January 9, 1998

52-3

MEMORANDUM TO: Theodore R. Quay, Director
Standardization Project Directorate
Division of Reactor Program Management

FROM: Timothy E. Collins, Chief /original signed by/
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: SRXB INPUT TO AP600 FSER, CHAPTER 20, GENERIC ISSUES

Enclosed is our update to AP600 FSER, Chapter 20, Generic Issues, addressing those issues for which SRXB has primary responsibility. Most issues are resolved based on recent responses from Westinghouse to staff RAIs. However, some issues are remained open as follow: A-31, Residual Heat Removal Shutdown Requirements, A-9, Anticipated Transient Without Scram (AWTS), and Issue 105, Interfacing System LOCA (ISLOCA).

If you have additional questions, please contact David Diec at 415-2834

Attachment:
As stated

cc: G. Holahan/S. Newberry (w/o attachment)
T. Collins (w/o attachment)
R. Caruso

DISTRIBUTION:

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| PDR | D. Diec | SRXB R/F |
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*see previous concurrence page

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Issue A-31: Residual Heat Removal Shutdown Requirements

As discussed in NUREG-0933, Issue A-31 addressed the ability to transfer heat from the reactor to the environment after shutdown, which is an important safety function. It was resolved in 1978 with the issuance of SRP Section 5.4.7, "Residual Heat Removal (RHR) System."

The safe shutdown of a nuclear power plant following an accident not related to a LOCA has typically been interpreted as achieving "hot standby" condition. The NRC has placed considerable emphasis on the hot-standby condition of a power plant in the event of an accident or other abnormal occurrence and, similarly, on long-term cooling, which is typically achieved by the residual heat removal (RHR) system. The RHR system starts to operate when the reactor coolant pressure and temperature are substantially lower than their hot-standby-condition values. Even though it may generally be considered safe to maintain a reactor in hot-standby condition for a long time, experience shows that certain events have occurred that required eventual cooldown or long-term cooling until the RCS is cold enough for personnel to inspect the problem and repair it.

In SSAR Section 1.9.4.2.2, Westinghouse stated that the AP600 design includes passive safety-related decay heat removal systems that establish and maintain the plant in a safe-shutdown condition following design-b is events and it is not necessary that these passive system achieve cold shutdown as defined in RG 1.139.

The passive core cooling system is design to maintain plant safe-shutdown conditions indefinitely. Cold shutdown condition is necessary only to gain access to the reactor coolant system for inspection, maintenance, or repair. For the AP600 design, cold shutdown conditions can be achieved using highly reliable, but non-safety-related systems, which have similar redundancy as current generation safety-related systems and are supplied with ac power from either onsite or offsite sources. Passive core cooling capability is discussed in Section 6.3 of the SSAR.

Westinghouse states that the passive residual heat removal system can achieve hot standby conditions immediately and can reduce the reactor coolant temperature to 215.6 °C (420 °F) within 36 hours. The reactor pressure is controlled and can be reduced to 1.72 MPa (250 psig). The passive RHR system also provides a closed cooling system to maintain long-term cooling. Therefore, the AP600 complies with GDC 34 by using a more reliable and simplified system for both hot standby and long-term cooling modes, and it is not necessary that these passive systems achieve cold shutdown as defined by RG 1.139.

GDC 34 requires a residual heat removal system to be provided with suitable redundancy in components and features to assure that, with or without onsite or offsite power, it can accomplish its safety functions so that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. No definition is specified as the safe-shutdown condition for which the RHR system should accomplish. EPRI in the Utility Requirements Document for Passive ALWRs proposed that the safe-shutdown condition be defined as 215.6 °C (420 °F) for the passive ALWR

designs. The staff has concluded that cold shutdown is not the only safe stable shutdown condition which can maintain the fuel and reactor pressure boundary within acceptable limits. In SECY-94-084, Section C, "Safe Shutdown Requirements," the staff recommended, and the Commission approved, that the EPRI's proposed 215.6 °C (420 °F) criteria or below, rather than the cold shutdown condition required by RG 1.139, be accepted as a safe stable condition, which the passive RHR system must be capable of achieving and maintaining following non-LOCA events. This acceptance is predicated on an acceptable passive safety system performance and an acceptable resolution of the issue of regulatory treatment of non-safety systems (RTNSS). The SECY paper also states that the passive safety system capabilities can be demonstrated by appropriate evaluations during detailed design analyses, including:

(1) A safety analysis to demonstrate that the passive systems can bring the plant to a safe stable condition and maintain this condition, that no transients will result in the specified acceptable fuel design limits and pressure boundary design limit being violated, that no high-energy piping failure being initiated from this condition will result in violation of 10 CFR 50.46 criteria; and

(2) A probabilistic reliability analysis, including events initiated from the safe-shutdown conditions, to ensure conformance with the safety goal guidelines. The PRA would also determine the reliability/availability missions of risk significant systems and components as a part of the effort for regulatory treatment of non-safety systems.

The resolution of Issue A-31 for the AP600 design remains open as the staff is evaluating both the passive system performance capability through testing and safety analyses, and the proper resolution of the RTNSS issue.

Issue A-9: Anticipated Transient Without Scram

As discussed in NUREG-0933, Issue A-9, addressed the issue of ensuring that the reactor can attain safe shutdown after incurring an anticipated transient with a failure of the reactor trip system (RTS). An anticipated transient without scram (ATWS) is an expected operational occurrence (such as loss of feedwater, loss of condenser vacuum, or loss of offsite power (LOOP) to the reactor) that is accompanied by a failure of the RTS to shut down the reactor.

The acceptance criterion for the resolution of Issue A-9 are as follow:

- Compliance with the mitigation requirement of 10 CFR 50.62(c)(1) that plant equipment must automatically initiate emergency feedwater (EFW) and turbine trip under conditions indicative of an ATWS. This equipment must function reliably and must be diverse and independent from the RTS.

- Compliance with the prevention requirement of 10 CFR 50.62 (c)(2) that the plant must have a scram system that is diverse and independent from the existing RTS.

In SSAR Section 1.9.4.2.2, Westinghouse stated that the AP600 design complies with the requirements in 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," and a discussion of the design features to address the probability of an ATWS is included in Sections 1.9.5 and 7.7 of the SSAR.

Westinghouse indicated that the AP600 design complies with the requirements of 10 CFR 50.62 with a diverse actuation system that includes the AMSAC (ATWS mitigation system actuation circuitry) protection features mandated by 10 CFR 50.62 by tripping the turbine and diversely actuating selected engineered safeguards functions. However, the AP600 design does not automatically initiate auxiliary FW, but instead includes automatic initiation of the PRHR cooling system. The AP600 design does not, therefore comply with 10 CFR 50.62.

There are other AP600 design features aimed at minimizing the probability of ATWS occurrence and mitigating the consequences, as discussed in Section 1.9.5 of the SSAR. For the AP600 design with passive core cooling systems, the staff requires that an ATWS analysis be performed to demonstrate that its ATWS response is consistent with that considered by the staff in its formulation of the 10 CFR 50.62 design requirements for current plant designs. The applicant has provided (response to RAI 440.26) the analysis of a complete loss of normal feedwater without reactor trip, using the LOFTRAN code. The staff has requested additional information concerning the AP600 ATWS analysis, which the applicant has agreed to provide. This is an open issue. The use of the PRHR system in lieu of automatic AFW is also an open item.

Therefore, Issue A-9 is not resolved for the AP600 design.

Issue A-17: Systems Interactions in Nuclear Power Plants

As discussed in NUREG-0933, Issue A-17 addressed concerns regarding adverse systems interactions (ASIs) in nuclear power plants. Depending on how they propagate, ASIs can be classified as functionally coupled, spatially coupled, and induced-human-intervention coupled. As discussed in NUREG-1229, "Regulatory Analysis for Resolution of USI A-17," dated August 1989, and GL 89-18, "Resolution of Unresolved Safety Issue A-17, Systems Interactions in Nuclear Power Plants," dated September 6, 1989, Issue A-17 concerns ASIs caused by water intrusion, internal flooding, seismic events, and pipe ruptures.

A nuclear power plant comprises numerous structures, systems, and components (SSCs) that are designed, analyzed, and constructed using many different engineering disciplines. The degree of functional and physical integration of these SSCs into any single power plant may vary considerably. Concerns have been raised about the adequacy of this functional and physical integration and coordination process. The Issue A-17 program was initiated to integrate the areas of systems interactions and consider viable alternatives for regulatory requirements to ensure that the ASIs have been or will be minimized in operating plants and new plants. Within the framework of the program, the staff requested, as

stated in NUREG-0933, that plant designers consider the operating experience discussed in GL 89-18 and use the probabilistic risk assessment (PRA) required for future plants to identify the vulnerability and reduce ASIs.

This issue identified the need to investigate the potential that unrecognized subtle dependencies, or systems interactions, among structures, systems, and components (SSCs) in a plant could lead to safety significant events. In NUREG-1174, intersystem dependencies are categorized based on the way they propagate into functionally-coupled, spatially-coupled, and induced human-intervention coupled systems interactions. The occurrence of an actual adverse systems interaction (ASI) or the existence of a potential ASI, as well as the potential overall safety impact, is a function of an individual plant's design and operational features. For AP600 with new or differently configured passive and active systems, a systematic search for ASIs is necessary.

Westinghouse submitted WCAP-14477, Revision 0, "The AP600 Adverse System Interaction Evaluation Report," dated February 1996 for staff review and approve. The purpose of the report was to identify possible adverse interactions among safety-related systems and between safety-related and non-safety-related systems, and to evaluate the potential consequences of such interactions. The staff reviewed WCAP-14477 report and provided Westinghouse with comments and questions. Westinghouse subsequently addressed the staff's questions and comments and issued a revision to the WCAP-14477 report. The staff reviews this issue as part of the regulatory treatment of non-safety systems (RTNSS) and has documented its review in Chapter 22 of the FSER.

The staff concludes that Westinghouse has adequately assessed possible adverse systems interactions and their potential consequences in WCAP-1447, revision- 1. In addition, the staff has conducted confirmatory testing involving potential systems interactions, and has performed analyses of selected accident scenarios in which nonsafety and/or safety systems could interact. Both the confirmatory tests and analyses showed that potential systems interactions did not have significant adverse effects on overall safety performance. Additionally, no additional unanticipated adverse systems interactions were observed. OIs 20.2-5 and 20.2-6 are closed. This issue is considered closed.

Issue A-26: Reactor Vessel Pressure Transient Protection

Since 1972, there have been many reported pressure transients which have exceeded the pressure-temperature limits specified in technical specifications (TS) for PWRs. The majority of these events occurred at relatively low reactor vessel temperatures at which the material has less toughness and is more susceptible to failure through brittle fracture. This is Issue A-26 in NUREG-0933 which was resolved with the issuance of SRP Section 5.2.2, "Overpressure Protection." Applicants for CPs and operating licenses were requested to design an overpressure protection system for light-water reactors (LWRs) following the guidance provided in SRP Section 5.2.2.

In its May 28, 1993, letter, Westinghouse stated that the AP600 design conforms to the criteria in Branch Technical Position (BTP) RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," of SRP Section 5.2.2.

The pressurizer is sized to accommodate most pressure transients and over pressure protection for the RCS is provided by either the pressurizer safety valves or the normal residual heat removal relief valves, as described in Section 5.2.2 of the SSAR.

The staff concludes that AP600 design satisfies the BTP RSB 5-2 requirements and therefore, considers this issue closed.

Issue B-22: LWR Fuel

Westinghouse identified in Table 1.9-2, of its May 28, 1993, letter that it considered Issue B-22 relevant to the AP600 design; however, this issue is not required for the AP600 design to meet 52.47(a)(1)(ii) or (iv).

As discussed in NUREG-0933, Issue B-22 addressed the staff concerns that individual reactor fuel rods sometimes failed during normal operations and many fuel rods are expected to fail during severe core accidents. Failure of fuel rods results in radioactive releases within a plant and is a potential source of release to the public. The resolution of this issue was to ensure that these fuel failures did not result in unacceptable releases to the public. Several problems were identified in the staff effort to improve the predictability of fuel performance and these were addressed in the revision to SRP Section 4.2, "Fuel System Design," in 1981. The staff concluded that the then existing requirements on fuel were adequate to ensure continued low fuel defect rates and additional requirements would not significantly decrease the number of fuel defects. This issue was then dropped from further consideration.

Westinghouse stated the AP600 reactor core design complies with SRP Section 4.2 and the discussion on the fuel system design is in Section 4.2 of the SSAR.

The staff has completed its review of the VANTAGE-5H fuel for the AP600 design. The details of fuel design and acceptance criteria are discussed in Section 4.2 of the final safety analysis report. The staff concludes that Westinghouse has satisfactorily resolved all questions raised during the staff review of the issue, and therefore, the staff considers this issue resolved. The Open Item 4.2.8-1 is closed.

Issue C-4: Statistical Methods for ECCS Analysis

As discussed in NUREG-0933, Issue C-4 addressed the statistical methods used for performance evaluation of the ECCS during a LOCA. In accordance with the requirements of 10 CFR 50.46 as amended on September 16, 1988, the NRC requires that the LOCA analyses for license applications use either the 10 CFR Part 50 (Appendix K) evaluation models or the statistical (realistic) models, including the uncertainty of calculation in the adverse direction. The realistic models must be supported by applicable experimental data. Uncertainties in the realistic models and input must be identified and assessed so that uncertainty in the calculated results can be estimated.

In SSAR Section 1.9.4.2.2, Westinghouse stated the AP600 methodology applied for LOCA analysis is discussed in SSAR Chapter 15.

Appendix K of 10 CFR Part 50 specifies the requirements for LWR ECCS analysis, which call for specific conservatism to be applied to certain models and correlations used in the analysis to account for data uncertainties at the time Appendix K was written. USI C-4 addresses NRC development of a statistical assessment of the uncertain level of the peak cladding temperature limit. In 1988, 10 CFR 50.46, "Acceptance Criteria for ECCS for Light Water Nuclear Power Reactors," was revised to allow the realistic ECCS evaluation model, in addition to the evaluation model conforming to the Appendix K requirements. This BE evaluation model will use analytical technique realistically describing the behavior of the reactor system during a LOCA, with comparisons to applicable experimental data. The realistic evaluation model must identify and account for uncertainties in the analysis method and inputs so that when the calculated ECCS cooling performance is compared to the acceptance criteria, there is a high level of probability that the criteria would not be exceeded.

As described in SSAR Chapter 15, computer codes WCOBRA/TRAC and NOTRUMP, respectively, are used for the large- and small-break LOCA analyses. WCOBRA/TRAC is a realistic code, and the uncertainties will be included in the analysis. NOTRUMP is a code using the Appendix K requirements.

Issue C-4 is closed.

Issue C-5: Decay Heat Update

As discussed in NUREG-0933, Issue C-5, addressed the specific decay heat models for the LOCA analysis models. In accordance with the requirements of 10 CFR 50.46 as amended on September 16, 1988, the LOCA analyses for license applications should use either the 10 CFR Part 50 (Appendix K) models, or the realistic models supported by applicable experimental data and including uncertainty of calculation in the adverse direction. When Appendix K models are used, the decay heat generation function should be based on ANS 5.0, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," plus a 20-percent uncertainty factor. When realistic models are used, the decay heat function in ANS 5.1, "Decay Heat Power in Light Water Reactors," is acceptable for licensing applications.

In SSAR Section 1.9.4.2.2, Westinghouse stated that the large-break LOCA analyses for the AP600 design, discussed in Section 15.6.5 of the SSAR, used the decay heat model identified in the 1979 ANSI 5.1 standard.

This issue involved following the work of research groups in determining best-estimate decay heat data and associated uncertainties for use in LOCA calculations.

Appendix K of 10 CFR Part 50 requires the use of 1971 ANS Standard, "Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors," times 1.2 be used for the heat generation rates from the radioactive decay of fission products in the

ECCS calculation. The staff has determined that the 1979 ANSI 5.1 is technically acceptable and has allowed this use in the realistic evaluation model. For the AP600 application, the 1971 ANS decay heat model and the 1979 ANSI decay heat model are used in NOTRUMP and WCOBRA/TRAC, respectively, for small- and large-break LOCAs. The staff has completed and documented its review of WCOBRA/TRAC and NOTRUMP in Chapter 15 of the AP600 FSER. The staff considers Issue C-5 closed.

Issue C-6: LOCA Heat Sources

As discussed in NUREG-0933, Issue C-6 addressed the issue identified in NUREG-0471 which involved staff evaluations of vendors' data and approaches for determining LOCA heat sources and the need for developing staff positions. The contributors to LOCA heat sources, along with their associated uncertainties and the manner in which they are combined, have an impact on LOCA calculations. the staff informed the Commission in SECY-83-472, "Emergency Core Cooling System Analysis Methods," November 17, 1983, that statistical combination of LOCA heat sources would be allowed to justify the relaxation of non-required conservatism in emergency core cooling system (ECCS) evaluation models.

In SSAR Section 1.9.4.2.2, Westinghouse stated that the discussion of LOCA heat sources for the AP600 design is included in Section 15.6.5 of the SSAR.

The staff has completed and documented its review of WCOBRA/TRAC and NOTRUMP in Chapter 15 of the AP600 FSER. The staff considers Issue C-6 closed.

Issue 22: Inadvertent Boron Dilution Events

As discussed in NUREG-0933, Issue 22 addressed the possibility of core criticality during cold shutdown conditions from inadvertent boron dilution events. Although this issue was resolved with no new requirements, the acceptance criterion is that plants shall minimize the consequences of such events by meeting SRP Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)." Specifically, the plant shall respond in such a way that the criteria regarding fuel damage and system pressure are met, and the dilution transient is terminated before the shutdown margin is eliminated. If operator action is required to terminate the transient, redundant alarms must be in place and the following minimum time intervals must be available between when an alarm announces an unplanned dilution and when shutdown margin is lost:

- during refueling (Mode 6) - 30 minutes
- during all other operating modes - 15 minutes

Section 15.4.6 of the SSAR provides a safety analysis which demonstrates that redundant alarms are available to enable operators to detect and terminate an inadvertent boron dilution event within the above required time intervals, before shutdown margin is lost.

In addition to the events in this issue, the staff has identified the following two boron dilution scenarios where a deborated water slug may accumulate in the RCS and a restart of the RCPs will cause this slug to pass through the core resulting in criticality or a power excursion:

- The first scenario occurs during a plant startup when the reactor is deborated as part of startup procedures. A loss of offsite power will result in tripping the RCPs and charging pump. The subsequent startup of the diesel generator will restart the charging pump and cause the accumulation of deborated water in the reactor lower plenum. The RCP restart with recovery of offsite power will cause this deborated water to pass through the core.

- The second scenario is related to transients or accidents, such as a small break LOCA with heat removal by reflux condensation natural circulation that may result in an accumulation of deborated water in the RCS loop. This water will pass through the core with an inadvertent restart of the RCPs.

The staff has completed and documented its review of inadvertent boron dilution issue in Section 15.2.6.5.4 of the staff FSER. The staff considers Issue-22 closed. The Open Item 20.3-2 is closed.

Issue 105: Interfacing System LOCA (ISLOCA) at LWRs

Issue 105, in NUREG-0933, was limited to pressure isolation valves (PIVs) in BWRs and was resolved by requiring leak-testing of the check valves that isolate low-pressure systems that are connected at the RCS outside of containment. It is related to Issue 96 which addressed PIVs between the RCS and RHR systems in PWRs. As stated in NUREG-0933, the staff issued Information Notice (IN) 92-36, Intersystem LOCA Outside Containment," dated May 7, 1992, on this subject. The individual plant examinations required by the staff on operating plants included analyses of these sequences. This issue was resolved without any new requirements for operating plants.

The staff position regarding intersystem LOCA protection, as stated in SECY 90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirement," as well as SECY 93-087, "Policy. Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," is that ALWR designs should reduce the possibility of a LOCA outside containment by designing, to the extent practicable, all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to full RCS pressure. The phrase "to the extent practicable" is a recognition that all systems must eventually interface with atmosphere and that it would be difficult or prohibitively expensive to design certain large tanks and heat exchangers with the URS equal to full RCS pressure. The degree of isolation or a number of barriers is not a sufficient justification for using low-pressure components that can be practically designed to the full RCS URS criteria. Piping runs should be designed to meet the URS criteria, as should all associated flanges, connectors, and packings, including valve stem seals, pump seals, heat exchanger tubes, valve

bonnets, and RCS drain and vent lines. The designer should attempt to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

In Section 3.9.3.1 of this report, the staff discusses its evaluation which establishes the minimum pressure for which low-pressure systems should be designed to ensure reasonable protection against burst failure, should the low-pressure system be subjected to full RCS pressure. The subsection within Section 3.9.3.1, "AP600 Design Criteria for ISLOCA," contains the design criteria proposed by Westinghouse for the low-pressure portion of the normal residual heat removal system (RNS). On the basis of this evaluation, the staff concludes that this criteria is acceptable to ensure that the low-pressure side of any applicable system has been designed to meet the full RCS URS criteria.

For all interfacing systems and components that do not meet the full RCS URS criteria, the applicant must justify why it is not practicable to reduce the pressure challenge any further, and also provide compensating isolation capability. For example, applicants should demonstrate that for each interface the degree and quality of isolation or reduced severity of the potential pressure challenges are compensated and justified for the safety of the low-pressure interfacing systems or components. The adequacy of pressure relief and the piping of relief back to primary containment are possible considerations. As identified in SECY-90-016, each of these interfacing systems that has not been designed to withstand full RCS pressure must also include the following protection measures: (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the control room when isolation valve operators are de-energized, and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of the attached low-pressure system and both isolation valves are not closed.

Section 1.9.5.1.7 of the SSAR discusses Westinghouse responses regarding compliance of the AP600 design with the staff position on ISLOCA. In WCAP-14425, "Evaluation of the AP600 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," dated July 1995, Westinghouse performed a systematic evaluation of the design responses of various systems interfacing the RCS of the ISLOCA challenges. The systematic evaluation process includes (1) a review of the AP600 piping and instrumentation diagrams to identify these primary interfacing systems or subsystems directly interfacing with the RCS, and the secondary interfacing systems or subsystems interfacing with the primary interfacing systems, (2) identification of primary and secondary systems and subsystems having its URS less than the RCS pressure. For those systems or subsystems not meeting the criterion of the URS greater than the RCS pressure, a design evaluation is made on whether it is inside containment, whether it meets the three criteria specified in SECY 90-016, and or whether it includes other design features specific to them that prevent an ISLOCA to the extent practicable. The report also provides its reasons why it is not practical to design large, low-design pressure tanks and tank structures that are vented to the atmosphere to the high pressure criterion. Interfacing systems or subsystems that connect directly to an atmospheric tank are excluded from further ISLOCA consideration. This is limited to the piping connected directly to the atmospheric tank, up to the first isolation valve other than a locked-upon, manual isolation valve. The staff evaluation of these systems follows.

(a) Normal Residual Heat Removal System:

The portions of the RNS from the RCS to the containment isolation valves (CIVs) outside containment are designed to the RCS operating pressure, and the portions downstream of the CIVs and upstream of the discharge line CIVs are designed so that its URS is not less than the RCS operating pressure. The mechanical shaft seal of the RNS pump with a design pressure of 900 psig is the only portion of the RNS having the URS lower than the RCS pressure. Subsection 3.1.3.2 of WCAP-14425 discusses the difficulties of designing the RNS pump seal to full RCS operating pressure. A fundamental problem is that any type of seal that can withstand the full RCS pressure will likely have abnormally fast wear of the seal faces during normal plant operation at low seal pressure. This increased wear at normal plant operating conditions could well prevent the seal from maintaining the pressure boundary if ever exposed to the full RCS pressure. The use of high pressure seals will also require more frequent maintenance during normal operation. Therefore, it is impractical to design a seal that would maintain the RCS pressure boundary with no leakage, and also operate satisfactorily at low-pressure conditions. The AP600 RNS pump mechanical seal is designed to minimize the amount of leakage if exposed to full RCS pressure. An INEL study on the Davis Besse Nuclear Power Station decay heat removal pump seal, with a design pressure of 450 psig, found that the rotating seal would maintain its structural integrity to pressure in excess of 2500 psi, and the mechanical seals could withstand a pressure of 1200 to 1250 psi without leaking. The AP600 RNS pump mechanical seal is similar to the Davis Besse DHR pumps, however, its design pressure is twice as high. The AP600 RNS pump also has a disaster bushing that limits the leakage from the pump to within the capabilities of the normal makeup system in case of catastrophic mechanical seal failure. The leakage can be controlled with the seal leakoff line routed to a floor drain and subsequently to the auxiliary building sump. This is more favorable than a seal specially designed for full RCS pressure at the expense of normal-condition reliability.

Subsection 5.4.7.2.2 of SSAR discusses the AP600 design features in the RNS specifically aimed at reducing the likelihood of an intersystem LOCA. On the suction side, there is a normally closed motor-operated isolation valve in the common suction line outside containment, and two normally closed motor-operated isolation valves in each parallel suction line inside the containment. There is also a relief valve with a set pressure of 563 psig connected to the RNS pump suction line inside containment. This valve is designed to provide low-temperature overpressure protection of the RCS and will reduce the risk of overpressurizing the RNS. On the discharge side, the common discharge line has a safety-related containment isolation check valve inside containment and a safety-related motor-operated isolation valve outside containment. The MOVs inside the containment are interlocked to prevent them from opening when the RCS pressure is above the RNS operating pressure of 450 psig. The power to these isolation valves is administratively blocked at the valve motor control centers to prevent inadvertent opening. In addition, the discharge header contains a relief valve, which discharges to the liquid radwaste system (WLS) effluent holdup tanks, to prevent overpressure in the RNS pump discharge line that could occur if the three check valves and the motor-operated CIV leaked back to the low pressure portions of the RNS.

Also, the RNS design includes an instrumentation channel that indicates pressure in each RNS pump suction line, and a high pressure alarm is provided in the main control room to alert the operator to a condition of rising RCS pressure that could eventually exceed the RNS design pressure. The motor-operated pressure isolation valves also have remote position indications in the main control room. However, the RNS design does not include the capability to conduct leak testing of the PIVs as required by SECY 90-016 for those systems not designing with the URS capable of withstanding the RCS pressure. AP600 technical specification LCO 3.4.16 requires the integrity of each RCS PIV be maintained with surveillance requirements of the PIV to verify its operability specified in SR 3.4.16.1 and 3.4.16.2. However, the TS Basis excludes the RNS PIVs from the TS 3.4.16, which is not consistent with SECY-90-016. The staff will require the RNS PIVs be included in TS 3.4.16 for periodic leak testing. This is an Open Item.

(b) Chemical and Volume Control System - Makeup systems:

Subsection 9.3.6 of the SSAR provides a detailed description of the design, functions, and operations of the chemical and volume control system (CVS). The purification flow path of the CVS is a high-pressure closed-loop design which is entirely within the containment. The potential contributors to an ISLOCA are the portions of the CVS located outside the containment, i.e., the letdown line to the liquid radwaste system, and the makeup system.

The CVS makeup pumps operate intermittently to make up for RCS leakage. The pumps start and stop automatically when the pressurizer level reaches the bottom and the top of the normal level band, respectively. The makeup pumps take suction from either the boric acid tank, or the demineralized water storage tank (DWST), and inject into the CVS purification loop return stream. The makeup pumps can also take suction from the waste holdup tanks or the spent fuel pool. The makeup line from the makeup pump discharge to the RCS has a design pressure greater than or equal to the RCS design pressure. However, the pump suction line piping and associated components have a design pressure of 150 psig with the URS less than the RCS operating pressure.

Subsection 3.3.3 of the WCAP-14425 contends that it is not practicable to design the low-pressure portions of the makeup suction piping to higher design pressure. It is not practicable to have a high design pressure for large tanks such as the boric acid tank, which are vented to the atmosphere, as well as the piping directly connected to these atmospheric tanks up to the first isolation valve. The suction lines each contain a check valve that separates the suction piping from a large atmospheric tank. These check valves are designed to open on low differential pressure, and have a high tendency to leak. If the two discharge line check valves are assumed to leak, then it is also reasonable to assume that the suction line check valves will leak. Therefore, designing the suction piping to a higher pressure will only increase the likelihood that the RCS leakeage will flow to one of the atmospheric tanks. The suction lines contain relief valves that protect the low-pressure portions of the piping from overpressure in the event of leaking check valves in the discharge line or thermal expansion in case of a loss of miniflow cooling. The relief valves direct any leakage from the discharge line check valves to the WLS effluent holdup tanks (EHTs), which is designed to handle radioactive fluids, and its level is monitored by remote instrumentation.

The passage of the high pressure reactor coolant to the CVS makeup suction is possible only when the makeup pumps are not running, or as a result of failures or leakage of multiple check valves on the makeup pump discharge side. There is a high-pressure alarm in the pump suction line to alert the operator of overpressurization. In the event of a suction-side overpressurization, the makeup pumps can be operated to terminate overpressurizing the suction piping. If the makeup pumps do not start, the makeup line containment isolation valves would automatically close to terminate the ISLOCA. In addition, the purification loop inlet isolation valves will also be closed on a safeguards actuation signal. These multiple, safety-related isolation valves mitigate an ISLOCA in the makeup suction line. The makeup line CIVs have the capability for leak-testing, and are provided with valve position indication in the control room at all times. The staff finds that protection measures meet the intent of SECY-90-016 ISLOCA position with the following exception of the CIV leak testing. The makeup line CIVs are also the pressure isolation valves. They should be subjected to the PIV leak test requirements, and be included in TS 3.4.16 for PIV LCO and surveillance requirements.

WCAP-14425 does not provide the relieving capacity and setpressure of the makeup system relief valves, as well as the analysis to demonstrate the acceptability of these values. The staff will require that the relieving capacity and setpressure of the relief valves in the makeup system along with their bases be documented. This is an Open Item.

(c) CVS Letdown - Liquid Radwaste System:

The CVS letdown line connects to the high-pressure purification loop inside containment. Immediately downstream of this connection is a high-pressure, multi-stage letdown orifice, which reduce pressure in the letdown line from the RCS operating pressure to below the design pressure of the low-pressure portion of the letdown line. Around the letdown orifice is a bypass line containing a locked-closed manual isolation valve that is opened only at shutdown when the RCS is depressurized to provide sufficient letdown flow when required. The letdown line is then equipped with two safety-related, normally-closed, fail-closed CIVs while penetrating containment to the liquid radwaste (WLS) degasifier package and EHTs. The letdown line down to and including the outboard CIV has a design pressure of 2485 psig. Downstream of the outboard CIV, the WLS letdown line has a design pressure of 150 psig, and therefore does not meet the RCS URS criteria.

Subsection 3.2.3 of WCAP-14425 contends that it is not practicable to design the low pressure portions of the letdown line to a higher design pressure. The WLS EHTs are large atmospheric tanks, and are therefore not practicable for higher design pressure. Nor are the letdown line, which is routed to the degasifier package or the EHTs, and the degasifier package, which discharges directly to the WLS EHTs. The CVS letdown system has the following features to meet the ISLOCA criteria: (1) the pressure drop across the CVS letdown orifice protects the WLS from overpressurization during letdown operations by reducing the pressure in the WLS, (2) in case of an inadvertent valve closure in the WLS during letdown, a relief valve, which discharges directly to the EHT, is provided that would protect the WLS from overpressurization, (3) due to the letdown orifice, a break in the WLS during letdown from the CVS would result in an RCS leak that is within the capability of the normal makeup system, (4) if an ISLOCA should occur, it would be terminated by automatic

isolation of the two purification loop isolation valves and two letdown isolation valves on low pressurizer level or a safeguards actuation signal, and (5) the letdown line CIVs have the capability for leak testing and have valve position indication in the control room at all time, and (6) the WLS degasifier column contains a high-pressure alarm that would warn the control room operators that the WLS pressure was approaching the design pressure. The staff finds these arguments acceptable except for the following: The CIVs are also the pressure isolation valves, and should be subjected to the PIV leak testing requirements, instead of leak testing for CIVs. These valves should be included in the AP600 technical specifications LCO 3.4.16 and associated surveillance requirements. This is an Open Item.

(d) Primary Sampling System:

The primary sampling system (PSS) collects representative samples of fluids from the RCS and associated auxiliary system process streams, and the containment atmosphere for analysis by the plant operating staff. Section 3.4 of WCAP-14425 provides an ISLOCA evaluation of the PSS. The PSS pipings are 3/8-inch small pipes. The whole PSS is designed to full RCS pressure and temperature, with the exception of the following low pressure portions: eductor water storage tank (EWST) and its drainage and level indication lines, eductor supply pump seal, and demineralized water supply line. These portions have design pressures with an URS below the RCS operating pressure. The applicant contends that it is not practical to design the low pressure portion of the PSS to a higher design pressure because they are at atmospheric pressure and connect to the low pressure demineralized water system (DWS). Designing the EWST to high pressure to meet ISLOCA criteria would require the DWS to be designed for high pressure, which is not practicable.

The PSS is connected to the RCS through the local sample points in the RCS hot legs, pressurizer vapor and liquid spaces, and the core makeup tanks. Each of these sampling connection lines contains a flow-restricting orifice that limits the flow from the RCS in the event of a sample line break, and also reduces the pressure in the sampling lines during sampling operations. Each sampling line also contains a normally closed isolation valve before connecting to a common header. The common header then penetrates outside the containment with two normally closed CIVs, which are also the PIVs and will be isolated on a safeguards signal if they are open for sampling operation. The sampling line then connects to a sample cooler and the sample bottles. In addition, one of the two lines connected to the low-pressure portion of PSS contains two check valves, and the other contains one check valve and one normally closed isolation valve. In the event that these valves leak, the leakage would not overpressurize the low pressure portions of the system, but would flow directly to the EWST. In the unlikely event of a gross failure of the high pressure check valves, the maximum flow rate from the RCS would be within the capability of the normal makeup system. The water level in the EWST is monitored, and a high alarm in this tank would alert the operator to a potential leak into the tank from the PSS sampling lines. The operator would then be able to isolate the leak by closing the CIVs. The CIVs have remote position indication in the control room and are subject to the CIV leakage test. Therefore, the PSS design meets the intent of the ISLOCA criteria with the exception of PIV leak test requirements the CIVs, which are also PIVs. These PIVs should be included in TS 3.4.16 for PIV LCO and surveillance requirements This is an Open Item.

(e) Solid Radwaste System:

The solid radwaste system (WSS), which provides storage facilities for both wet and dry solid wastes prior to and subsequent to processing and packaging, is connected to the high-pressure CVS demineralizers to facilitate transfer of the spent resin from the CVS demineralizers to the spent resin storage tanks (SRSTs). The spent resin header connects to each of the three CVS demineralizers with an individual, normally closed isolation valve, and then penetrates containment with two normally closed locked-closed CIVs to the SRSTs outside. A manual valve is placed downstream of the outboard CIV to isolate the downstream piping to facilitate CIV leak testing. The portion of piping downstream of the manual isolation valve is a low-pressure design with the URS below the RCS operating pressure. Section 3.5.2 of WCAP-14425 contends that it is not practical and necessary to design the WSS to a higher design pressure because the system contains many low-pressure components such as the SRST and resin transfer and mixing pumps.

The WSS spent resin line is normally isolated by locked closed manual valves, which are administratively controlled, have position indications in the control room, and are leak-tested in accordance with the inservice testing plan of SSAR subsection 3.9.6. The CVS demineralizers are inside containment and normally circulate reactor coolant at RCS pressure. Resin transfer operations are conducted only during refueling operations when the RCS is fully depressurized. During normal power operation, the only pathway to the low-pressure portion of the WSS is for all three closed isolation valves to fail. Should that extremely unlikely event happen, the recirculation loop isolation valves can be closed to isolate the purification loop and the WSS from the RCS. In addition, downstream of the inboard CIV in the resin transfer line, there is a relief valve which discharges to the WLS containment sump inside containment. Therefore, the WSS spent resin lines are not required to be designed to a higher design pressure.

(f) Demineralized Water Transfer and Storage System:

The demineralized water transfer and storage system (DWS) receives water from the demineralized water treatment system, and provides a reservoir of demineralized water to supply the condensate storage tank and for distribution throughout the plant. The design and functional details of the DWS is provided in Subsection 9.2.4 of the SSAR. The demineralized water transfer pumps take suction from the demineralized water storage tank (DWST) and supply water through a catalytic oxygen reduction unit to the demineralized water distribution header. From this header, demineralized water is supplied to various systems in the plant. The DWS supply line penetrates containment to a supply header inside containment, which provides interface of the DWS interface with the PSS and the CVS demineralizers. The DWS provides demineralized water to the PSS to flush the PSS lines prior to RCS sampling, and to the CVS demineralizers to sluice resin to the WSS.

The DWS is a low-pressure system design with the URS below the RCS operating pressure. However, the only possible overpressurization pathways from the RCS are the connections to the PSS and the CVS demineralizers inside containment. Overpressurization of the DWS can only occur if there are multiple failures and misalignments of isolation valves and check valves in the high-pressure systems. A relief valve has been added to the DWS

header inside containment to preclude the possibility of overpressurizing the DWS. In addition, an overpressurization of the DWS would most likely result in the rupture of the DWS header inside containment, and therefore is not a concern of ISLOCA.

The staff concludes that, with the exception discussed below, the AP600 design is consistent with the staff position discussed in SECY-90-016 regarding ISLOCA. Therefore, Issue 105 is resolved subject to the resolution of the following open items:

OI 20.3-105.1, All PIVs in the RNS, CVS, and PSS should be included in AP600 TS 3.4.16 and subject to periodic PIV leak testing. This open item is documented in an NRC letter to Westinghouse (W. Huffman to N. Liparulo), "Open Item Associated with AP600 Pressure Isolation Valve Leak Testing," October 16, 1997.

OI 20.3-105.2, the relieving capacity and setpressure of the CVS makeup system relief valves, as well as their bases should be documented.

Issue 122.2: Initiating Feed and Bleed

As discussed in NUREG-0933, Issue 122.2 investigated the findings of the NRC inspection in 1985 of the loss-of-feedwater event at Davis Besse on June 9, 1985. The issue dealt with the adequacy of emergency procedures, operator training, and available plant monitoring systems for determining the need to initiate feed-and-bleed cooling following the loss of the SDG heat sink (i.e., loss of feedwater). In an analysis of the loss-of-feedwater event, the staff found that operators were hesitant to initiate feed-and-bleed operations, and that the control room instrumentation was inadequate to alert operators to the need to initiate feed and bleed. A loss-of-feedwater in combination with a failure to diagnose and take corrective actions (i.e., initiate feed and bleed) would result in loss of core cooling.

The staff has completed its review of Westinghouse provided information relating the feed and bleed emergency guidelines AFR -H.1, "AP600 Response to Loss of Heat Sink," and has concluded that the feed and bleed emergency guidelines are acceptable. Therefore, this issue is resolved for the AP600 design. Open Item 20.3-18 is closed.

Issue I.D.5(3): Control Room Design - On-Line Reactor Surveillance Systems

As discussed in NUREG-0933, Issue I.D.5(3) addressed the benefit to plant safety and operations of continuous on-line automated surveillance systems. Systems that automatically monitor reactor performance can benefit plant operations and safety by providing continuous diagnostic information to the control room operators, to predict anomalous plant behavior.

Various methods of on-line reactor surveillance have been used, including neutron noise-monitoring in BWRs to detect vibrations in internal components, and pressure noise surveillance at TMI-2 to monitor primary loop degasification. On-line surveillance data have been used to assess loose thermal shields.

Continuous on-line surveillance of the NSSS involves the following areas for which acceptance criteria are separately defined:

- vibration monitoring of reactor internals
- RCPB leakage detection
- loose-parts monitoring

The acceptance criteria for the resolution of Issue I.D.5(3) for monitoring vibrations in internal components are in ANSI/ASME OM-5-1981, "Inservice Monitoring of Core Support Barrel Axial Preload in Pressurized Water Reactors." This standard makes recommendations on the use of ex-core neutron detector signals for monitoring core barrel axial preload loss. This standard also documents a program containing baseline, surveillance, and diagnostic phases and makes recommendations for data acquisition frequency and analysis.

The acceptance criteria for leak monitoring are in RG 1.45 that documents acceptable methods for channel separation, leakage detection, detection sensitivity and response time, signal calibration, and seismic qualification of RCPB leakage detection systems. It defines the regulatory position for an acceptable design of these systems.

The acceptance criteria for loose-parts monitoring are in RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors." This RG gives guidelines on such system characteristics as sensitivity, channel separation, data acquisition, and seismic and environmental conditions for operability. It also identifies alert levels, data acquisition modes, safety analysis reports, and TS pertaining to a LPMS.

AP600 design includes the reactor coolant pressure boundary leakage detection system as required by 10 CFR 50, Appendix A, General Design Criterion 30, and the conformance to the staff regulatory positions, as indicated in RG 1.133, for the design of the loose-parts monitoring system. The detailed system design discussions are in SSAR Chapters 5 & 7.

The staff has reviewed the functional requirements of the metal impact monitoring system (MIMS), which monitors the reactor coolant system for the presence of loose metallic parts according to the regulatory position requirements as indicated in RG 1.133, rev 1, May 1981, and concludes that the MIMS functional design requirements satisfied RG 1.133, with the exceptions of system surveillance and reporting requirements, which are most appropriately addressed by the Combined Operating Licensees for plant specific design.

Issue I.D.5(3) is resolved.

Issue II.D.3: Coolant System Valves - Valve Position Indication

As discussed in NUREG-0933, Issue II.D.3 addresses the requirements in NUREG-0737 for positive indication in the control room of RCS relief or safety valve position. The acceptance criterion for the resolution of this issue is that the plant design shall include safety and relief valve indication derived from a reliable valve-position detection device or a

reliable indication of flow in the discharge pipe in accordance with the requirements in NUREG-0737. This indication shall have the following design features.

- Unambiguous safety and relief valve indication shall be provided to the control room operator.
- Valve position should be indicated within the control room and should be alarmed.
- Valve position indication may be either safety or control grade; if it is control grade, it must be powered from a reliable (e.g., battery-backed) instrument bus (see RG 1.97).
- Valve position indication should be seismically qualified consistent with the component or system to which it is attached.
- Valve position indication shall be qualified for the appropriate operating environment which includes the expected normal containment environment and an OBE.
- Valve position indication shall be human-factors engineered.

As discussed in the staff AP600 DSER, confirmatory item 20.4-1 requires that Westinghouse update SSAR Table 3.11-1 to include remote positive indication for the pressurizer safety valve, normal RHR relief valve, and steam generator safety valves. The staff has reviewed the latest SSAR revision 9, Table 3.11-1, which indicates that positive indications have been included for these valves. Item 20.4-1 is considered closed. TMI action item II.D.3 is resolved.

Issue II.E.2.2: Research on Small Break LOCAs and Anomalous Transients

As discussed in NUREG-0933, Issue II.E.2.2, addressed the NRC research programs focused on small break LOCAs (SBLOCAs) and reactor transients. The programs included experimental research in the loss of flow tests (LOFT), semiscale LOFT, Babcock and Wilcox integral systems test facilities, systems engineering, and material effects programs, as well as analytical methods development and assessments in the code-development program.

The programs called for in this issue were completed by the NRC and showed that ECCSs will provide adequate core cooling for SBLOCAs and anomalous transients consistent with the single-failure criteria of Appendix K to 10 CFR Part 50. The application of the experimental data from the research programs to validate the conservatism of the licensing codes used in the SBLOCAs are addressed in Issue II.K.3(30) in this section.

Westinghouse did not address this issue in its May 28, 1993, letter. It concluded, in Table 1.9-2 of that letter, that this issue was not relevant to the AP600 design because this issue was resolved with no new requirements.

Because the AP600 design is the first passive advanced LWR design to be reviewed by NRC, the staff is considering how the research for the non-passive LWRs apply to this

design. The distinguishing feature of the AP600 is a dependence on safety systems whose operation is driven by natural forces , such as gravity and stored mechanical energy.

While passive systems may be conceptually simpler than conventional active systems, they may be potentially more susceptible to system interactions that can upset the balance of forces upon which the passive systems depend on for their operation. It should be noted that these "passive" systems still rely one some active operation to place them in operation.

For a design with passive safety systems and without a prototype plant that will be tested over an appropriate range of normal, transient, and accident conditions, the following requirements, the following is required by 10 CFR 52.47(b)(2)(I)(a):

- The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof.

- Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof.

- Sufficient data exist on the safety features to the design to assess the analytical tools used for safety and analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

Westinghouse has developed test programs for the AP600 design to investigate the passive reactor and containment safety systems, including component phenomenological (separate-effects) test, and integral-systems tests. The staff has completed and documented its review of the AP600 testing programs in Chapter 21 of the AP600 FSER. Issue II.E.2.2 is considered closed.

Issue II.E.5.1: Design Evaluation

As discussed in NUREG-0933, Issue II.E.5.1, addressed the requirement for B&W licensees to propose recommendations on hardware and procedural changes relative to the need for methods for damping primary system sensitivity to perturbations in the once-through SG. In 10 CFR 50.34(f)(20(xvi), it is stated that a design criterion should be established for the allowable number of actuation cycles of the ECCS and RPS consistent with the expected occurrence rate of severe overcooling events considering anticipated transients and accidents.

Westinghouse identified in Section 1.9.3 of the SSAR that it considered issue II.E.5.1 relevant to the AP600 design and stated that although this issue applies only to B&W designs, the AP600 design uses the passive core cooling system to provide emergency reactor coolant inventory control and emergency decay heat removal. Component design criteria has been established for the number of actuation cycles for the passive core

cooling system. The identified actuation cycles include inadvertent actuation, as well as the system response to expected plant trip occurrences, including overcooling events. Operation of the ADS is not expected for either design basis or best estimate overcooling events. Section 3.9.1 of the SSAR has additional information.

The staff reviewed Table 1.9-2, which provided status of TMI and USI/GSI related items discussions, including Item II.E.5.1 in Section 1.9 of the SSAR. The staff considers Open Item 20.4-15 closed.

Issue II.F.2: Identification of and Recovery From Conditions Leading to Inadequate Core Cooling

As discussed in DSER Open Item 20.4-17, 10CFR 50.34(f), Additional TMI-related Requirements, requires that instruments be provided in the control room, which have unambiguous indication of inadequate core cooling (ICC), such as primary coolant saturation meters in PWRs, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs. NUREG-0737, TMI action plan item II.F.2, discusses the ICC phenomena and the need to have a reactor water level indication system that provides indication of reactor coolant void fraction when the reactor coolant pumps (RCPs) are operating, and reactor vessel water level when the RCPs are tripped.

Prior to the TMI accident, an accepted operational practice of PWRs was to operate the RCPs, if they were available, during a LOCA to provide continued core cooling. During the TMI LOCA event with the stuck open PORVs, the reactor coolant continued to leak through the open valves, the pressurizer level indicated high, and subsequent ICC occurred because the reactor coolant was highly voided. Nevertheless, core cooling was maintained with the continued operation of the RCPs. Subsequently, the RCPs were tripped and because of high void content in the coolant, the water level dropped below the top of the core causing fuel damage. As a result of the TMI lessons learned, the reactor vessel water level indication system was added, specifically for PWRs, to ensure operator action to trip the RCPs following a LOCA, rather than later in the LOCA sequence to prevent ICC event. NUREG/CR-5374, Summary of Inadequate Core Cooling Instrumentation for U.S. Nuclear Power Plants discusses acceptable approaches to instrumentations used to address ICC.

In response to staff RAI #440.162, Westinghouse explained that the AP600 design concept is different from current operating plants in that the AP600 design automatically trips the RCPs and initiates safeguard injections through the passive safety systems such as CMT, ADS, PRHR and IRWST to maintain core cooling in the event of a SBLOCA. It does not rely on a reactor vessel level indication system as do existing reactors, where reactor vessel level indication is important for operator actions to trip the RCPs, to monitor coolant mass in the vessel and to manually depressurize the RCS in the event of ICC. There is no need in the AP600 for the operator to trip the RCPs, to inject water into the core or to manually depressurize the plant during a SBLOCA.

The instruments typically used in current PWRs include subcooling margin monitoring capability, core-exit thermocouples, and reactor vessel level indication system, which

together would provide the operator with the ability to monitor the coolant conditions and to appropriately take actions to ensure core cooling during the approach to and to recover from the inadequate core cooling conditions. The AP600 design includes subcooling margin monitoring capability, core-exit thermocouples and the hot-leg level indication system. The AP600 hot-leg level indication system is different from the reactor vessel level indication systems currently used in Westinghouse plants.

The AP600 hot-leg level indication is a safety-related level indication system, which consists of separate pressure taps that connect to the bottom of the hot leg, and to the top of the hot leg bend leading to the steam generator and has the ability to provide indication of reactor water vessel level for a range spanning from the bottom of the hot leg to approximately the elevation of the vessel mating surface.

In addition, during the operation of the ADS to depressurize the plant, the reactor vessel water level will vary greatly and will not provide a reliable indication of ICC. The AP600 hot-leg water level indication is not used to direct operator actions even when the water level may potentially drop below the hot leg level. Therefore, the water level is not an important indication for mitigation of ICC in the AP600 design. The hot-leg level indication system is used, however, as a verification of reactor water inventory to terminate the recovery action in the ERGs for the ICC event.

Because the AP600 design automatically trips the RCPs during a SBLOCA event and because the operators are not prone to be mislead by forced two phase flow, the core exit temperature is an important and sufficient indication of an approach to ICC condition. The temperature reading provided by core-exit thermocouples has been appropriately included in the ERGs for plant recovery.

The staff has reviewed the Westinghouse response and has determined that for a SBLOCA event a safeguard signal would automatically trip the RCPs, passive safety systems such as the CMT would automatically inject water into the core, the ADS would automatically initiate to depressurize the plant, the reactor coolant would automatically be cooled by the PRHR, and subsequent injection from the IRWST would occur. The staff has also dertemined that for AP600 design, the core-exit thermocouples and the subcooling margin monitoring together would provide unambiguous indication of an approach to ICC and the safety-related hot leg level indication is only used to terminate the recovery action in the ERGs for the ICC event. Therefore, the requirements for ICC, as discuss in 10CFR 50.34(f), have been satisfied and the issue is resolved.

Issue II.K.1(3): Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents

As discussed in NUREG-0933, Issue II.K.1(3) requested licensees to have operating procedures for recognizing, preventing, and mitigating void formation in the RCS during transients and accidents to avoid loss of the core-cooling capability during natural circulation.

The staff has reviewed the resolution of Issue I.C.I and its related ERGs AES-0.2, "Natural Circulation Cooldown," and has concluded the guidelines direct the operators to cooldown and depressurize the plant using natural circulation conditions by dumping steam and subsequent RNS operation. These steps are specified to preclude any possible upper head voids formation and also direct the operators to verify that a steam void does not exit in the vessel. The staff concludes that the ERGs provide directions to plant operators to recognize and to preclude voids formation in the vessel and therefore, the staff considers Issue II.K.1(3) closed.

Issue II.K.1(4d): Review Operating Procedures and training to Ensure that Operators are Instructed Not to Rely on Level Alone in Evaluating Plant Conditions

As discussed in NUREG-0933, Issue II.K.1(4D) asked licensees to provide operating procedures to ensure that operators shall not rely on level indication alone in evaluating plant conditions. As stated in NUREG-0933, the staff determined that this issue was covered by Issues I.A.3.1, I.C.1, and II.F.2, and is resolved.

Issue I.A.3.1, "Revise Scope and Criteria for Licensing Examinations," was implemented by NRC by a rule change to 10 CFR Part 55, "Operators Licenses," to require simulator as part of the reactor operator licensing examinations. The staff will impose the requirements of 10 CFR 55.45 on simulators on the COL applicant referencing the AP600 design; therefore, Westinghouse and the staff does not have to address Issue I.A.3.1 for compliance with 10 CFR 52.47(a)(1)(iv).

Westinghouse did not address this issue in its May 28, 1993, letter. It concluded, in Table 1.9-2 of that letter, that this issue was not relevant to the AP600 design because this issue is not a design certification issue, but is the responsibility of the COL applicant. However, in response to the staff request for additional information (RAI), Westinghouse stated that the design portion of this item is addressed in the proposed resolution to Issues I.C.1 and II.F.2.

The staff has completed its review of Issues I.C.I and II.F.2 and has concluded that AP600 ERGs do not instruct the operators to rely on level indication alone in evaluating plant conditions. The status of core cooling is determined by indications of core exit thermocouple temperature, RCS subcooling, and RCS hot leg temperature in addition with RCS level. The staff considers these issues resolved and therefore, Issue II.K.1(4d) is closed.

Issue II.K.1(17): Trip Pressurizer Level Bistable so that Pressurizer Low Pressure Will Initiate Safety Injection

As discussed in the staff DSER Open Item 20.4-22, TMI action plan item II.K.1(17) addresses the requirement for Westinghouse plants to trip the pressurizer level bistable so that the pressurizer low pressure, rather than the pressurizer low pressure and pressurizer low level coincidence, would initiate safety injection.

AP600 design does not depend on pressurizer low pressure and pressurizer low level coincidence to initiate safety injection in the event of LOCAs. Safety injection in AP600

design is automatic. The following safeguard signals would initiate safety injection: Low-1 pressurizer pressure or Hi-1 containment pressure or Low compensated steam line pressure or Low-3 cold leg temperature. In addition, the AP600 design also gives the operator manual safety injection capability. The staff concludes that any single safeguard signals mentioned above would initiate safety injection. Therefore, this issued is resolved.

Issue II.K.1(24): Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip

Issue II.K.1(24), of NUREG-0933 required PWR licensees to perform a LOCA analysis for a range of small-break sizes and a range of time lapses between reactor trip and RCP trip. The staff determined in NUREG-0933 that this issue for PWRs was covered by Issue I.C.1, "Short-Term Analysis and Procedures Revision."

Westinghouse has provided for staff review of AP600 Emergency Response Guidelines (ERGs), which addresses Issue I.C.1. The staff has reviewed the responses to Issue I.C.1 and specifically the emergency response guideline AE-0, "AP600 Reactor Trip or Safety Injection" for small break LOCA that addresses item II.K.1(24) and has concluded that the AP600 design automatically trips the RCPs during a LOCA event. The guideline directs the operators to verify that all reactor coolant pumps have been tripped, and if not, the operators are directed to manually trip the reactor coolant pumps. Based on the plant design features and the appropriate operator's actions using ERGs, the staff considers item II.K.1(24) resolved. Open Item 20.4-23 is closed.

Issue II.K.1(25): Develop Operator Action Guidelines

As discussed in NUREG-0933, Issue II.K.1(25) required PWR licensees to develop operator action guidelines based on the analyses performed in response to Issue II.K.1(24), which is discussed above. The staff determined in NUREG-0933 that this issue was covered by Issue I.C.1.

Westinghouse did not address this issue in its May 28, 1993, letter. It concluded, in Table 1.9-2 of that letter, that this issue was not relevant to the AP600 design because the issue had been superseded by one or more other issues. Although this issue was covered by Issue I.C.1, as stated above, Westinghouse also did not address this latter issue because it considered Issue I.C.1 the sole responsibility of the COL applicant.

The final procedures would be the responsibility of the COL applicant; however, the range of LOCA analyses for a range of time lapses and the specific information to go into the procedures would be the responsibility of the designer, or Westinghouse in the case of the AP600 design. Westinghouse addresses accidents for the AP600 design in SSAR Chapter 15. The staff requests that Westinghouse address operator action guidelines, or EPGs, of I.C.1 and the role of the COL applicant in Issue II.K.1(25). This is Open Item 20.4-24.

The staff has completed its review of Issue I.C.1 and has concluded that Issue I.C.I is closed, therefore, Issue II.K.1(25) or Open Item 20.4-24 is also closed.

Issue II.K.1(27): Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling

As discussed in the staff DSER, the AP600 design should describe analyses of ICC conditions and develop guidelines and procedures to mitigate an ICC event, and that this issue is dependent on the resolution of Action Items II.F.2 and I.C.1, Identification of and Recovery from Conditions Leading to ICC and Guidance for Evaluation and Development of Procedures for Transients and Accidents, respectively.

Westinghouse has submitted AP600 Emergency Response Guideline (ERG) for staff review, and also responded to staff DSER Open Item 20.4-17 to address Action Items I.C.1 and II.F.2, respectively. The staff has reviewed Westinghouse response to Action Item II.F.2 and a detailed discussion of this item is documented in its respective section. In the AP600 ERG, Westinghouse provides high-level guidance to deal with inadequate core cooling conditions. The staff has reviewed AFR-C.1, AP600 Response to Inadequate Core Cooling procedure and analysis bases, which describes how passive safety-related systems would automatically trip the RCS pumps, initiate and depressurize the RCS to inject water into the core upon receiving a safeguard signal. In this procedure, the operators are instructed to monitor plant conditions using core exit temperature and indicated hot leg level, which is designed to provide indication of an approach to ICC and to recover from an ICC condition. The operators are also instructed to manually initiate injection when automatic passive safety injections fail. Passive safety-related system actuation indications of CMT, ADS, PRHR, and IRWST are integrated into the procedures, which provide operators with directions to ensure that adequate core cooling will be maintained. Therefore, the staff concludes that Westinghouse has appropriately provided analyses and procedures to mitigate ICC conditions. Issue II.K. 1(27) or Open Item 20.4-25 is closed.

Issue II.K.3(6): Instrumentation to Verify Natural Circulation

As discussed in NUREG-0933, Issue II.K.3(6), addressed requiring licensees to provide instrumentation to verify natural circulation during transient conditions. The staff determined in NUREG-0933 that this issue was covered by Issues I.C.1, II.F.2, and II.F.3.

Westinghouse has provided the staff with pertinent information about the AP600 design, which addresses TMI action items I.C.1, II.F.2 and II.F.3. The staff has reached a conclusion that those issues relevant to the resolution of the TMI action item II.K.3(6) have been resolved. The detailed discussion of the related issues are addressed in their respective TMI item discussions. Therefore, this issue is resolved.

Issue II.K.3(8): Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of Steam Generator

As discussed in NUREG-0933, Issue II.K.3(18) addressed further staff consideration of the need for diverse decay heat removal methods which were independent of the steam generators. The staff determined in NUREG-0933 that this issue was covered by Issues II.C.1, "Interim Reliability Evaluation Program," and II.E.3.3, "Coordinated Study of Shut-

down Heat Removal Requirements." In NUREG-0933, the staff also stated that Issue II.E.3.3 was addressed in Issue A-45, "Shutdown Decay Heat removal Requirements."

Westinghouse has provided the AP600 shutdown evaluation report for staff review. The report describes multiple decay heat removal capabilities independent of the steam generator. The detailed discussion of the multiple decay heat capabilities is included in Chapter 19.3 of the FSAR. The staff, therefore, concludes that Issue II.K.3(8) or Open Item 20.4-27 is closed.

Issue II.K.3(30): Revised SBLOCA Methods to Show Compliance with 10 CFR Part, Appendix K

As discussed in NUREG-0933, Issue II.K.3(30) required licensees to revise and submit analytical methods for small-break LOCA analyses for compliance with Appendix K to 10 CFR Part 50 for NRC review and approval. The revision was to account for comparisons with experimental data, including data from LOFT test and semiscale test facilities. Alternatively, licensees were to provide additional justification for the acceptability of their SBLOCA models with LOFT and semiscale test data. Clarifications were issued in NUREG-0737. The staff has reviewed NOTRUMP code and has documented its discussions in Chapters 15 and 21 of the FSER. The staff, therefore, considers this issue closed.

**BL/GL STATUS**

| BL/GL TITLE | DSER RESOLUTION | FSER RESOLUTION |
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| BL-80-12, Decay heat removal | This bulletin dealt with reducing the likelihood of losing DHR capability. W stated that this issue is discussed in SSAR 7.4.1 | The staff evaluates this issue in FSER Section 6.3. This GL is CLOSED. |
| BL-80-18, maintenance of adequate mini flow through CCP following secondary side high energy line rupture | W stated the design do not have CCP as part of SI and that it is not applicable to AP600. | N/A. CLOSED. |
| BL-86-01, mini flow logic problem that could disable RHR pumps | AP600 does not have valves in mini flow lines. Issue resolved | N/A. CLOSED. |
| BL-89-03, potential loss of required shutdown margin during refueling | The staff indicated that movement and placement of fuel during refueling is within the scope of AP600 core design. This issue is a COL action item | SSAR Section 9.1 discusses fuel storage and handling including the refueling equipment which is used to safely move and store fuels. Additionally, IRWST provides large quantities of borated water that maintains the required shutdown margin. during refueling. The BL-89-03 issue is CLOSED and the COL action item is still valid regarding plant specific guidelines. |
| GL-80-01, report on ECCS | The staff requested that W address NUREG-0630, "Cladding, Swelling and Rupture Models for LOCA Analysis." | The safety evaluation of this issue is in FSER Chapter 15. The GL-80-01 is CLOSED. |
| GL-80-014, LWR primary coolant system pressure isolation valves | W needed to indicate where in Section 1.9 of the SSAR that this issue is discussed | W stated that SSAR 1.9.4.1.2 and USi-B-63, discuss this issue. The staff discusses this issue in the ISLOCA context and is having an outstanding open issue (TMI/Issue 105). This is an OPEN ITEM. |

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| GL-80-019, resolution of enhanced fission gas release concern | W stated that the fission gas release models are accounted for in WCAP-10851-P-A and WCAP-11873-A, "improved fuel performance models for W fuel rod design and safety evaluations." This issue is resolved | The staff position has not changed. No action is needed. The GL-80-019 is CLOSED. |
| GL-81-021, Natural circulation cooldown | The staff stated that W should address the ERG for this event. | W has submitted AP600 ERG-GW-GJR-100, Rev 3 dated 5/97 for staff review. The staff has reviewed this submittal and its related natural circulation ERG and has determined that guidelines are sufficiently given to the operator to cool down the plant using natural circulation means. This GL is CLOSED. |
| GL-83-11, licensee qualifications for performing safety analyses in supporting licensing actions | W should address the qualifications for performing safety analysis for AP600 design. | This issue is COL responsibility. This GL is CLOSED. |
| GL-84-21, long-term, low-power operation in PWRs | Core peaking factor may be greater than assumed in safety analysis for extended low-power operation following a return to full power ops. | The safety evaluation of this issue is in FSER Chapter 15. This GL is CLOSED. |
| GL-85-16, high boron concentration | This GL is resolved because AP600 design does not have BIT and born concentration from CMT is much lower than BIT (22,000 ppm). | The staff position has not changed. No action is required. This GL is CLOSED. |
| GL-87-12, loss of RHR with RCS partially filled. | This GL addressed potential for loss of RHR during midloop operation | W has submitted WCAP-14837, Rev 2 (11/97) that discusses shutdown risk concerns, including potential loss of RNS. The staff has resolved this issue and its SE is discussed in FSER chapter 19. This GL is CLOSED. |

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| GL-88-17, loss of DHR | This GL addressed potential for loss of RHR during midloop operation | W has submitted WCAP-14837, Rev 2 (11/97) that discusses shutdown risk concerns, including potential loss of RNS. The staff has resolved this issue and its SE is discussed in FSER Chapter 19. This GL is CLOSED. |
| GL-91-07, RCP seal failures | GSI-23 discussed RCP pump seal failures. W addressed this issue in SSAR Sections 5.1.3.3, and 1.9.4.2.3 | The staff has resolved GSI-23 issue because W design does not have RCP pump seals, and the GL is not applicable to AP600 canned pump design. This GL is CLOSED. |
| GL-93-04, rod control system failure | W should revise the WCAP-13559 to include this item of discussion for AP600 design | WCAP has been revised (8/96) to include reference of this item of discussion in SSAR 3.9.4. The staff SE is discussed in FSER Chapter 4. This GL is CLOSED. |
| GL-83-22, safety evaluation of ERGs | THIS ISSUE WAS NOT INCLUDED IN DSER | The staff has reviewed W AP600 ERG-GW-GJR-100, Rev 3 dated 5/97 and has documented its evaluation in FSER Section 18.9.3. This GL is CLOSED. |
| GL-86-07, NUREG-1190 regarding the San Onofre Unit 1 loss of power and water hammer | THIS ISSUE WAS NOT INCLUDED IN DSER | This issue is not part of SRXB responsibility and it is most appropriately addressed by HHFB. |

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| BL-96-01, rod control problem | THIS ISSUE WAS NOT INCLUDED IN DSER | The BL was issued because of incomplete control rod insertion (IRI) evaluation at the South Texas and Wolf Creek plants. It has been determined that the IRI was caused by the thimble tube distortion resulting from excessive load. Since this is a fuel design problem, and W has not committed to any fuel types and that this problem is mostly resolved by the fuel manufacturers, the staff has concluded W does not have to address this issue, unless it has committed to certain fuel designs discussed in the BL. This issue should be appropriately addressed by the COL applicant. This is a new COL action item. The BL is CLOSED. |
| GL-86-16, ECCS evaluation models | The staff requested that W discuss the ECCS evaluation models for AP600 design. | W discussed this issue in SSAR Sections 6.3.5 and 15.0.11. The staff has evaluated Westinghouse ECCS models and has discussed this issue in FSER Chapter 15. This GL is CLOSED. |
| GL-85-05, Inadvertent Boron Dilution | | W discussed this issue in SSAR Section 15.4.6. The staff has evaluated and has discussed this issue in FSER Chapter 15. This GL is CLOSED. |
| GL-96-04, Boraflex and Spent Fuel Racks | | This issue is not part of SRXB responsibility. It is most appropriately addressed by SPLB. |

Note: All TMI Items ( II.B.1, II.G.1, II.K1(28), II.K.2(16), II.K.3(2), II.K.3(5) and II.K.3(25) indicated in the 12/10/97 note from J.Sebrosky have been resolved in the DSER. The staff does not see any changes in position regarding its evaluation of these items. Also, Issue 23 has been resolved and is reflected in DSER. No change in the staff position is anticipated.