Containment Cooling System
W Standard Plant Designs 5/31/95	Commission Paper on Status of Ten Key Technical and Policy Issues
Thermal Hydraulic Phenomena 7/26-27/95	Qualification Document for the WCOBRA/TRAC Code
Thermal Hydraulic Phenomena 1/18-19/96	Qualification Document for the WCOBRA/TRAC Code
Thermal Hydraulic Phenomena 2/22-23/96	RES Program for Demonstrating Adequacy of the RELAP5/MOD3 Code to Assess Behavior of AP600 Design
Thermal Hydraulic Phenomena 5/9-10/96	TAP - Overview
Severe Accidents 6/5/96	Probabilistic Risk Assessment of Severe Accidents
W Standard Plant Designs 7/19/96	SECY-96-128, "Policy and Key Technical Issues Pertaining to the AP600 Design"
433rd ACRS Meeting 8/8/96	SECY-96-128, "Policy and Key Technical Issues Pertaining to the AP600 Design" ACRS Report Issued 8/15/96
W Standard Plant Designs 12/4/96	Chap. 4: Reactor Chap. 5: Reactor Coolant System and Connected Systems Chap. 9: Auxiliary Systems Chap. 11: Radioactive Waste Management

Thermal Hydraulic Phenomena 12/18-19/96	TAP - Scaling and PIRT Closure Report
Thermal Hydraulic Phenomena 2/12-14/97	RES Program for Demonstrating Adequacy of the RELAP5/MOD3 Code to Assess Behavior of AP600 Design
Thermal Hydraulic Phenomena 2/19/97	RES - ROSA-AP600 Confirmatory Test Program
Thermal Hydraulic Phenomena 3/28/97	TAP - Long-Term Cooling with WCOBRA/TRAC Code
442nd ACRS Meeting 6/13/97	AP600 Containment Spray System ACRS Report issued 6/17/97
Thermal Hydraulic Phenomena 7/29-30/97	TAP - NOTRUMP Small-Break LOCA Code
Thermal Hydraulic Phenomena 9/29-30/97	TAP - Passive Containment Cooling System
Thermal Hydraulic Phenomena 12/9-10/97	TAP - PIRT; Scaling of Reactor Coolant System; NOTRUMP Code
Thermal Hydraulic Phenomena 12/11-12/97	TAP - WGOTHIC Containment System Code
Advanced Reactor Designs 2/3-4/98	Chap. 7: Instrumentation and Controls Chap. 8: Electrical Power Chap. 13: Conduct of Operations Chap. 18: Human Factors Engineering
448th ACRS Meeting 2/5/98	TAP Chap. 1: Introduction and General Discussion Chap. 4: Reactor Chap. 5: Reactor Coolant System and Connected Systems Chap. 7: Instrumentation and Controls Chap. 8: Electrical Power Chap. 11: Radioactive Waste Management Chap. 13: Conduct of Operations Chap. 18: Human Factors Engineering Interim ACRS letter issued 2/19/98
Advanced Reactor Designs 3/30 - 4/1/98	Chap. 2: Site Characteristics Chap. 9: Auxiliary Systems

	Chap. 10: Steam and Power Conversion Chap. 12: Radiation Protection Chap. 13: Conduct of Operations (Security) Chap. 15: Accident Analyses
451st ACRS Meeting 4/2/98	TAP Chap. 2: Site Characteristics Chap. 9: Auxiliary Systems Chap. 10: Steam and Power Conversion Chap. 12: Radiation Protection Chap. 13: Conduct of Operations (Security) Chap. 15: Accident Analyses Interim ACRS Letter 2 Issued April 9, 1998
Thermal Hydraulic Phenomena 5/11-12/98	TAP - Primary Coolant System
Advanced Reactor Designs 5/13-15/98	Chap. 1: Introduction and General Discussion Chap. 6: Engineered Safety Features Chap. 14: Initial Test Program Chap. 16: Technical Specifications Chap. 17: Quality Assurance Levels 2 and 3 PRA Regulatory Treatment of Nonsafety Systems
453rd ACRS Meeting 6/3/98	TAP Chap. 3: Design of Structures, Components, Equipment, and Systems Chap. 6: Engineered Safety Features Chap. 9: Appendix A - Fire Protection Analysis Chap.14: Initial Test Program Chap.16: Technical Specifications Chap.17: Quality Assurance PRA Interim ACRS Letter 3 Issued June 15, 1998
Thermal Hydraulic Phenomena 6/11-12/98	TAP - Passive Containment Cooling System
Advanced Reactor Designs 6/17-18/98	ITAAC; Level 1 PRA; Adverse Interaction Evaluation Report; and Containment Spray System
Advanced Reactor Designs 7/7/98	TAP and Responses to ACRS Questions