ATTACHMENTS

1. *Statement of Policy: Further Guidance for Power Reactor Operating Licenses*, CLI-80-42, 12 NRC 654 (1980)

2. NRC, Backgrounder on the Three Mile Island Accident, www.nrc.gov/reading-rm/doc-collections/fact-sheets/3mile-isle.html

3. ACRS, *Report on the safety aspects of the Southern Nuclear Operating Company combined license application for Vogtle Electric Generating Plant*, Units 3 and 4, January 24, 2011.

4. Letter to Chairman Jaczko from Rep. Markey, March 7, 2011

5. Letter from Westinghouse to NRC, "AP1000 Containment Cleanliness - DCD Markup for Rev. 19," February 23, 2011.

6. Commission Voting Record, SECY-11-0002, February 9, 2011

7. ACRS, Long-term core cooling for the Westinghouse AP1000 pressurized water reactor, December 20, 2011

8. NRC's schedule for review of new reactor licensing applications. Available at www.nrc.gov/reactors/new-reactors/new-licensing-files/new-rx-licensing-app-legend.pdf

Cite as 12 NRC 654 (1980) **CLI-80-42**

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

COMMISSIONERS

John F. Ahearne, Chairman
Victor Gilinsky
Joseph M. Hendrie
Peter A. Bradford

In the Matter of

PR-Miscellaneous Notice (45 FR 417398)

STATEMENT OF POLICY: FURTHER COMMISSION GUIDANCE FOR POWER REACTOR OPERATING LICENSES

December 18, 1980

The Commission issues a revised Statement of Policy concerning the implementation of TMI-related requirements into the licensing process.

MEMORANDUM AND ORDER

Recently the Commission, by a vote of 3-2, issued a Statement of Policy entitled "Further Commission Guidance for Power Reactor Operating Licenses." 45 FR 41738 (June 20, 1980). In essence, the Statement of Policy announced the intent of the Commission that in future actions on nuclear power reactor operating license applications, it would look to the list of "Requirements for New Operating Licenses" found in NUREG-0694 (June 1980) as setting forth requirements for new operating licenses which should be "necessary and sufficient for responding" to the accident at Three Mile Island ("TMI"). Consequently, current operating license applications were to be judged against present NRC regulations, as supplemented by these TMI-related requirements. Insofar as certain of the provisions of NUREG-0694 sought to impose operating license requirements beyond those necessary to show compliance with the regulations:

> although the [licensing and appeal] boards may entertain contentions asserting that the supplementation is unnecessary (in full or in part) and they may entertain contentions that one or more of the supplementary requirements are not being complied with; they may not entertain contentions asserting that additional supplementation is required. *Id.*

On November 3, 1980, by a vote of 2-2, the Commission denied a request for a stay of the Statement of Policy filed by the Union of Concerned Scientists and the Shoreham Opponents Coalition.

On October 28, 1980, by a vote of 4-0, the Commission approved NUREG-0737, "Clarification of TMI Action Plan Requirements," which is a letter from D.G. Eisenhut, Director of the Division of Licensing, NRR, to licensees of operating power reactors and applicants for operating licenses forwarding post-TMI requirements. NUREG-0737 now supersedes NUREG-0694, the latter being the document which forms the core of the substantive requirements in the aforementioned Statement of Policy. NUREG-0737 makes numerous significant changes in NUREG-0694. In some instances, the requirements in NUREG-0694 are made more flexible, especially as to implementation schedules. In some instances, the requirements in NUREG-0694 are made more strict. In addition, NUREG-0737 adds new requirements, taken from previously issued Bulletins and Orders, which were not part of NUREG-0694.

The Commission's approval of NUREG-0737 requires that some changes be made in the previously adopted Statements of Policy. Moreover, the Commission has now had more time to reflect upon the distinction between interpretive and supplementary requirements, as originally set forth in NUREG-0694 and as modified in NUREG-0737, and believes that the number of supplementary requirements may be quite small. For these reasons, the Commission has decided that the Statement of Policy should be amended as set forth in the Appendix to this Memorandum and Order.[1]

It is so ORDERED.

For the Commission,

SAMUEL J. CHILK
Secretary of the Commission

Dated at Washington D.C.
this 18th day of December, 1980.

[1] Chairman Ahearne concurs in amending the policy statement, but disagrees in how it should be amended. His dissenting views are attached to the Appendix.

U.S. NUCLEAR REGULATORY COMMISSION

FURTHER COMMISSION GUIDANCE

FOR POWER REACTOR OPERATING LICENSES

REVISED STATEMENT OF POLICY

I. BACKGROUND

After the March 1979 accident at Three Mile Island, Unit 2, the Commission directed its technical review resources to assuring the safety of operating power reactors rather than to the issuance of new licenses. Furthermore, the Commission decided that power reactor licensing should not continue until the assessment of the TMI accident had been substantially completed and comprehensive improvements in both the operation and regulation of nuclear power plants had been set in motion.

At a meeting on May 30, 1979, the Nuclear Regulatory Commission decided to issue policy guidance addressing general principles for reaching licensing decisions and to provide specific guidance for near-term operating license cases.[1] In November 1979, the Nuclear Regulatory Commission issued the policy guidance in the form of an amendment to 10 CFR Part 2 of its regulations,[2] describing the approach to be taken by the Commission regarding licensing of power reactors. In particular, the Commission noted that it would "be providing case-by-case guidance on changes in regulatory policies." The Commission has now acted on four operating licenses, has given extensive consideration to issues arising as a result of the Three Mile Island accident, and is able to provide general guidance. Following the accident at Three Mile Island 2, the President established a Commission to make recommendations regarding changes necessary to improve nuclear safety. In May 1979, the Nuclear Regulatory Commission established a Lessons Learned Task Force,[3] to determine what actions were required for new operating licenses and chartered a Special Inquiry Group to examine

[1]"Staff Requirements - Discussion of Options Regarding Deferral of Licenses," memorandum from Samuel J. Chilk, Secretary to Lee V. Gossick, Executive Director for Operations, May 31, 1979.

[2]"Suspension of 10 CFR 2.764 and Statement of Policy on Conduct of Adjudicatory Proceedings," 44 FR 65050 (November 9, 1979).

[3]"Lessons Learned from TMI-2 Accident," Roger Mattson to NRR staff, May 31, 1979.

all facets of the accident and its causes. These groups have published their reports.[4]

The Lessons Learned Task Force led to NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" and NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report." The Commission addressed these reports in meetings on September 6, September 14, October 14, and October 16, 1979. Following release of the report of the Presidential Commission the Commission provided a preliminary set of responses to the recommendations in that report.[5] This response provided broad policy directions for development of an NRC Action Plan, work on which was begun in November 1979. During the development of the Action Plan, the Special Inquiry Group Report was received, which had the benefit of review by panels of outside consultants representing a cross section of technical and public views. This report provided additional recommendations. The Action Plan[6] was developed to provide a comprehensive and integrated plan for the actions judged appropriate by the Nuclear Regulatory Commission to correct or improve the regulation and operation of nuclear facilities based on the experience from the accident at TMI-2 and the official studies and investigations of the accident. In developing the Action Plan, the various recommendations and possible actions of all the principal investigations were assessed and either rejected, adopted or modified. A detailed summary of the development and review process for the Action Plan was initially provided in NUREG-0694,[7] "TMI-Related Requirements For New Operating Licenses," and can now be found, as changed, in NUREG-0737, "Clarification of TMI Action Plan Requirements."[8]

[4]Report of the President's Commission on The Accident at Three Mile Island, "The Need for Change: The Legacy of TMI," October 1979;
U.S. Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0578, July 1979;
U.S. Nuclear Regulatory, "TMI-2 Lessons Learned Task Force Status Report, " NUREG-0585, August 1979;
U.S. Nuclear Regulatory Commission Special Inquiry Group, "Three Mile Island: A Report to the Commissioners and to the Public," January 1980.

[5]U.S. Nuclear Regulatory Commission, "NRC Views and Analysis of the Recommendations of the President's Commission on the Accident at Three Mile Island," NUREG-0632, November 1979.

[6]U.S. Nuclear Regulatory Commission, "NRC Action Plans Developed as a Result of the TMI-2 Accident," NUREG-0660.

[7]U.S. Nuclear Regulatory Commission, "TMI-Related Requirements for New Operating Licenses," NUREG-0694, June 1980.

[8]U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.

Actions to improve the safety of nuclear power plants now operating were judged to be necessary immediately after the accident and could not be delayed until the Action Plan was developed, although they were subsequently included in the Action Plan. Such actions came from the Bulletins and Orders issued immediately after the accident, the first report of the Lessons-Learned Task Force issued in July 1979, the recommendations of the Emergency Preparedness Task Force, and the NRC staff and Commission. Before these immediate actions were applied to operating plants, they were approved by the Commission. Many of the required immediate actions have already been taken by licensees and most are scheduled to be completed in the near future.

On February 7, 1980, based on its review of initial drafts of the Action Plan, the Commission approved a listing of near-term operating license (NTOL) requirements, as being necessary but not necessarily sufficient TMI-related requirements, for granting new operating licenses. Since then, the fuel load requirements on the NTOL list have been used by the Commission in granting operating licenses, with limited authorizations for fuel loading and low power testing, for Sequoyah, North Anna, Salem, and Farley. Full operating licenses were granted, based on the NTOL list, for Sequoyah and North Anna.

On May 15, 1980, after review of the last version of the Action Plan, the Commission approved a list of "Requirements For New Operating Licenses," contained in NUREG-0694, which the staff recommended for imposition on current operating license applicants. That list was recast from the previous NTOL list and sets forth four types of TMI-related requirements and actions for new operating licenses: (1) those required to be completed by a license applicant prior to receiving a fuel-loading and low-power testing license, (2) those required to be completed by a license applicant to operate at appreciable power levels up to full power, (3) those the NRC will take prior to issuing a fuel-loading and low-power testing or full-power operating license, and (4) those required to be completed by a licensee prior to a specified date.

The Commission also approved the staff's recommendation that the remaining items from the TMI reviews should be implemented or considered over time to further enhance safety.

On October 28, 1980, the Commission approved a "Clarification of TMI Action Plan Requirements," now contained in NUREG-0737, which supersedes NUREG-0694. More explicit requirements, revisions in previous requirements, different time schedules for implementation, and new

requirements in NUREG-0694, but taken from previously issued Commission bulletins and orders, form the core of NUREG-0737.

In approving the schedules for developing and implementing changes in requirements, the Commission's primary considerations were the safety significance of the issues and the immediacy of the need for corrective actions. As discussed above, many actions were taken to improve safety immediately or soon after the accident. These actions were generally considered to be interim improvements. In scheduling the remaining improvements, the availability of both NRC and industry resources was considered, as well as the safety significance of the actions. Thus, the Action Plan approved by the Commission presents a sequence of actions that will result in a gradually increasing improvement in safety as individual actions are completed and the initial immediate actions are replaced or supplemented by longer term improvements.

II. COMMISSION DECISION

Based upon its extensive review and consideration of the issues arising as a result of the Three Mile Island accident — a review that is still continuing — the Commission has concluded that the list of TMI-related requirements for new operating licenses found in NUREG-0737 can provide a basis for responding to the TMI-2 accident. The Commission has decided that current operating license applications should be measured by the NRC staff against the regulations, as augmented by these requirements.[9] In general, the remaining items of the Action Plan should be addressed through the normal process for development and adoption of new requirements rather than through immediate imposition on pending applications.

III. LITIGATION OF TMI-2 ISSUES IN OPERATING LICENSE PROCEEDINGS

In the November 1979 policy statement, the Commission provided the following guidance for the conduct of adjudicatory proceedings:

> In reaching their decisions, the Boards should interpret existing regulations and regulatory policies with due consideration to the implications for those regulations and policies of the Three Mile Island Accident. In this regard, it should be understood that as a result of analyses still underway, the Commission may change its present regulations and regulatory policies in important aspects and thus compliance with existing regulations may turn out to no longer warrant approval of a license application.

[9]Consideration of applications for an operating license should include the entire list of requirements unless an applicant specifically requests an operating license with limited authorization (e.g., fuel loading and low-power testing).

The Commission is now able to give the Boards more guidance.

The Commission believes the TMI-related operating license requirements list as derived from the process described above should be the principal basis for consideration of TMI-related issues in the adjudicatory process. There are good reasons for this. First, this represents a major effort by the staff and Commissioners to address more than one hundred issues and recommendations in a coherent and coordinated fashion. This entire process cannot be reproduced in individual proceedings. Second, the NRC does not have the resources to litigate the entire Action Plan in each proceeding. Third, many of the decisions involve policy more than factual or legal decisions. Most of these are more appropriately addressed by the Commission itself on a generic basis than by an individual licensing board in a particular case. Consequently, the Commission has chosen to adopt the following policy regarding litigation of TMI-related issues in operating license proceedings.

The "Clarification of Action Plan Requirements" in NUREG-0737, like the TMI-related "Requirements For New Operating Licenses" in NUREG-0694, can, in terms of their relationship to existing Commission regulations, be put in two categories: (1) those that interpret, refine or quantify the general language of existing regulations, and (2) those that supplement the existing regulations by imposing requirements in addition to specific ones already contained therein. Insofar as the first category — refinement of exisiting regulations — is concerned, the parties may challenge the new requirements as unnecessary on the one hand or insufficient on the other within the limits of the regulations. Insofar as the second category — supplementation of existing regulations — is concerned, the parties may challenge either the necessity for or sufficiency of such requirements. It would be useful if the parties in taking a position on such requirements stated (a) the nexus of the issue to the TMI-2 accident, (b) the significance of the issue, and, (c) any differences between their positions and the rationale underlying the Commission consideration of additional TMI-related requirements. It would be helpful if any certifications of questions regarding such positions to the Commission included the same information and such certifications are encouraged where Boards are in doubt as to the Commission's intentions in approving NUREG-0737. The Atomic Safety and Licensing and Appeal Boards' present authority to raise issues *sua sponte* under 10 CFR 2.760a extends to both categories.

In order to focus litigation of TMI-related issues, the staff and the Boards should use the Commission's existing summary disposition procedures, where applicable, in responding to TMI-related contentions.

The Commission believes that where the time for filing contentions has expired in a given case, no new TMI-related contentions should be accepted absent a showing of good cause and balancing of the factors in 10 CFR 2.714(a)(1). The Commission expects adherence to its regulations in this regard.

Also, present standards governing the reopening of hearing records to consider new evidence on TMI-related issues should be adhered to. Thus, for example, where initial decisions have been issued, the record should not be reopened to take evidence on some TMI-related issue unless the party seeking reopening shows that there is significant new evidence, not included in the record, that materially affects the decision.

Finally, the Commission will continue to monitor developments with regard to the litigation of our Action Plan requirements and will continue to offer guidance where appropriate.

Samuel J. Chilk
Secretary of the Commission

Dated at Washington, D.C.
the 18th day of December 1980.

Backgrounder on the Three Mile Island Accident

Printable Version

On this page:

The accident at the Three Mile Island Unit 2 (TMI-2) nuclear power plant near Middletown, Pa., on March 28, 1979, was the most serious in U.S. commercial nuclear power plant operating history, even though it led to no deaths or injuries to plant workers or members of the nearby community. But it brought about sweeping changes involving emergency response planning, reactor operator training, human factors engineering, radiation protection, and many other areas of nuclear power plant operations. It also caused the U.S. Nuclear Regulatory Commission to tighten and heighten its regulatory oversight. Resultant changes in the nuclear power industry and at the NRC had the effect of enhancing safety.

The sequence of certain events – equipment malfunctions, design-related problems and worker errors – led to a partial meltdown of the TMI-2 reactor core but only very small off-site releases of radioactivity.

Summary of Events

The accident began about 4:00 a.m. on March 28, 1979, when the plant experienced a failure in the secondary, non-nuclear section of the plant. The main feedwater pumps stopped running, caused by either a mechanical or electrical failure, which prevented the steam generators from removing heat. First the turbine, then the reactor automatically shut down. Immediately, the pressure in the primary system (the nuclear portion of the plant) began to increase. In order to prevent that pressure from becoming excessive, the pilot-operated relief valve (a valve located at the top of the pressurizer) opened. The valve should have closed when the pressure decreased by a certain amount, but it did not. Signals available to the operator failed to show that the valve was still open. As a result, cooling water poured out of the stuck-open valve and caused the core of the reactor to overheat.

As coolant flowed from the core through the pressurizer, the instruments available to reactor operators provided confusing information. There was no instrument that showed the level of coolant in the core. Instead, the operators judged the level of water in the core by the level in the pressurizer, and since it was high, they assumed that the core was properly covered with coolant. In addition, there was no clear signal that the pilot-operated relief valve was open. As a result, as alarms rang and warning lights flashed, the operators did not realize that the plant was experiencing a loss-of-coolant accident. They took a series of actions that made conditions worse by simply reducing the flow of coolant through the core.

Because adequate cooling was not available, the nuclear fuel overheated to the point at which the zirconium cladding (the long metal tubes which hold the nuclear fuel pellets) ruptured and the fuel pellets began to melt. It was later found that about one-half of the core melted during the early stages of the accident. Although the TMI-2 plant suffered a severe core meltdown, the most dangerous kind of nuclear power accident, it did not produce the worst-case consequences that reactor experts had long feared. In a worst-case accident, the melting of nuclear fuel would lead to a breach of the walls of the containment building and release massive quantities of radiation to the environment. But this did not occur as a result of the three Mile Island accident.

The accident caught federal and state authorities off-guard. They were concerned about the small releases of radioactive gases that were measured off-site by the late morning of March 28 and even more concerned about the potential threat that the reactor posed to the surrounding population. They did not know that the core had melted, but they immediately took steps to try to gain control of the reactor and ensure adequate cooling to the core. The NRC=s regional office in King of Prussia, Pa., was notified at 7:45 a.m. on March 28. By 8:00, NRC Headquarters in Washington, D.C., was alerted and the NRC Operations Center in Bethesda, Md., was activated. The regional office promptly dispatched the first team of inspectors to the site and other agencies, such as the Department of Energy and the Environmental Protection Agency, also mobilized their response teams. Helicopters hired by TMI's owner, General Public Utilities Nuclear, and the Department of Energy were sampling radioactivity in the atmosphere above the plant by midday. A team from the Brookhaven National Laboratory was also sent to assist in radiation monitoring. At 9:15 a.m., the White House was notified and at 11:00 a.m., all non-essential personnel were ordered off the plant's premises.

By the evening of March 28, the core appeared to be adequately cooled and the reactor appeared to be stable. But new concerns arose by the morning of Friday, March 30. A significant release of radiation from the plant=s auxiliary building, performed to relieve pressure on the primary system and avoid curtailing the flow of coolant to the core, caused a great deal of confusion and consternation. In an atmosphere of growing uncertainty about the condition of the plant, the governor of Pa., Richard L. Thornburgh, consulted with the NRC about evacuating the population near the plant. Eventually, he and

NRC Chairman Joseph Hendrie agreed that it would be prudent for those members of society most vulnerable to radiation to evacuate the area. Thornburgh announced that he was advising pregnant women and pre-school-age children within a 5-mile radius of the plant to leave the area.

Within a short time, the presence of a large hydrogen bubble in the dome of the pressure vessel, the container that holds the reactor core, stirred new worries. The concern was that the hydrogen bubble might burn or even explode and rupture the pressure vessel. In that event, the core would fall into the containment building and perhaps cause a breach of containment. The hydrogen bubble was a source of intense scrutiny and great anxiety, both among government authorities and the population, throughout the day on Saturday, March 31. The crisis ended when experts determined on Sunday, April 1, that the bubble could not burn or explode because of the absence of oxygen in the pressure vessel. Further, by that time, the utility had succeeded in greatly reducing the size of the bubble.

Health Effects

Detailed studies of the radiological consequences of the accident have been conducted by the NRC, the Environmental Protection Agency, the Department of Health, Education and Welfare (now Health and Human Services), the Department of Energy, and the State of Pa.. Several independent studies have also been conducted. Estimates are that the average dose to about 2 million people in the area was only about 1 millirem. To put this into context, exposure from a chest x-ray is about 6 millirem. Compared to the natural radioactive background dose of about 100-125 millirem per year for the area, the collective dose to the community from the accident was very small. The maximum dose to a person at the site boundary would have been less than 100 millirem.

In the months following the accident, although questions were raised about possible adverse effects from radiation on human, animal, and plant life in the TMI area, none could be directly correlated to the accident. Thousands of environmental samples of air, water, milk, vegetation, soil, and foodstuffs were collected by various groups monitoring the area. Very low levels of radionuclides could be attributed to releases from the accident. However, comprehensive investigations and assessments by several well-respected organizations have concluded that in spite of serious damage to the reactor, most of the radiation was contained and that the actual release had negligible effects on the physical health of individuals or the environment.

Impact of the Accident

The accident was caused by a combination of personnel error, design deficiencies, and component failures. There is no doubt that the accident at Three Mile Island permanently changed both the nuclear industry and the NRC. Public fear and distrust increased, NRC's regulations and oversight became broader and more robust, and management of the plants was scrutinized more carefully. The problems identified from careful analysis of the events during those days have led to permanent and sweeping changes in how NRC regulates its licensees – which, in turn, has reduced the risk to public health and safety.

Here are some of the major changes which have occurred since the accident:

- Upgrading and strengthening of plant design and equipment requirements. This includes fire protection, piping systems, auxiliary feedwater systems, containment building isolation, reliability of individual components (pressure relief valves and electrical circuit breakers), and the ability of plants to shut down automatically;
- Identifying human performance as a critical part of plant safety, revamping operator training and staffing requirements, followed by improved instrumentation and controls for operating the plant, and establishment of fitness-for-duty programs for plant workers to guard against alcohol or drug abuse;
- Improved instruction to avoid the confusing signals that plagued operations during the accident;
- Enhancement of emergency preparedness to include immediate NRC notification requirements for plant events and an NRC operations center that is staffed 24 hours a day. Drills and response plans are now tested by licensees several times a year, and state and local agencies participate in drills with the Federal Emergency Management Agency and NRC;
- Establishment of a program to integrate NRC observations, findings, and conclusions about licensee performance and management effectiveness into a periodic, public report;
- Regular analysis of plant performance by senior NRC managers who identify those plants needing additional regulatory attention;
- Expansion of NRC's resident inspector program – first authorized in 1977 – whereby at least two inspectors live nearby and work exclusively at each plant in the U.S. to provide daily surveillance of licensee adherence to NRC regulations;
- Expansion of performance-oriented as well as safety-oriented inspections, and the use of risk assessment to identify vulnerabilities of any plant to severe accidents;
- Strengthening and reorganization of enforcement as a separate office within the NRC;
- The establishment of the Institute of Nuclear Power Operations (INPO), the industry's own "policing" group, and formation of what is now the Nuclear Energy Institute to provide a unified industry approach to generic nuclear regulatory issues, and interaction with NRC and other government agencies;
- The installing of additional equipment by licensees to mitigate accident conditions, and monitor radiation levels and plant status;
- Employment of major initiatives by licensees in early identification of important safety-related problems, and in collecting and assessing relevant data so lessons of experience can be shared and quickly acted upon; and
- Expansion of NRC's international activities to share enhanced knowledge of nuclear safety with other countries in a number of important technical areas.

Current Status

Today, the TMI-2 reactor is permanently shut down and defueled, with the reactor coolant system drained, the radioactive water decontaminated and evaporated, radioactive waste shipped off-site to an appropriate disposal site, reactor fuel and core debris shipped off-site to a Department of Energy facility, and the remainder of the site being monitored. In 2001, FirstEnergy acquired TMI-2 from GPU. FirstEnergy has contracted the monitoring of TMI-2 to Exelon, the current owner and operator of TMI-1. The companies plan to keep the TMI-2 facility in long-term, monitored storage until the operating license for the TMI-1 plant expires, at which time both plants will be decommissioned.

Below is a chronology of highlights of the TMI-2 cleanup from 1980 through 1993.

Date	**Event**
July 1980	Approximately 43,000 curies of krypton were vented from the reactor building.
July 1980	The first manned entry into the reactor building took place.
Nov. 1980	An Advisory Panel for the Decontamination of TMI-2, composed of citizens, scientists, and State and local officials, held its first meeting in Harrisburg, PA.
July 1984	The reactor vessel head (top) was removed.
Oct. 1985	Defueling began.
July 1986	The off-site shipment of reactor core debris began.
Aug. 1988	GPU submitted a request for a proposal to amend the TMI-2 license to a "possession-only" license and to allow the facility to enter long-term monitoring storage.
Jan. 1990	Defueling was completed.
July 1990	GPU submitted its funding plan for placing $229 million in escrow for radiological decommissioning of the plant.
Jan. 1991	The evaporation of accident-generated water began.
April 1991	NRC published a notice of opportunity for a hearing on GPU's request for a license amendment.
Feb. 1992	NRC issued a safety evaluation report and granted the license amendment.
Aug. 1993	The processing of 2.23 million gallons accident-generated water was completed.
Sept. 1993	NRC issued a possession-only license.
Sept. 1993	The Advisory Panel for Decontamination of TMI-2 held its last meeting.
Dec. 1993	Post-Defueling Monitoring Storage began.

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Additional Information

Further information on the TMI-2 accident can be obtained from sources listed below. The documents can be ordered for a fee from the NRC's Public Document Room at 301-415-4737 or 1-800-397-4209; e-mail pdr.resource@nrc.gov. The PDR is located at 11555 Rockville Pike, Rockville, Maryland; however the mailing address is: U.S. Nuclear Regulatory Commission, Public Document Room, Washington, D.C. 20555. A glossary is also provided below.

Additional Sources for Information on Three Mile Island

NRC Annual Report - 1979, NUREG-0690, "Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station," NUREG-0558

"Environmental Assessment of Radiological Effluents from Data Gathering and Maintenance Operation on Three Mile Island Unit 2," NUREG-0681

"Report of The President's Commission on The Accident at Three Mile Island," October, 1979

"Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement," NUREG-0600

"Three Mile Island; A Report to the Commissioners and to the Public," by Mitchell Rogovin and George T. Frampton, NUREG/CR-1250, Vols. I-II, 1980

"Lessons learned From the Three Mile Island - Unit 2 Advisory Panel," NUREG/CR-6252

"The Status of Recommendations of the President's Commission on the Accident at Three Mile Island," (A ten-year review), NUREG-1355

"NRC Views and Analysis of the Recommendations of the President's Commission on the Accident at Three Mile Island," NUREG-0632

"Environmental Impact Statement related to decontamination and disposal of radioactive wastes resulting from March 28, 1979 accident Three Mile Island Nuclear Station, Unit 2," NUREG-0683

"Answers to Questions About Updated Estimates of Occupational Radiation Doses at Three Mile Island, Unit 2," NUREG-1060

"Answers to Frequently Asked Questions About Cleanup Activities at Three Mile Island, Unit 2," NUREG-0732

"Status of Safety Issues at Licensed Power Plants" (TMI Action Plan Reqmts.), NUREG-1435

Walker, J. Samuel, **Three Mile Island: A Nuclear Crisis in Historical Perspective**, Berkeley: University of California Press, 2004.

TOP

Glossary

Auxiliary feedwater - (see emergency feedwater)

Background radiation - The radiation in the natural environment, including cosmic rays and radiation from the naturally radioactive elements, both outside and inside the bodies of humans and animals. The usually quoted average individual exposure from background radiation is 300 millirem per year.

Cladding - The thin-walled metal tube that forms the outer jacket of a nuclear fuel rod. It prevents the corrosion of the fuel by the coolant and the release of fission products in the coolants. Aluminum, stainless steel and zirconium alloys are common cladding materials.

Emergency feedwater system - Backup feedwater supply used during nuclear plant startup and shutdown; also known as auxiliary feedwater.

Fuel rod - A long, slender tube that holds fuel (fissionable material) for nuclear reactor use. Fuel rods are assembled into bundles called fuel elements or fuel assemblies, which are loaded individually into the reactor core.

Containment - The gas-tight shell or other enclosure around a reactor to confine fission products that otherwise might be released to the atmosphere in the event of an accident.

Coolant - A substance circulated through a nuclear reactor to remove or transfer heat. The most commonly used coolant in the U.S. is water. Other coolants include air, carbon dioxide, and helium.

Core - The central portion of a nuclear reactor containing the fuel elements, and control rods.

Decay heat - The heat produced by the decay of radioactive fission products after the reactor has been shut down.

Decontamination - The reduction or removal of contaminating radioactive material from a structure, area, object, or person. Decontamination may be accomplished by (1) treating the surface to remove or decrease the contamination; (2) letting the material stand so that the radioactivity is decreased by natural decay; and (3) covering the contamination to shield the radiation emitted.

Feedwater - Water supplied to the steam generator that removes heat from the fuel rods by boiling and becoming steam. The steam then becomes the driving force for the turbine generator.

Nuclear Reactor - A device in which nuclear fission may be sustained and controlled in a self-supporting nuclear reaction. There are several varieties, but all incorporate certain features, such as fissionable material or fuel, a moderating material (to control the reaction), a reflector to conserve escaping neutrons, provisions for removal of heat, measuring and controlling instruments, and protective devices.

Pressure Vessel - A strong-walled container housing the core of most types of power reactors.

Pressurizer - A tank or vessel that controls the pressure in a certain type of nuclear reactor.

Primary System - The cooling system used to remove energy from the reactor core and transfer that energy either directly or indirectly to the steam turbine.

Radiation - Particles (alpha, beta, neutrons) or photons (gamma) emitted from the nucleus of an unstable atom as a result of radioactive decay.

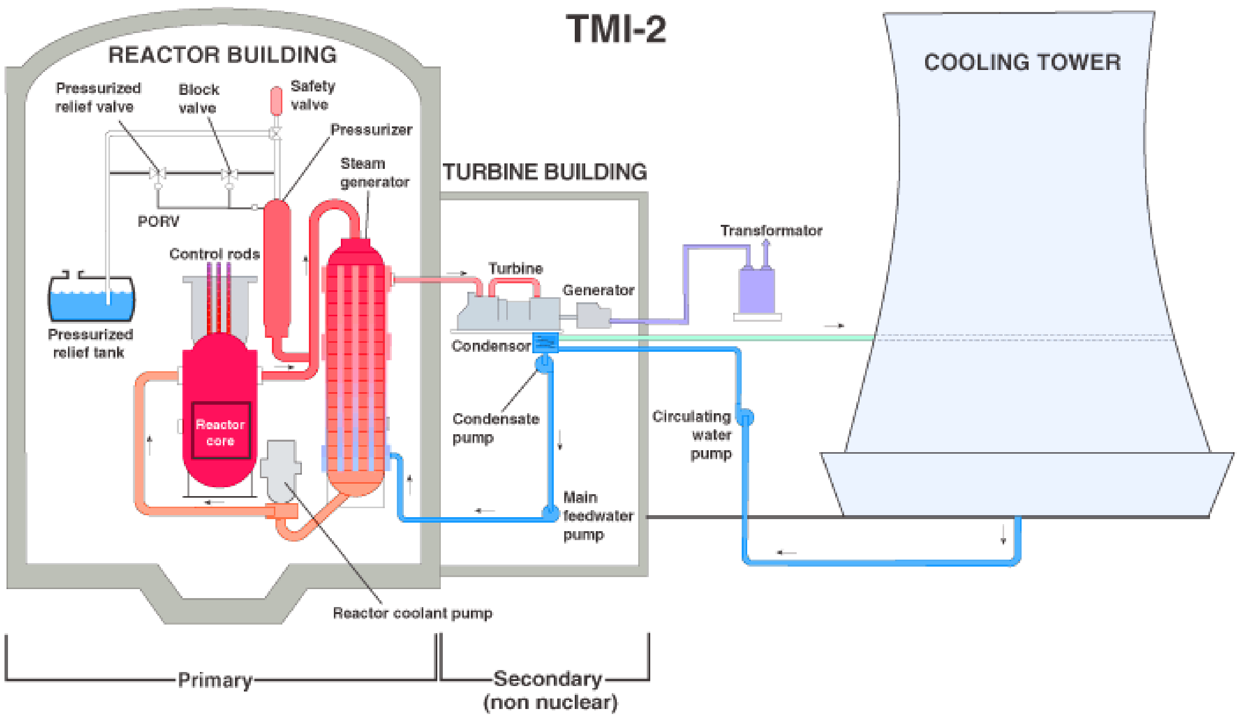
Reactor Coolant System - (see primary system)

Secondary System - The steam generator tubes, steam turbine, condenser and associated pipes, pumps, and heaters used to convert the heat energy of the reactor coolant system into mechanical energy for electrical generation.

Steam Generator - The heat exchanger used in some reactor designs to transfer heat from the primary (reactor coolant) system to the secondary (steam) system. This design permits heat exchange with little or no contamination of the secondary system equipment.

Turbine - A rotary engine made with a series of curved vanes on a rotating shaft. Usually turned by water or steam. Turbines are considered to be the most economical means to turn large electrical generators.

TOP



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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

January 24, 2011

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: REPORT ON THE SAFETY ASPECTS OF THE SOUTHERN NUCLEAR OPERATING COMPANY COMBINED LICENSE APPLICATION FOR VOGTLE ELECTRIC GENERATING PLANT, UNITS 3 AND 4

Dear Chairman Jaczko:

During the 579th meeting of the Advisory Committee on Reactor Safeguards (ACRS), January 13-15, 2011, we reviewed the NRC staff's Advanced Safety Evaluation Report (ASER) for the pending Southern Nuclear Operating Company (SNC) Combined License Application (COLA) for Vogtle Electric Generating Plant (VEGP), Units 3 and 4. This COLA incorporates by reference the Westinghouse Electric Company (WEC) AP1000 Design Certification Amendment (DCA) application and SNC VEGP Early Site Permit (ESP). Our AP1000 subcommittee also held four meetings (June 24-25, July 21-22, September 20-21, and December 15-16, 2010) to review various chapters of the COLA and the staff's ASER. During these meetings, we had the benefit of discussions with representatives of the NRC staff, NuStart Energy Development, LLC (NuStart)[1], SNC, SNC's supporting vendors, and the public. We also had the benefit of the documents referenced. This report fulfills the requirement of 10 CFR 52.53 that the ACRS report on those portions of the application which concern safety.

CONCLUSION AND RECOMMENDATIONS

1. There is reasonable assurance that VEGP, Units 3 and 4, can be built and operated without undue risk to the health and safety of the public. The SNC COLA for VEGP should be approved following its final revision.

2. The containment interior cleanliness limits on latent debris should be included in the Technical Specifications.

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[1]NuStart is a multi-utility consortium group. Each of the current and planned combined license applicants referencing the AP1000 reactor design is a member of NuStart.

3. A regulatory requirement focused on the development of an operational in-service inspection/in-service testing (ISI/IST) program for squib valves should be established, including a review of the lessons-learned from the design and qualification process for these valves.

4. An explicit requirement should be established to assure the accuracy of the feedwater flow measurement by in-plant testing.

5. The staff should review with us the changes in design or commitments that are not yet incorporated in the COLA or referenced in the Design Control Document (DCD), which significantly deviate from those presented during our review.

BACKGROUND

By letter dated March 28, 2008, SNC submitted an application to the U.S. Nuclear Regulatory Commission (NRC) for a combined license for VEGP, Units 3 and 4, in accordance with the requirements of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." In the application, SNC stated that VEGP, Units 3 and 4, would be two Westinghouse AP1000 advanced passive pressurized water reactors and would be located adjacent to the sites of the operating reactors (VEGP, Units 1 and 2). By letter dated April 28, 2009, NuStart informed the NRC that the AP1000 Design-Centered Work Group has designated the SNC COLA for VEGP, Units 3 and 4, as the AP1000 reference plant.

DISCUSSION

Containment Vessel Exterior Surface

The containment vessel (CV) exterior is subject to a continual flow of outside air, which is an inherent passive safety feature of the AP1000 design. The annular space between the CV and the surrounding shield building includes a baffle to direct air flow; water distribution weirs and associated dams, distribution boxes, and supports; and structures to provide personnel access for inspection and maintenance of the CV exterior. The inorganic zinc exterior coating of the 1.75 in. thick steel CV is of particular interest due to its importance in protecting the pressure boundary from corrosion.

The potential for airborne debris to accumulate on surfaces and in crevices to facilitate undetected corrosion of the CV was reviewed. SNC described the CV exterior coating inspection and maintenance program, which complies with 10 CFR Part 50 Appendix B, applicable ASTM standards, and regulatory guidance. This program is acceptable and is expected to ensure against undetected corrosion of the CV pressure boundary.

Also, the potential for debris to accumulate and impede the performance of the CV exterior water distribution system and cooling during an accident was reviewed. Protective screens and grates are provided in the design which, in combination with in-service inspection of the containment exterior, will ensure acceptable performance.

Containment Interior Debris Limitation

In our December 20, 2010, letter we concluded that the long-term core cooling requirements were adequately met, provided that the stringent cleanliness requirements specified for the containment interior is maintained. These requirements should not be relaxed without additional analyses, a much wider range of experiments at prototypical conditions, and NRC review.

The cleanliness requirements during operation, limiting latent debris to not more than 59 kg of which not more than 3 kg is fiber, are challenging but achievable. In order to ensure that they are not relaxed during plant life without consideration by the NRC staff of the provisions stated in our letter and to make them highly visible to both the plant operators and to the NRC staff, we recommend that the requirements be included in the plant Technical Specifications. We make this recommendation due to the importance these limits have in this instance, recognizing that debris limits are normally not part of the Technical Specifications.

ISI/IST Program Requirements for Squib Valves

The Automatic Depressurization System (ADS) ADS-4 squib valves must operate to achieve post loss-of-coolant accident (LOCA) passive long-term cooling. They are actuated by an explosive charge and are one-time-use valves until the internals are replaced. The development of an effective ISI/IST program to assure operability of the valves is needed. Periodic removal and firing of the explosive charge that initiates operation of the valve may not be sufficient for these critical components. SNC stated that, jointly with Westinghouse, it will develop ISI/IST procedures based on the final valve design and lessons-learned from the valve qualification process. While the AP1000 DCD includes Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) to confirm squib valve qualification, we recommend that a regulatory requirement be established focused on the development of the ISI/IST program, including a review of the lessons-learned from the valve design and qualification process.

Seismic Margin Analysis

The VEGP site-specific safe shutdown earthquake (SSE) design response spectra are the site-specific ground motion response spectra (GMRS) approved in the ESP. The GMRS slightly exceeded the certified seismic design response spectrum (CSDRS) in the lower frequency range. Therefore, in accordance with provisions in the DCD, plant-specific seismic evaluations were performed to demonstrate that the AP1000 plant designed for the CSDRS was acceptable for the VEGP site.

SNC performed an alternative site-specific analysis of soil-structure interaction using a three-dimensional model that uses the operating basis earthquake damping values of 4% specified in Regulatory Guide (RG) 1.61. The result indicated that the VEGP GMRS excitation will not compromise structures, systems, and components (SSC) under design-basis loads.

In response to a request for additional information, SNC provided additional seismic margin analyses confirming that the AP1000 certified design meets the 1.67 margin specified in SECY-93-087 at the VEGP site. A review-level earthquake equal to 0.5g was set for the seismic margin analysis and used to demonstrate the specified margin over the SSE of 0.3g. SNC also conducted a seismic margin analysis demonstrating that site-specific high confidence of low probability of failure values are equal to or greater than 1.67 times the GMRS of the design-basis SSE. Further, SNC completed a site-specific analysis of phenomena with the potential to reduce seismic margin. Evaluations were made of the potential for soil liquefaction and its effect on bearing capacity as well as nuclear island demand and seismic stability. The results of these additional analyses also demonstrated an adequate seismic margin of 1.67 times the VEGP GMRS, in accordance with SECY-93-087.

<u>Technical Support Center</u>

In a departure from the certified design, the SNC COLA provides for the Technical Support Center (TSC) for the new Units 3 and 4 to be combined with that for the existing Units 1 and 2 in a central Communication Support Center located between the power blocks for Units 2 and 3. This was reflected in the approved ESP, and human factors considerations for the combined TSC were discussed in the COLA review. However, insufficient detail is available at this time to evaluate how the TSC will function to assure that the four units, of two different designs, will be effectively supported in an emergency affecting one or more units. The COLA includes an ITAAC to demonstrate the capability of the TSC equipment and data displays to clearly identify and reflect the affected unit.

The staff should review with us the need for generic design guidance to assure adequate display of information at a multi-unit TSC.

During our review of the VEGP cyber security plan (CSP), we noted that the level of protection designated for the TSC (Level 2) was less than that for the respective units (Level 3 or 4). While it is recognized that control function decisions will be made only in the plant, and that the TSC is limited to advisory and management functions, this difference raised a concern as to the possible consequences during an emergency response if the information displayed in the TSC was corrupted as a result of the lower level of cyber security assigned. Since the CSP is consistent with RG 5.71 guidance, this is a potentially generic concern. The staff stated that this would be addressed in an ACRS Digital I&C Systems subcommittee meeting planned in the near future. We look forward to this further review of the appropriate level of protection for the TSC.

<u>Power Measurement Uncertainty</u>

The amended DCD states that the combined license holder will calculate the primary power calorimetric uncertainty using, "…an NRC acceptable method and confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated values." The initial reactor power for a large-break LOCA, as well as for certain mass and energy release calculations, is assumed to be within 1% of the licensed power. To measure power, SNC proposes to use a secondary side heat balance which requires measurement of certain pressures, temperatures, and flow rates. The largest contributor to uncertainty in the estimate of power is the measurement of the feedwater flow rate.

The Caldon Check Plus™ Leading Edge Flow Meter (LEFM), which is an ultrasonic flow measurement system, will be used to measure feedwater flow rate. The staff has approved this device to support a 1% power measurement uncertainty, provided two criteria for a newly constructed system are met. SNC proposes to address these criteria using an ITAAC to confirm that the instrumentation has been installed correctly, a License Condition to provide confirmation that the administrative controls are in place, and some COLA changes to be incorporated in a future application revision.

One of the criteria allows for use of a calibrated LEFM, where calibration was performed off-site at a lower Reynolds number than would exist in the plant, provided that acceptable justification is provided. Part of this justification is provided by confirmatory in-plant tests following installation. These tests assure that actual performance is within the uncertainty bounds established for the instrumentation.

The NRC should require that SNC make an explicit commitment to perform calibrations with representative piping configurations and conduct in-plant confirmatory tests.

Site-Specific PRA

We expected the COLA PRAs to be revised to include all available plant and site-specific information. This is not the case for the SNC COLA because Chapter 19 of the AP1000 DCD provides guidance to combined license applicants to identify plant-specific information and compare it with specified interface requirements. If the interface requirements are satisfied, the DCD PRA results will be conservative and are considered adequate for the COLA PRA. We find such a bounding approach acceptable at the combined license stage, given that substantial plant-specific, as-built information is not yet available.

NRC regulations require a full-scope, plant-specific PRA before fuel load. This PRA should meet the criteria of RG 1.200, providing a realistic picture of the plant risk, including uncertainty. The passive safety features of the AP1000 design were developed to eliminate or greatly reduce many of the more important contributors to plant risk. However, this improvement in risk comes via a replacement of active high pressure, high flow cooling systems with gravity driven systems.

Possible upsets to adequate performance of the passive phenomena relied upon in the design could be important contributors to risk and should be incorporated into the PRA, if it is to be considered a complete calculation of the risk and used for risk-informed applications or in Reactor Oversight Program (ROP) evaluations. For example, if an inspection should find many times the allowed inventory of fibrous material inside containment, the PRA must be able to show the potential impact of that finding, if it is to be useful in the ROP. (The DCD PRA acknowledges that core damage frequency would increase by a factor of 6,000 if failures of containment recirculation and in-containment refueling water storage tank screens occur, but uses only a "conservative" screen failure rate, rather than a model that would account for debris.) Another example would be the discovery of deposits, grease, or unauthorized paint on the exterior of the containment vessel; again, the DCD PRA is not structured to account for such departures from the assumptions of the passive design.

At this time, it is not as important that such possibilities be fully amenable to engineering analysis as it is to include the possible failure modes and uncertainties in the PRA. For example, they could be addressed using an expert elicitation of the likelihood of failure in the presence of the best available experimental, theoretical, and analytical information.

Incorporation of DCD Changes

The SNC COLA review was conducted in parallel with the review of the AP1000 DCA application. As a consequence, the SNC COLA references Revision 17 of the DCD, whereas the current version is Revision 18, and there may be a further revision prior to certification rulemaking. The staff has described the licensing steps needed to complete the COLA Final Safety Evaluation Report. These include a revision to the COLA following the final DCD revision prior to rulemaking. As described, the process does not provide for further ACRS review of either the DCD or COLA revisions that incorporate changes in design and commitments made by applicants during our review. The staff should review with us the changes and commitments which deviate significantly from those presented during our review.

In summary, we agree with the staff's resolution of all of the open items for the SNC COLA for VEGP, Units 3 and 4, with respect to the specific safety issues. We conclude that there is reasonable assurance that VEGP, Units 3 and 4, can be built and operated without undue risk to the health and safety of the public. The SNC COLA for VEGP, Units 3 and 4, should be approved following its final revision.

Dr. Said Abdel-Khalik did not participate in the Committee's deliberations regarding this matter.

Sincerely,

/RA/

J. S. Armijo
Vice-Chairman

REFERENCES

1. Letter to U.S. Nuclear Regulatory Commission, "Southern Nuclear Operating Company Application for Combined License for Vogtle Electric Generating Plant, Units 3 and 4," March 28, 2008 (ML081050133)

2. During the course of ACRS review, the staff provided the following ASER chapters:

Chapter	Chapter Title	Transmittal Memo to ACRS (Accession Numbers)	ASER (Accession Numbers)
1	Introduction and Interfaces	ML103100006	ML092810005
2	Sites Characteristics	ML100950499	ML100320032
3	Design of Structures, Components, Equipment, and Systems	ML100950532	ML093210002
4	Reactor	ML100331243	ML092610415
5	Reactor Coolant System and Connected Systems	ML100480787	ML092610460
6	Engineered Safety Features	ML100910118	ML100920459
7	Instrumentation and Controls	ML100360236	ML093230696
8	Electric Power	ML100880411	ML092870782
9	Auxiliary Systems	ML100910147	ML093560006
10	Steam and Power Conversion Systems	ML100540758	ML092720790
11	Radioactive Waste Management	ML100340674	ML092610470
12	Radiation Protection	ML100890389	ML092650039
13	Conduct of Operations	ML100910470	ML100820408
14	Initial Test Programs	ML100880449	ML092650048
15	Accident Analysis	ML100900320	ML100130006
16	Technical Specifications	ML100900407	ML092650055
17	Quality Assurance	ML100890413	ML092650063
18	Human Factors Engineering	ML100910031	ML093000107
19	Probabilistic Risk Assessment	ML100920066	ML092650121
19 Appendix 19.A	Loss of Large Areas of the Plant due to Explosions or Fires (LOLA)	ML103090198	ML103260024 (Public Version), ML101810029 (Non-Public Version)
Appendix A	License Conditions, ITAAC, and FSAR Commitments	ML103100006	ML103330312

Letter to The Honorable Gregory B. Jaczko, Chairman, from Said Abdel-Khalik, ACRS Chairman, dated January 24, 2011

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE SOUTHERN NUCLEAR OPERATING COMPANY COMBINED LICENSE APPLICATION FOR VOGTLE ELECTRIC GENERATING PLANT, UNITS 3 AND 4

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March 7, 2011

The Honorable Greg Jaczko
Chairman
Nuclear Regulatory Commission
11555 Rockville Pike
Rockville, MD 20852

Dear Chairman Jaczko:

I write to urge the Commission not to finalize its pending approval of the AP1000 reactor design until serious safety concerns about its shield building have been addressed. These concerns include those raised by one of the Commission's most long-serving staff that there is a risk that an earthquake at, or aircraft impact on, the AP 1000 could result in a catastrophic core meltdown. The danger of terrorist attacks on nuclear power plants, and the importance of their structural resilience, was made very clear on February 24, 2011. A man was arrested in Texas for allegedly planning to blow up nuclear plants using explosive chemicals he purchased online.

The Commission has recently voted to approve the design of the AP 1000. As a result, the NRC's proposed rule for the AP1000 Design Certification Amendment was published in the Federal Register on February 24, 2011. The proposed rule is set to be finalized in the next few months, following a public comment period that ends May 10, 2011 and a 30 day review of public comments. However, the Commission has taken this step toward final approval despite serious safety concerns about the Westinghouse design for the reactor shield building that have been raised by Dr. John Ma. Dr. Ma has been with the NRC since it was created by Congress in 1974. He was the Commission's lead structural reviewer charged with evaluating the design of the reactor shield to determine whether it met NRC safety standards. Dr. Ma has identified potential loopholes, which, if left open, allow designs for unsafe reactors to go forward despite the risk that an earthquake or aircraft impact could result in a catastrophic core meltdown.

While I appreciate the substantive assistance and time spent by your staff in addressing my staff's questions related to the AP 1000 review process, I remain concerned about the safety of the reactor design. I therefore request that the Commission definitively resolve these potential loopholes prior to the finalization of the NRC licensing process.

As you know, the shield building for the AP1000 serves the critical safety function of preventing catastrophic damage to the reactor that could cause fuel melting and radiation releases. The shield building physically protects the highly radioactive core of the nuclear reactor (as well as critical operating equipment) against earthquakes, storms, and airplane strikes. The shield building is intended to ensure safe shutdown following such impacts. As it is designed, the

AP1000 shield building supports a water storage unit on top of it. This water is part of the vital cooling system for the reactor, which is necessary to prevent the sort of overheating that led to core melt at the Three Mile Island reactor in Pennsylvania in 1979.

NRC regulations are intended to ensure that any new reactor design will be able to withstand the dangers of earthquakes, storms, or commercial airplane strikes. The consequences of failure could be severe: According to the report of the 9/11 Commission, Al-Qaeda considered attacking a nuclear power plant as part of its September 11th plot. The Energy Policy Act of 2005 thus included my language that required the NRC to consider the "events of September 11, 2001" and the potential for "suicide attacks" and "air-based threats" in making rules for how reactors will be able to withstand a variety of scenarios related to terrorist attacks. I have long agreed with your 2006 statement that "We should be requiring they design these plants to withstand such attacks.[1]"

On June 12, 2009, NRC issued a rule, 10 CFR 50.150, requiring applicants for new reactors to include an assessment of the ability of the reactor design to withstand the impact of a large, commercial aircraft. The NRC issued its aircraft impact rule after having already issued a final rule certifying the design of the AP1000 on January 27, 2006.[2] In anticipation of the rule change on aircraft impact, Westinghouse amended its design to address aircraft impact, by submitting Revision 16 of its AP1000 design to NRC on May 26, 2007. The NRC is currently considering Revision 18 of the AP1000 design, submitted December 1, 2010[3].

When reviewing the design for the shield building, Dr. John Ma grew concerned that the structure was too brittle and could fail if struck by a natural or manmade catastrophe. He was so concerned by this and other issues that he filed a "Non-Concurrence" statement of dissent[4] on November 4, 2010. Despite the Non-Concurrence, NRC staff issued a positive Advanced Final Safety Evaluation Report (AFSER) on December 28, 2010. The Non-Concurrence accompanied the AFSER throughout a series of approval stages, allowing you and other reviewers to know that these concerns have been raised.

If the NRC approves the AP1000, then it may have widespread use throughout the United States, making questions about its safety of crucial national importance. Among the applications for the construction of 28 new reactors being considered by NRC, the AP1000 would be the design for 7 Combined License applications covering 14 reactors, to be built in Alabama, Florida, North Carolina, South Carolina, and Georgia.[5] The Department of Energy has approved

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[1] http://www.nytimes.com/2006/11/09/us/09nuke.html

[2] http://www.nrc.gov/reactors/new-reactors/design-cert/ap1000.html

[3] The current revision is a Design Certification Amendment application that would revise the AP1000 Design Control Document, which is the overall design that NRC certified in 2006.

[4] The Non-Concurrence (NRC Form 757), the response to it by other Division of Engineering staff, and Dr. Ma's rebuttal to this response are all internal NRC documents, Accession Number ML103370648 within the Agencywide Documents Access and Management System (http://www.nrc.gov/reading-rm/adams/web-based.html). The Non-Concurrence Package was published on December 3, 2010.

[5] http://www.nrc.gov/reactors/new-reactors/col.html. The proposed sites include Jackson County, Alabama (Tennessee Valley Authority's Bellefonte site); Levy County, Florida (Progress Energy Florida, Inc.'s site); Homestead, Florida (Florida Power and Light Co.'s Turkey Point site); Wake County, North Carolina (Progress Energy Carolinas, Inc.'s Harris site); Cherokee County, South Carolina (Duke Energy's William States Lee III site);

an application for a loan guarantee of $8.3 billion to Georgia Southern for two proposed AP1000 reactors, conditional on NRC approving the AP1000. Taxpayer dollars should not be spent on unsafe reactors. The Non-Concurrence identifies several potential loopholes. I am asking the Commission to reconsider its approval of the AP1000, in light of these loopholes, the most serious of which I summarize below:

1. The AP 1000 shield building failed tests because it is brittle, and could shatter "like a glass cup"

If a reactor shield is too brittle, it may fail in an earthquake or if struck by an airplane or an automobile or other missile carried by a storm. In fact, Dr. Ma warned that if the AP1000 shield was struck, it could shatter like a "glass cup." The reason for Dr. Ma's statement is that the AP1000 shield building failed, or failed to complete, physical tests designed to evaluate whether the structure has adequate toughness for these sorts of impacts.

In its new design in response to the aircraft impact rule, Westinghouse changed the composition of the shield building from reinforced concrete to a combination of steel and concrete. This "steel-concrete module" is a first-of-its-kind design for nuclear power plants. About 60 percent of the shield building would consist of a module design (module #2) that "failed miserably" in a direct physical test of its toughness. According to the NRC Design Certification Application Review of the AP1000, "test results for out-of-plane shear showed that the modules with **[redacted]** failed in a brittle manner.[6]" A second physical test, of in-plane shear, could not be completed "due to laboratory safety constraints." These shear tests are intended to determine whether the structure will be brittle or "ductile." Ductility enables an object to deform and stretch under force, rather than breaking. Both in-plane and out-of-plane shear would act on the shield building during an earthquake. As you note in comments accompanying your "Yes" vote on the AP1000, the module that would be used for 60 percent of the shield building "was unable to satisfy the experimental protocol developed by Westinghouse and agreed to by the [NRC] staff."

The potential loophole here is that the Commission has apparently accepted Westinghouse's argument that the brittle module design would only be used in regions of the building that are unlikely to encounter high loads. Thus the failing tests were ignored. Instead of relying on the results from the test intended to prove the shield building's design, Westinghouse substituted results from computer simulations that may be a poor approximation of reality.

In his Non-Concurrence, Dr. Ma asks, "How could the [NRC] staff justify using a lower standard, by accepting a brittle structural module for about **[redacted]** of the [steel-concrete] wall for AP1000 shield building, which has more safety functions and greater consequence if the wall collapses, than other types of [reinforced concrete] shield buildings that are required to design to a higher standard of ACI [American Concrete Institute] Code?" Dr. Ma also points to NRC codes stating that the standard to which a design is held must be "commensurate with the

Fairfield County, South Carolina (South Carolina Electric & Gas' Virgil C. Summer Nuclear Station site); and Burke County, Georgia (Southern Nuclear Operating Co.'s Vogtle site).

[6] Design Certification Application Review – AP1000 Amendment. Chapter 3, page 155. http://www.nrc.gov/reactors/new-reactors/design-cert/amended-ap1000.html

importance of the safety function to be performed".[7] The AP1000 design should not be approved when the material making up 60 percent of the shield building, an essential structural component that is meant to withstand earthquakes, storms, and airplane strikes, has failed a critical physical test showing it to be brittle.

Additionally, the AP1000 shield building design has evidently failed to meet the standards of the American Concrete Institute, despite these being endorsed by NRC[8]. Westinghouse has not complied with the American Concrete Institute (ACI) "Code Requirements for Nuclear Safety-Related Concrete Structures" (ACI-349). . The design fails to meet the Code, because ACI-349 requires the structure to be ductile, would require different spacing between the steel tie-bars, and would not allow substitution of computer models in place of physical tests. Dr. Ma notes that the Safety Evaluation Report "has not provided justifications as to why its acceptance standard, which is lower than that of the ACI Code, is adequate".

To ensure the safety of the AP1000, and any future reactor designs involving steel-concrete composites, I urge you to develop a standard for this novel type of design that would apply both to the AP 1000 and other reactor designs that might seek to use it in the future. The NRC Advisory Committee on Reactor Safeguards notes that "the effort and scope of analysis and assessment required for the shield building in this case suggests that if SC [steel-concrete] composites are to be more widely used in nuclear applications, a consensus code should be developed, as has been done for other types of nuclear construction." You echoed this concern in comments accompanying your "Yes" vote for the AP1000, noting "the lack of a directly acceptable design and construction consensus standard." You write that "it would be advantageous to have such a detailed standard developed independent of any specific design approval. Therefore, I also encourage the [NRC] staff to aid in any effort ... to develop a standard." However, developing such a standard after approving the AP1000 is like planning to comply with building codes to prevent fires after the building has burned down. I ask the Commission to reverse its approval of the AP1000 until such a standard is developed, and then apply this standard to the AP1000 before reconsidering the design.

2. Weak computer simulations were used to "prove" the reactor shield is "strong enough"

Westinghouse's assertion that the brittle module is "strong enough" is based on questionable computer simulations in place of the physical tests that it should have done. The computer analysis that Westinghouse did was flawed, because it used off-the-shelf, commercially available codes to evaluate a first-of-its-kind design that could not be expected to be accurately modeled in this manner. The shield building's steel-concrete structure is novel and complex, as is the overall design of the reactor. Given the novelty and complexity of the design, Westinghouse should have developed custom code.

Additionally, Westinghouse relied on a technique known as a static "push-over" simulation. A push-over simulation imagines that an earthquake functions like a finger slowly

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[7] Codes and standards: 10 CFR 50.55a(a)(1). http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0055a.html

[8] Regulatory Guide 1.142 - Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments). http://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/rg/01-142/

pushing a cup until it falls over. Dr. Ma notes that such an analysis is not appropriate, because the shield building would experience several types of forces simultaneously during an earthquake, rather than just one simple "push." In a Technical Evaluation of Westinghouse's modeling work, scientists at Brookhaven National Laboratory agreed, stating that Westinghouse's "models may be inappropriate for static analyses intended to represent cyclic dynamic loading (i.e. earthquake); the effect of load cycling on the effective stress-strain relationship apparently is not considered [**redacted**].[9]" Westinghouse does not appear to have considered the back-and-forth forces ("cyclic dynamic loading") that occur during an actual earthquake. Instead Westinghouse appears to have fantasized that an earthquake acts like a constant force in one direction. Had Westinghouse included dynamic cyclic loading, the effective "stress-strain" curve would have had a "backbone" shape; instead, it appeared to be a monotonic curve which is consistent with Westinghouse leaving out the dynamic cyclic loading that occurs in an earthquake. The "static push-over" analysis that Westinghouse did may therefore have been inappropriate because it failed to accounts for the real back-and-forth forces in an earthquake.

Unfortunately, the Technical Evaluation document that details the software's limitations consists mostly of text redacted by NRC staff on Westinghouse's request, but the text that remains is overwhelmingly negative about Westinghouse's simulations. In addition to concerns about how Westinghouse modeled the effects of an earthquake, Westinghouse's results were presented sloppily: There is "no confidence that an appropriate level of quality assurance was implemented in the conduct of the [redacted] analyses." There were "numerous confusing, misleading, or erroneous statements." The concerns raised in this May 30, 2010 Technical Evaluation do not appear to have been addressed by Westinghouse or NRC.

I urge you to require Westinghouse, and other reactor license applicants, to complete and pass physical tests of all materials used in the design, rather than using computer models to substitute for tests that their materials have failed. There should be clear regulations indicating any exceptions where computer analyses are appropriate – and these regulations should require the use of code that is suitable to the design of the particular reactor under consideration. Where computer models are necessary, the NRC should set standards defining the quality of the models that applicants are required to use, and should conduct independent validations of those models and of the original code. Original code and data should be made available for public review, while accounting for real proprietary and security concerns. As it stands, Westinghouse may be relying on defective models that provide no meaningful assurance of whether the reactor is safe.

3. **Earthquake Forces May Have Been Underestimated by Westinghouse**

Westinghouse exploited an apparent loophole in how NRC defines earthquake forces. Westinghouse underestimated the earthquake forces that the reactor would be subjected to through use of a "seismic wave incoherency model to effectively reduce… ground motion"

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[9] R. Morante, M. Miranda, J. Nie. Technical Evaluation: AP1000 Shield Building Design Report, Revision 2. Dated 5/30/2010. Submitted as part of Dr. Ma's rebuttal to the staff response to the Non-Concurrence statement. Accession Number ML103370648 within the Agencywide Documents Access and Management System (http://www.nrc.gov/reading-rm/adams/web-based.html).

during an earthquake.[10] It is a "manifestation of mathematical concept that has not been verified and validated by experiments," according to a letter sent by Dr. Ma to your office and mine on November 8, 2010. Indeed, the "interim staff guidance" on incoherency appears to be based on a solitary report of the Electric Power Research Institute, rather than consensus in the peer-reviewed scientific literature. In his letter to my office and to you, Dr. Ma wrote that even assuming these reduced earthquake forces are correct, "the design margin in the shield wall is practically non-existent; the design will be grossly inadequate if the 'correct' and actual earthquake analyses were used." I ask that the Commission require that estimates of seismic forces be drawn from consensus, peer-reviewed scientific literature. Please ensure that Westinghouse re-does its analyses to demonstrate that the AP1000 can withstand real earthquake forces, without minimizing these forces using ill-founded assumptions.

I would note that, generally speaking, the NRC staff responses to the Non-Concurrence statements do not dispute the concerns raised by Dr. Ma. Instead, they appear to have acknowledged the flaws associated with Westinghouse's analysis, agreed that addressing the non-concurring staff member's concerns would improve the design, and then shrugged their collective shoulders and chose to abdicate responsibility to further investigate these matters prior to providing a positive Safety Evaluation Report on the shield building of the AP1000 reactor.

In fact, in your January 31 vote to approve the AP 1000 design, you acknowledge that "While it is clear that the use of a ductile material in all areas of the shield building would provide an additional enhancement to safety, I am not convinced that such a design requirement exists..." This is a far cry from a ringing endorsement: you could have said that you are convinced that the design is safe, but you do not go this far. All you say is that there is nothing requiring you to disapprove the design.

There appear to be many unresolved concerns about the AP1000 shield building design, concerns that may justify reversing your vote of approval. Consequently, I ask for your prompt assistance in responding to the following questions.

1. Why did you not require improvements to the AP1000 design to enable it to pass direct physical tests of ductility? Have past reactor shield designs approved by the NRC succeeded in meeting ductility tests that the AP1000 has failed (out-of-plane shear) or has not even completed (in-plane shear)? If so, why is a weaker standard being allowed for the AP1000, which is supposed to be even tougher than past reactor shield designs to meet the aircraft impact rule?

2. There are uncertainties associated with the modeling codes used by the applicant to analyze the accident responses of the highly complex shield building design. Given these uncertainties, are you able to provide me a guarantee that use of brittle modules for about 60 percent of the AP1000 shield building design will not significantly degrade the capability of the wall to resist being hit by a missile propelled by a storm or by an airplane, relative to a design that does not use a brittle module? If so, on what basis, and if not, then why did the Commission vote to approve the design?

[10] Design Certification Application Review - AP1000 Amendment. Chapter 3, page 58. http://www.nrc.gov/reactors/new-reactors/design-cert/amended-ap1000.html

3. There are uncertainties associated with Westinghouse's use of generic computer modeling codes and sloppily presented analyses, the "seismic wave incoherency model," and the static "push-over" analyses of the accident responses of the highly complex shield building design. Given these uncertainties, are you able to provide to me a guarantee that use of brittle modules for the majority of the AP1000 shield building design will not significantly degrade the capability of the shield building to resist an earthquake, relative to a design that does not rely on a brittle module? If so, please explain the basis for such a conclusion. If not, then why did the Commission vote to approve the design?

4. Are you certain that the brittle module is strong enough to withstand the combined stress (in-plane shear, out-of-plane shear, axial force) during a "safe shutdown earthquake"? If so, on what basis did you reach this conclusion? If not, then why did the Commission vote to approve the design?

5. What is the magnitude of earthquake for which the AP1000 would be able to maintain its ability to safely shut down the reactor? Will the NRC require that the AP1000 be able to withstand earthquakes of the magnitudes experienced in all regions of the US, or otherwise limit their deployment to areas in which earthquakes beyond the threshold, "design-basis" magnitude have never been experienced? Why or why not?

6. The shield building design includes two types of steel-concrete modules. Module #2, which failed, has wider spacing of the steel ties that go through the concrete. Module #1 has narrower spacing, which makes it tougher and enabled it to pass the out-of-plane shear test. Instead of accepting Westinghouse's flawed simulations, will the Commission reverse its approval of the AP1000 and instruct Westinghouse to simply replace the brittle module # 2 with a tougher module, such as module #1? If not, why not?

7. Given that there are applications for 14 new reactors using the AP1000 design, will NRC develop a consensus design code for this type of reactor, as has been done for other types of nuclear construction? If yes, will you reverse your approval of the AP1000 design until this code is developed and applied to the AP1000? If not, why not?

8. There are many pages in the Non-Concurrence that have been entirely redacted. For each substantive redaction, please provide me with the legal basis used to justify the redaction in question. If no appropriate basis exists, please ensure that an un-redacted version of the page in question appears in the docket for the AP1000 rule. I also ask that the Non-Concurrence package itself be placed in the docket, since it does not appear to be included among the documents that support the AP1000 rule.[11] The public should be made aware of the existence of the Non-Concurrence when commenting on the proposed design approval.

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[11] The AP1000 documents are available through the Federal e-Rulemaking website at http://www.regulations.gov by searching under Docket ID NRC-2010-0131.

Thank you for your attention to this important matter. Please provide your response no later than March 28. If you have any questions, please have your staff contact Dr. Ilya Fischhoff or Dr. Michal Freedhoff of my staff at 202-225-2836.

Sincerely,

Edward J. Markey

Edward J. Markey

Westinghouse

Westinghouse Electric Company
Nuclear Power Plants
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U S Nuclear Regulatory Commission
Two White Flint North
11555 Rockville Pike
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Direct tel: 412-374-2035
Direct fax: 724-940-8505
e-mail: ziesinrf@westinghouse.com

Your ref: Docket No. 52-006
Our ref: DCP_NRC_003113

February 23, 2011

Subject: AP1000 Containment Cleanliness – DCD Markup for Rev. 19

Westinghouse is submitting a response to the U.S. Nuclear Regulatory Commission (NRC) regarding the Advisory Committee on Reactor Safeguards (ACRS) letter to the Chairman, U.S. NRC, dated December 20, 2010. This letter is submitted in support of the AP1000 Design Certification Amendment Application (Docket No. 52-006). The information included in these responses is generic and is expected to apply to all Combined Operating License (COL) applicants referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application.

The recent ACRS letter, dated December 20, 2010, on acceptability of long term core cooling performance for AP1000 concluded that any future relaxation of cleanliness requirements will require substantial data and analysis and requested a licensing path be identified for future regulatory approval of any increase in the containment debris limits.

Westinghouse has reviewed the potential application of the specific cleanliness limits as information to be contained in the AP1000 Technical Specifications (TS). Westinghouse does not believe the debris limits meet the appropriate level of safety impact to be included in the TS. Our interpretation is based on review of 10 CFR 50.36, and is consistent with the operating plants TS, which have detailed evaluations of long term cooling (including debris limits) and do not include the specific debris limits in their TS.

The containment debris limits identified for the operating fleet vary significantly with regard to the ease of compliance, i.e., some limits are closer to the expected debris findings; however, it is not the difficulty of compliance that determines the need to include the debris limits in the TS. These limits do not meet the criteria for including limiting conditions for operations as provided in 10 CFR 50.36.

10 CFR 50.36 criterion (c)(2)(ii)(B) is as follows:

- *Criterion 2.* A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

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NRO

The impact of not meeting this operating restriction has not previously been considered to meet this criterion, as evidenced by it not being included in any of the NRC approved Standard Technical Specifications NUREGs.

Therefore, Westinghouse is proposing to make a change to the Design Control Document (DCD), Revision 18 to identify the total amount of resident debris and fiber limits as Tier 2* information.

Tier 2* is defined in the regulations (10 CFR Part 52 Appendix D) as: "*Tier 2** means the portion of the Tier 2 information, designated as such in the generic DCD, which is subject to the change process in Section VIII.B.6 of this appendix."

Westinghouse believes that the inclusion of the containment debris limits as Tier 2* information, as well select general screen design criteria (See Enclosure 1 for specific DCD markups), meets the intent of the ACRS for receiving prior NRC approval prior to any departure from these limits.

Questions or requests for additional information related to the content and preparation of this response should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,

R. F. Ziesing
Director, U.S. Licensing

/Enclosure

1. DCD Markups for Rev. 19 – Intro Table 1-1, Tier 2 Section 6.3.2.2.7.1, and Tier 2 Section 6.3.8.1

cc:	D. Jaffe	-	U.S. NRC	1E
	E. McKenna	-	U.S. NRC	1E
	P. Buckberg	-	U.S. NRC	1E
	T. Spink	-	TVA	1E
	P. Hastings	-	Duke Energy	1E
	R. Kitchen	-	Progress Energy	1E
	A. Monroe	-	SCANA	1E
	P. Jacobs	-	Florida Power & Light	1E
	C. Pierce	-	Southern Company	1E
	E. Schmiech	-	Westinghouse	1E
	G. Zinke	-	NuStart/Entergy	1E
	R. Grumbir	-	NuStart	1E
	M. Melton	-	Westinghouse	1E
	D. Lindgren	-	Westinghouse	1E
	B. Carpenter	-	Westinghouse	1E

Enclosure 1

DCD Markups for Rev. 19 – Intro Table 1-1, Tier 2 Section 6.3.2.2.7.1, and Tier 2 Section 6.3.8.1

Table 1-1 (Cont.)
Index of AP1000 Tier 2 Information Requiring NRC Approval for Change

Item	Expiration at First Full Power	Tier 2 Reference
Maximum Fuel Rod Average Burnup	No	4.3.1.1.1
Reactor Core Description (First Cycle)	Yes	Table 4.3-1
Nuclear Design Parameters (First Cycle)	Yes	Table 4.3-2
Reactivity Requirements for Rod Cluster Control Assemblies	Yes	Table 4.3-3
ASME Code Piping Design Restrictions	Yes	5.2.1.1
Reactor Coolant Pump Design	No	5.4.1.2.1
MOV Design and Qualification	Yes	5.4.8.1.2
Other Power-Operated Valves Design and Qualification	Yes	5.4.8.1.3
Motor Operated Valves	Yes	5.4.8.5.2
Power Operated Valves	Yes	5.4.8.5.3
ASME Code Cases	Yes	Table 5.2-3 Table 3.9-9 Table 3.9-10 5.2.1.2
General Screen Design Criteria	No	6.3.2.2.7.1
WCAP-17201-P, "AC160 High Speed Link Communication Compliance to DI&C-ISG-04 Staff Position 9, 12, 13, and 15," Rev 0, February 2010	Yes	Table 1.6-1 7.1.7
WCAP-15927 (Non-Proprietary), "Design Process for AP1000 Common Q Safety Systems," Rev 2	Yes	Table 1.6-1 7.1.2.14.1 7.1.7
WCAP-17179, "AP1000 Component Interface Module Technical Report"	Yes	Table 1.6-1 7.1
WCAP-16097-P-A, "Common Qualified Platform," Rev 0	Yes	Table 1.6-1 7.1
WCAP-16096-NP-A, "Software Program Manual for Common Q Systems," Rev 01A	Yes	Table 1.6-1 7.1
Verification and Validation	Yes	7.1.2.14
Hard-wired DAS manual actuation	No	7.7.1.11
Nuclear Island Fire Areas	No	Figure 9A-1
Turbine Building Fire Areas	No	Figure 9A-2
Annex I & II Building Fire Areas	No	Figure 9A-3
Radwaste Building Fire Areas	No	Figure 9A-4

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Comment [rmk16]: 28

Deleted: WCAP-14605, "Westinghouse Setpoint Methodology for Protection Systems, AP600," Rev 0

Comment [tlw17]: 17

3. *[Metal reflective insulation is used on ASME class 1 lines because they are subject to loss-of-coolant accidents. Metal reflective insulation is also used on the reactor vessel, the reactor coolant pumps, the steam generators, and on the pressurizer because they have relatively large insulation surface areas and they are located close to large ASME class 1 lines. As a result, they are subject to jet impingement during loss-of-coolant accidents.]** A suitable equivalent insulation to metal reflective may be used. A suitable equivalent insulation is one that is encapsulated in stainless steel that is seam welded so that LOCA jet impingement does not damage the insulation and generate debris. Another suitable insulation is one that may be damaged by LOCA jet impingement as long as the resulting insulation debris is not transported to the containment recirculation screens, to the IRWST screens, or into a direct vessel injection or a cold leg LOCA break that becomes submerged during recirculation. In order to qualify as a suitable equivalent insulation, testing must be performed that subjects the insulation to conditions that bound the AP1000 conditions and demonstrates that debris would not be generated. If debris is generated, testing and/or analysis must be performed to demonstrate that the debris is not transported to an AP1000 screen or into the core through a flooded break. It would also have to be shown that the material used would not generate chemical debris. In addition, the testing and/or analysis must be approved by the NRC.

   *[In order to provide additional margin, metal reflective insulation is used inside containment where it would be subject to jet impingement during loss-of-coolant accidents that are not otherwise shielded from the blowdown jet.]** As a result, fibrous debris is not generated by loss-of-coolant accidents. Insulation located within the zone of influence (ZOI), which is a spherical region within a distance equal to 29 inside diameters (for Min-K, Koolphen-K, or rigid cellular glass insulation) or 20 inside diameters (for other types of insulation) of the LOCA pipe break is assumed to be affected by the LOCA when there are intervening components, supports, structures, or other objects.

   *[The ZOI in the absence of intervening components, supports, structures, or other objects includes insulation in a cylindrical area extending out a distance equal to 45 inside diameters from the break along an axis that is a continuation of the pipe axis and up to 5 inside diameters in the radial direction from the axis.]** A suitable equivalent insulation to metal reflective may be used as discussed in the previous paragraph.

   *[Insulation used inside the containment, outside the ZOI, but below the maximum post-DBA LOCA floodup water level (plant elevation 110.2 feet), is metal reflective insulation, jacketed fiberglass, or a suitable equivalent.]** A suitable equivalent insulation is one that would be restrained so that it would not be transported by the flow velocities present during recirculation and would not add to the chemical precipitates. In order to qualify as a suitable equivalent insulation, testing must be performed that subjects the insulation to conditions that bound the AP1000 conditions and demonstrates that debris would not be generated. If debris is generated, testing and/or analysis must be performed to demonstrate that the debris is not transported to an AP1000 screen or into the core through a flooded break. It would also have to be shown that the material used would not generate chemical debris. In addition, the testing and/or analysis must be approved by the NRC.

   [Insulation used inside the containment, outside the ZOI, but above the maximum post-design basis accident (DBA) LOCA floodup water level, is jacketed fiberglass, rigid cellular

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

glass, or a suitable equivalent]* A suitable equivalent insulation is one that when subjected to dripping of water from the containment dome would not add to the chemical precipitates; suitable equivalents include metal reflective insulation.

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4. Coatings are not used on surfaces located close to the containment recirculation screens. The surfaces considered close to the screens are defined in subsection 6.3.2.2.7.3. Refer to subsection 6.1.2.1.6. These surfaces are constructed of materials that do not require coatings.

5. The IRWST is enclosed which limits debris egress to the IRWST screens.

6. Containment recirculation screens are located above lowest levels of containment.

7. Long settling times are provided before initiation of containment recirculation.

8. Air ingestion by safety-related pumps is not an issue in the AP1000 because there are no safety-related pumps. The normal residual heat removal system pumps are evaluated to show that they can operate with minimum water levels in the IRWST and in the containment.

9. A commitment for cleanliness program to limit debris in containment is provided in subsection 6.3.8.1.

10. [*Other potential sources of fibrous material, such as ventilation filters or fiber-producing fire barriers, are not located in jet impingement damage zones or below the maximum post-DBA LOCA floodup water level*]*

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11. Other potential sources of transportable material, such as caulking, signs, and equipment tags installed inside the containment are located:

    - Below the maximum flood level, or
    - Above the maximum flood level and not inside a cabinet or enclosure.

    Tags and signs in these locations are made of stainless steel or another metal that has a density $\geq$100 lbm/ft^3. Caulking in these locations is a high density ($\geq$100 lbm/ft^3).

    The use of high-density metal prevents the production of debris that could be transported to the containment recirculation screens, to the IRWST screens, or into a direct vessel injection or a cold leg LOCA break location that is submerged during recirculation. If a high-density material is not used for these components, then the components must be located inside a cabinet or other enclosure, or otherwise shown not to transport; the enclosures do not have to be watertight, but need to prevent water dripping on them from creating a flow path that would transport the debris outside the enclosure. For light-weight (< 100 lb_m/ft^3) caulking, signs or tags that are located outside enclosures, testing must be performed that subjects the caulking, signs, or tags to conditions that bound the AP1000 conditions and demonstrates that debris would not be transported to an AP1000 screen or into the core through a flooded break. Note that in determining if there is sufficient water flow to transport these materials, consideration needs to be given as to whether they are within the ZOI (for the material used) because that determines whether they are in their original geometry or have been reduced to

*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

smaller pieces. It would also have to be shown that the material used would not generate chemical debris. In addition, the testing must be approved by the NRC.

12. An evaluation consistent with Regulatory Guide 1.82, Revision 3, and subsequently approved NRC guidance, has been performed (Reference 3) to demonstrate that adequate long-term core cooling is available considering debris resulting from a LOCA together with debris that exists before a LOCA. As discussed in subsection 6.3.2.2.7.1, a LOCA in the AP1000 does not generate fibrous debris due to damage to insulation or other materials included in the AP1000 design. The evaluation considered resident fibers and particles that could be present considering the plant design, location, and containment cleanliness program. The determination of the characteristics of such resident debris was based on sample measurements from operating plants. The evaluation also considered the potential for the generation of chemical debris (precipitants). The potential to generate such debris was determined considering the materials used inside the AP1000 containment, the post-accident water chemistry of the AP1000, and the applicable research/testing.

    The evaluation considered the following conservative considerations:

    - [*The COL cleanliness program will limit the total amount of resident debris inside the containment to ≤130 pounds and the amount of the total that might be fiber to ≤6.6 pounds*.]*

    - In addition to the resident debris, the LOCA blowdown jet may impinge on coatings and generate coating debris fines, which because of their small size, might not settle. The amount of coating debris fines that can be generated in the AP1000 by a LOCA jet will be limited to less than 70 pounds for double-ended cold leg and double-ended direct vessel injection LOCAs. In evaluating this limit, a ZOI of 4 IDs for epoxy and 10 IDs for inorganic zinc will be used. A DEHL LOCA could generate more coating debris; however, with the small amount of fiber available in the AP1000 following a LOCA, the additional coating debris fines that may be generated in a DEHL LOCA are not limiting.

    - The total resident and ZOI coating debris available for transport following a LOCA is ≤ 193.4 pounds of particulate and ≤ 6.6 pounds of fiber. The percentage of this debris that could be transported to the screens or to the core is as follows:

        – Containment recirculation screens is ≤100 percent fiber and particles

        – IRWST screens is ≤50 percent fiber and 100 percent particles

        – Core (via a direct vessel injection or a cold leg LOCA break that becomes submerged) is ≤90 percent fiber and 100 percent particles

    - Fibrous insulation debris is not generated and transported to the screens or into the core as discussed in item 3.

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*NRC Staff approval is required prior to implementing a change in this information; see DCD Introduction Section 3.5.

6.3.7.6.2.2 In-Containment Refueling Water Storage Tank Injection Motor-Operated Valve Controls

The motor-operated valves in each in-containment refueling water storage tank injection line are normally open during all modes of normal plant operation. Power to these valves is locked out. Redundant valve position indication and alarms are provided to alert the operator if a valve is inadvertently closed. The technical specifications specify surveillances to show that these valves are open. These valves also receive a safeguards actuation signal to confirm that they are open in the event of an accident. As a result of the power lock out, the redundant position indication and alarms and the technical specifications the valve controls are nonsafety-related.

6.3.7.6.2.3 Passive Residual Heat Removal Heat Exchanger Inlet Motor-Operated Valve Control

The motor-operated valve in the passive residual heat removal heat exchanger inlet line is normally open during normal plant operation. Power to this valve is locked out. Redundant valve position indications and alarms are provided to alert the operator if the valve is open. This valve also receives an actuation signal to confirm that it is open in the event of an accident.

6.3.7.7 Automatic Depressurization System Actuation at 24 Hours

A timer is used to automatically actuate the automatic depressurization system if offsite and onsite power are lost for about 24 hours. This prevents discharging the Class 1E dc power sources such that they are no longer able to operate the automatic depressurization system valves. If power becomes available to the dc batteries and they are no longer discharging prior to activation of the timer, then the automatic depressurization system actuation would be delayed. If the plant does not need actuation of the automatic depressurization system based on having stable pressurizer level, full core makeup tanks, and high and stable in-containment refueling water storage tank levels, the operators are directed to de-energize all loads on the 24-hour batteries. This action will block actuation of the automatic depressurization system and allow for its actuation later should the plant conditions unexpectedly degrade.

6.3.8 Combined License Information

6.3.8.1 Containment Cleanliness Program

The Combined License applicants referencing the AP1000 will address preparation of a program to limit the amount of debris that might be left in the containment following refueling and maintenance outages. The cleanliness program will limit the storage of outage materials (such as temporary scaffolding and tools) inside containment during power operation to items that do not produce debris (physical or chemical), which could be transported to the containment recirculation screens, to the IRWST screens, or into a direct vessel injection or a cold leg LOCA break that becomes submerged during recirculation. The cleanliness program shall limit the amount of latent debris and fibrous material located within the containment, as identified in subsection 6.3.2.2.7.1, item 12.

Comment [rmk7]: 28

Deleted: ~~to less than 130 pounds with less than or equal to 6.6 pounds being composed of fibrous material~~

Comment [rmk8]: 28

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SECRETARY

February 9, 2011

COMMISSION VOTING RECORD

DECISION ITEM: SECY-11-0002

TITLE: PROPOSED RULE: AP1000 DESIGN CERTIFICATION AMENDMENT (RIN 3150-AI81)

The Commission (with all Commissioners agreeing) approved the subject paper as recorded in the Staff Requirements Memorandum (SRM) of February 9, 2011.

This Record contains a summary of voting on this matter together with the individual vote sheets, views and comments of the Commission.

Annette L. Vietti-Cook
Secretary of the Commission

Attachments:
1. Voting Summary
2. Commissioner Vote Sheets

cc: Chairman Jaczko
Commissioner Svinicki
Commissioner Apostolakis
Commissioner Magwood
Commissioner Ostendorff
OGC
EDO
PDR

VOTING SUMMARY - SECY-11-0002

RECORDED VOTES

	APRVD	DISAPRVD	ABSTAIN	NOT PARTICIP	COMMENTS	DATE
CHRM. JACZKO	X				X	1/30/11
COMR. SVINICKI	X				X	2/2/11
COMR. APOSTOLAKIS	X				X	1/28/11
COMR. MAGWOOD	X				X	1/28/11
COMR. OSTENDORFF	X				X	1/27/11

COMMENT RESOLUTION

In their vote sheets, all Commissioners approved the staff's recommendation and provided some additional comments. Subsequently, the comments of the Commission were incorporated into the guidance to staff as reflected in the SRM issued on February 9, 2011.

NOTATION VOTE

RESPONSE SHEET

TO: Annette Vietti-Cook, Secretary

FROM: Chairman Gregory B. Jaczko

SUBJECT: SECY-11-0002 – PROPOSED RULE: AP1000 DESIGN CERTIFICATION AMENDMENT (RIN 3150-AI81)

Approved __X__ Disapproved _____ Abstain _____

Not Participating _____

COMMENTS: Below ___ Attached _X_ None ___

SIGNATURE

1/31/11

DATE

Entered on "STARS" Yes _x_ No ___

Chairman Jaczko's Comments on SECY-11-0002, "Proposed Rule: AP1000 Design Certification Amendment"

I continue to believe that certification of reactor designs through rulemaking is important to promoting design standardization, ensuring safety and security through rigorous independent technical and engineering reviews, promoting early resolution of technical and regulatory issues, and providing greater regulatory certainty and efficiencies to applicants seeking combined licenses. I approve the staff's recommendation to publish the proposed rule that will amend the AP1000 Design Certification Rule subject to my comments below.

The amendment to the AP1000 Design Certification Rule is a substantial improvement over the AP1000 design previously approved by the Commission. Many significant changes have been made by Westinghouse to resolve issues previously deferred to the combined license applicants referencing the AP1000 standard design, to resolve design acceptance criteria, to increase the detail of the design, to address a number of technical issues, and to address the aircraft impact issues. The review elicited a number of differing views from the staff in several non-concurrences. These differences are a visible example of how the staff exhibits the NRC's Organizational Values, in particular by their consistent commitment to our mission and their abiding respect for differing views. I applaud the staff for the professional manner in which they dealt with these issues. Most importantly, I applaud the staff for ensuring their review was focused on the protection of public health and safety in the face of persistent schedule pressures. The commitment and respect demonstrated by the staff and ACRS during this process furthers the type of open collaborative work environment that is key to our success as an agency.

There are many technical areas of importance reviewed by the staff in preparation of the proposed rule for the design certification amendment. I want to comment on the most significant continued point of disagreement among members of the staff, the ability of the shield building to meet the agency's requirements for seismic loads. This is an area of technical complexity, but the staff presented a clear explanation in the documents related to the non-concurrence. As with so many of the issues we deal with as an agency, even matters of technical complexity often come down to subjective judgments and interpretations of regulations, guidance, codes, and standards.

As I understand the issue, the disagreement rests on the necessity of the structural elements of the shield building to perform in a ductile manner. In revising the shield building design to satisfy staff concerns, Westinghouse proposed two types of modules to comprise the bulk of the shield building. Since these modules represent a new type of steel-concrete composite structure previously unused in the nuclear context in the United States, the staff required Westinghouse to confirm many of the structural properties of these modules through a series of tests. One of these modules, which would be used in approximately 60 percent of the shield building, was unable to satisfy the experimental protocol developed by Westinghouse and agreed to by the staff. In particular, this structural module failed the out of plane shear test in a brittle manner and therefore failed to exhibit ductile behavior. As I understand the issue, had the second module type satisfied the test protocols, there would be no disagreement among the staff. (This was in fact the case for the first module type used in the areas of the shield building which are expected to experience higher loads during the design basis event.)

The point of contention appears to me to be whether this is necessary to comply with the agency's regulations. The staff believes it does not because the forces that the shield building would experience in the regions where these modules would be used would be much lower than the loads that would lead to failure of the module, in other words the module is strong enough. This has been determined by Westinghouse through simulation and reviewed and approved by the staff. As a result, the *overall structure* would exhibit ductile behavior because the second module type would not be expected to suffer significant deformation. In addition, the areas of the shield building in which the energy dissipation are concentrated would involve the first module type, which did exhibit ductile failure in experimental tests. Moreover the staff believes that the most relevant code here American Concrete Institute (ACI)-349 does not *require* ductility of all elements of the structure.

The non-concurrer, however, believes this does matter, because the most relevant code ACI-349-approved by the staff as an acceptable code for demonstrating compliance with seismic and structural regulations requires each element of the structure to demonstrate ductile failure *even for loads* which exceed the expected design loads of the design basis event, namely the safe shutdown earthquake. As I understand the position of the non-concurrer, the ductility requirement is a defense in depth measure to account for the inability to predict all the possible loads on a structure, but still ensure that there is not a catastrophic collapse if actual forces during an earthquake or other event are different than the forces analyzed by Westinghouse and the staff. Moreover ductility is an inherent property of the material determined by a test protocol which subjects the material to forces several times the forces necessary to deform steel. As a result, the ductility property is independent of the specific forces of any specific scenario.

Many individuals have reviewed this disagreement, including the Advisory Committee on Reactor Safeguards, and have found the approach taken by the staff acceptable. Based on the information, I have seen at this point there appears to be no one technically correct judgment in this case. Rather, the many reviewers of the shield building have different philosophical approaches to acceptable design. I applaud the non-concurrer for pursuing his view of the most appropriate manner in which to provide reasonable assurance of adequate protection.

I am not convinced at this time, however, that the design as presented does not comply with the Commission's regulations. While it is clear that the use of a ductile material in all areas of the shield building would provide an additional enhancement to safety, I am not convinced that there is a clear case that such a design requirement exists in the most relevant ACI code or any of the other codes referenced by Westinghouse and the staff and therefore would be seen as a necessary condition for approval by the staff. I suspect stakeholders will comment on this issue during the proposed rule stage and I encourage the Commission to specifically develop one or more questions to frame the issue and guide stakeholders to comment in the most productive manner for the Commission's consideration of the final rule for the design certification.

As part of their review, the staff effectively developed a standard for steel-concrete composite structures; however, I believe it would be more effective to develop such an approach apart from any specific design review. It is clear from the staff's safety evaluation that one of the challenges that they faced in reviewing the AP1000 shield building was the lack of a directly acceptable design and construction consensus standard. The lack of a directly applicable standard necessitated the reliance on portions of closely related standards produced by ACI, American Institute of Steel Construction, Japan Electric Association Code, and Federal Emergency Management Agency. If this type of construction is to be continued in the United States for facilities regulated by the NRC, it would be advantageous to have such a detailed standard developed independent of any specific design approval. Therefore, I also encourage the staff to aid in any effort by the ACI or other consensus standard organization to develop a standard that covers the proper design and construction of steel-concrete composite structure that form part of a nuclear power plant and that has nuclear safety-related functions.

As the staff evaluates comments on this proposed rule, I am confident that the staff will continue to demonstrate their commitment to public health and safety and respect for differing views by their thoughtful consideration of the public comments that may be submitted on the proposed rule and the technical changes to the AP1000 standard design, specifically, the shield building and instrumentation and controls.

1/31/11

Gregory B. Jaczko Date

NOTATION VOTE

RESPONSE SHEET

TO: Annette Vietti-Cook, Secretary

FROM: COMMISSIONER SVINICKI

SUBJECT: SECY-11-0002 – PROPOSED RULE: AP1000 DESIGN CERTIFICATION AMENDMENT (RIN 3150-AI81)

Approved __XX__ Disapproved _____ Abstain _____

Not Participating _____

COMMENTS: Below __XX__ Attached __XX__ None ___

I approve the proposed amendment to 10 CFR Part 52 for publication in the *Federal Register*, subject to the attached edits. I commend the staff's successful demonstration of the NRC's nonconcurrence process, which allows issues to be raised, evaluated, and dispositioned with finality.

SIGNATURE

02/ 2 /11

DATE

Entered on "STARS" Yes ✓ No ___

NOTATION VOTE

RESPONSE SHEET

TO: Annette Vietti-Cook, Secretary

FROM: COMMISSIONER MAGWOOD

SUBJECT: SECY-11-0002 – PROPOSED RULE: AP1000 DESIGN CERTIFICATION AMENDMENT (RIN 3150-AI81)

Approved X Disapproved ____ Abstain ____

Not Participating ____

COMMENTS: Below ___ Attached X None ___

SIGNATURE

28 January 2011
DATE

Entered on "STARS" Yes X No ___

Commissioner Magwood's Comments on SECY-11-0002

Proposed Rule: AP1000 DESIGN CERTIFICATION AMENDMENT

I approve the publication of the proposed amendment to 10 CFR Part 52 with minor editorial edits attached.

I commend the staff for their diligence and tenacity in the performance of the safety review of this amendment. As the agency faces ever emerging challenges and new responsibilities, our priority remains, as always, the adequate protection of public health and safety. This rigorous safety review is an example of the agency's resolute work ethic that perpetuates NRC's worldwide reputation as a strong, stable, predictable regulator.

It was edifying to see the NRC's Non-Concurrence Process in action. This process, which allows employees to document their concerns early in the decision-making process and have them addressed as the issue moves through the management chain, is a healthy practice and contributes to more robust end products.

1/28/11

William D. Magwood, IV   date

NOTATION VOTE

RESPONSE SHEET

TO: Annette Vietti-Cook, Secretary

FROM: COMMISSIONER OSTENDORFF

SUBJECT: SECY-11-0002 – PROPOSED RULE: AP1000 DESIGN CERTIFICATION AMENDMENT (RIN 3150-AI81)

Approved ✓ Disapproved _____ Abstain _____

Not Participating _____

COMMENTS: Below ___ Attached ✓ None ___

SIGNATURE

1/27/11

DATE

Entered on "STARS" Yes ✓ No ___

Commissioner Ostendorff's Comments on SECY 11-0002

"Proposed Rule: AP1000 Design Certification Amendment (RIN 3150-AI81)"

I approve the staff's proposed AP1000 rulemaking for publication in the *Federal Register.* The NRC staff and ACRS's review of the AP1000 design for compliance with aircraft impact requirements and evaluation of numerous updates to the original AP1000 design represents exceptional service and contribution to the NRC's safety mission. Of particular exemplary effort was the staff's identification of AP1000 shield building vulnerabilities that had existed in an earlier proposed design. In reviewing this first-of-a-kind design, the staff appropriately demonstrated a questioning, safety-focused attitude to identify and resolve critical-safety issues. These issues were handled with high technical and managerial competence. Ultimately, the applicant made significant modifications to the shield building design which the staff and ACRS independently determined to be acceptable. I also commend the staff for embodying an open collaborative work environment that allows diverse or dissenting views to be raised and appropriately assessed using the NRC's established processes. I considered this particular AP1000 review prototypical of an NRC strength to vet issues openly and foster constructive resolution. The Commission is ultimately best served when safety issues are addressed in this manner. I look forward to reviewing the final rule and the staff's evaluation of comments on the proposed rule.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, DC 20555 - 0001

December 20, 2010

The Honorable Gregory B. Jaczko
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: LONG-TERM CORE COOLING FOR THE WESTINGHOUSE AP1000 PRESSURIZED WATER REACTOR

Dear Chairman Jaczko:

During the 577th and 578th meetings of the Advisory Committee on Reactor Safeguards (ACRS), November 4-6, and December 2-4, 2010, we reviewed the NRC staff's safety evaluation of the adequacy of long-term core cooling as it applies to the AP1000 design certification amendment application. AP1000 long-term core cooling performance was also reviewed during subcommittee meetings held on November 19-20, 2009, October 5, November 17-19, and December 1, 2010. During these meetings, we had the benefit of discussions with representatives of the NRC staff and the Westinghouse Electric Company (WEC or applicant). We also had the benefit of the documents referenced.

CONCLUSION AND RECOMMENDATION

1. The regulatory requirements for long-term core cooling for design basis accidents have been adequately met, and the issue is closed for the AP1000 design.

2. This conclusion is based on the cleanliness requirements specified in the amendment. Any future proposed relaxation of these requirements will require substantial additional data and analysis.

BACKGROUND

On May 8, 2008, the Commission issued a Staff Requirements Memorandum (SRM) stating that, "the ACRS should advise the staff and Commission on the adequacy of the design basis long-term core cooling approach for each new reactor design based, as appropriate, on either its review of the design certification or the first license application referencing that reactor design." The main focus of the Commission's concern was the ability of the safety systems to provide adequate core cooling over extended time periods when the Emergency Core Cooling System (ECCS) recirculation mode is activated during a design basis accident (DBA).

The AP1000 is a pressurized light water reactor design that incorporates new passive safety features not found in current operating pressurized water reactors (PWRs). These include a Passive Containment Cooling System (PCS) to transport heat to the ultimate heat sink for accident scenarios.

Many aspects of long-term cooling (LTC), excluding the effects of debris, were considered as part of the AP1000 certification process that was completed in January 2006. This letter report addresses the effect of debris on LTC.

DISCUSSION

For AP1000 LTC, coolant is driven by gravity head through the core. The coolant exits, as a steam-water mixture, mainly through the Automatic Depressurization System (ADS-4) valves. The steam flowing out from the core removes decay heat and is condensed on the inside of the steel containment shell. The condensed water flows down the containment walls, is collected in the In-Containment Refueling Water Storage Tank (IRWST), and is recirculated. Screens placed between the IRWST and the core capture debris. Sump screens are placed in another possible flow path, which is through the loop compartment to the core.

During loss-of-coolant accidents (LOCAs) the level in the IRWST tank drops, redistributing water to the region around the reactor vessel and associated piping, causing much of the piping to be submerged. Breaks in this piping, such as in the cold legs or the direct vessel injection (DVI) lines, can be submerged and provide an unfiltered flow path to the reactor core.

The main sources of debris are: 1) latent containment debris, such as hair and clothing fibers; 2) debris generated by LOCA jets and exposure to post LOCA conditions; and 3) chemical precipitates that form in the recirculating water stream. WEC has taken advantage of what has been learned with regard to the GSI-191 issue for the fleet of operating PWRs. Efforts have been made in the design to minimize LOCA-generated debris by selecting low fiber, low particulate insulation and LOCA resistant qualified coatings. Stringent containment cleanliness requirements have been imposed in the amendment that limit fibrous latent debris and the amount of aluminum that can be submerged. Sump screens have been designed to assure negligible reduction in recirculation flows due to debris accumulation on them.

Because of these actions, any potential problems with LTC would primarily be due to flow blockage in the core which may trap materials that pass through the screens, and more importantly, materials that enter the core directly through submerged breaks. The possibility that unfiltered water, carrying in some cases a major portion of the suspended fibrous and particulate debris, will gain ingress directly to the core is unique to the AP1000 design. Furthermore, the gravity head available in the AP1000 for driving flow through a core in which debris has accumulated is limited. Both of these factors add to the difficulties in determining the adequacy of AP1000 LTC.

In the certified design, the applicant carried out a series of calculations using WCOBRA/TRAC, which had been accepted for analysis of LTC, without considering debris. Resolution of debris effects was deferred to the combined license (COL) stage but is now being addressed in the amendment. In the calculations for the design certification amendment, WCOBRA/TRAC was also used. The effect of debris, which mainly causes in-vessel head losses, was modeled by introducing a constant loss factor at the core inlet. The purpose of these calculations was to determine how the loss factor affected ADS-4 vent qualities (the mass fraction of steam), pressure loss across the debris bed, and mass flux through the core. Based on analysis of the results, the applicant proposed what is effectively an acceptance criterion that requires pressure drop through the debris bed to be less than a specified amount at a specified flow rate. When the criterion was met, the WCOBRA/TRAC results indicated that the ADS-4 vent quality would be less than 50 percent which resulted in acceptable boron concentration. At our request, additional results were obtained with higher loss factors to elucidate the margins inherent in the proposed acceptance criterion with regard to critical heat flux and boron concentrations. These indicated sufficient margin to account for uncertainties, and we agree that the acceptance criterion should be as proposed by the applicant.

To determine whether blockage under representative debris loadings and flow conditions would meet the acceptance criterion, the applicant conducted a series of tests in a pumped flow loop. The loop incorporated a part-length fuel bundle with representative inlet and spacer geometries. Flow rates were varied to simulate the transient mass flux through the core as the debris bed built up, though the lowest flow rates studied were somewhat higher than the value of the flow rate for the acceptance criterion. Fibrous and particulate debris loadings were conservative but were varied over a narrow range. An approved surrogate material was added over a period of time to simulate the effect of chemical precipitates, such as aluminum oxyhydroxide that might form. The reference experimental protocol was selected to follow the sequence of events expected for the long term recirculation phase of a DBA. However, the exact protocol that should be used is unclear, and tests have shown that variations in protocol can result in significant differences in pressure losses. For example, in a test where the protocol was inadvertently varied to follow a non-representative event sequence, a significantly larger debris-bed pressure loss was obtained than for the same case run with the reference protocol. However, the pressure loss still remained within acceptable limits.

For the tests used to determine whether the acceptance criterion could be met, the fibrous debris used was derived from NUKON insulation which may not be typical of the latent debris that might accumulate in the core in an AP1000 DBA. Two tests were conducted with non-reference protocols using debris containing hair and clothing fibers. While the pressure loss behavior was somewhat different from that observed in the NUKON-based test, the pressure losses were within the acceptable range.

Most of the tests were conducted at room temperature. In two exploratory tests, debris-bed pressure losses decreased significantly when the temperature was raised to values closer to those expected during LTC. The lower pressure losses are consistent with the effect of increasing temperatures on water viscosity. However, the net effect of increased temperature on head loss is still uncertain since organic materials may behave differently at LTC temperatures than the NUKON-based debris used in the tests. Absent additional experiments at LTC temperatures using organic fibers (hair, clothing) and prototypical water chemistry, it is not certain that the observed benefit of higher temperature will provide additional margin.

In the tests, the head losses that arose from debris accumulation in the fuel inlet region were rather low when the debris consisted of fibers and particulates alone. However, when the surrogate chemical precipitates were added gradually, head losses rose sharply initially, but generally leveled off as more was added. The effect of the chemical precipitates will depend on the rate of their formation. Although this is uncertain, the rate at which surrogates were added in the tests appears to be conservative.

Radiolysis in the containment atmosphere and doses to cable insulation might form small amounts of nitrogen oxides and hydrogen chloride which may acidify the water condensed on the containment wall. The acidified water may leach zinc from the containment coating. If some zinc does dissolve into the recirculating water stream, the chemical load that should be considered in evaluating debris head losses would be increased. While the experiments indicating that head losses level off with the addition of chemical surrogates suggest that the effect of the possible zinc load could be small, the effect has not been investigated and adds to the uncertainties.

In view of the relatively narrow range of conditions explored in the applicant's test program and the significant uncertainties in the results, a site visit was conducted to better understand the AP1000 related results in the context of in-core debris effects found in the PWR Owner's Group (PWROG) experiments. These cover a wide range of conditions and, while not directly applicable to the AP1000, offer valuable insights into the effects of various experimental parameters.

As a result of the issues arising in subcommittee meetings and the site visit, additional experimental results from the PWROG program at lower flow rates and higher fiber loadings were made available to us.

When the additional WCOBRA/TRAC analyses and the additional experimental results are taken into account, in-core debris bed pressure losses appear to meet the acceptance criterion with sufficient margin to account for the uncertainties, including those due to chemical effects, experimental protocol, and debris constituents. This conclusion is based on the limits on latent debris and submerged aluminum specified in the amendment. These cleanliness specifications should not be relaxed without additional analyses, a much wider range of experiments at prototypical conditions, and NRC review of these findings.

In summary, debris generation during DBAs has been minimized by the choice of LOCA-resistant insulation and coatings. This, together with the large flow area sump screens, results in negligible head losses except in the inlet regions of the core. With regard to in-vessel debris effects, the acceptance criterion established by the applicant is adequate to assure LTC. The criterion is met with sufficient margin to account for uncertainties provided the stringent cleanliness requirements specified in the amendment are maintained. The AP1000 design, therefore, meets the regulatory requirements for LTC during design basis accidents.

Sincerely,

/RA/

Said Abdel-Khalik
Chairman

References:

1. Letter to Edwin M. Hackett, “AP1000 Subcommittee Review of Selected Chapters of the Advanced Safety Evaluation Report – AP1000 Design Certification Amendment,” 09/20/2010 (ML102580168, Chapter 6 SER ML102410012)

2. Letter to U.S. Nuclear Regulatory Commission, Submittal of APP-GW-GLE-002 Revision 7 – “Impacts to the AP1000™ to Address Generic Safety Issue (GSI) – 191,” 07/13/2010 (ML101970030)

3. Westinghouse Technical Report, WCAP-17028-P Revision 6, “Evaluation of Debris-Loading Head-Loss Tests for AP1000™ Fuel Assemblies During Loss of Coolant Accidents,” 06/29/2010 (ML102030227; ML102030219; ML102030221; ML102030222; ML102030223; ML102030215)

4. Westinghouse Technical Report, WCAP-16914-PR0, “Evaluation of Debris Loading Head Loss Tests for AP1000 Recirculation Screens and In-Containment Refueling Water Storage Tank Screens,” 07/14/2010 (ML102000156)

5. Letter to U.S. Nuclear Regulatory Commission, “Transmittal of IRWST and CR Screen Related Documents,” 02/26/2010 (ML100640574)

6. Letter to U.S. Nuclear Regulatory Commission, “Transmittal of IRWST and CR Screen Related Documents, Enclosure 21,” 02/26/2010 (ML100640578)

7. Letter to U.S. Nuclear Regulatory Commission, “Transmittal of IRWST and CR Screen Related Documents, Enclosure 22, APP-GW-GLR-092 Proprietary Rev. 0,” 02/26/2010 (ML100640585)

8. Letter to U.S. Nuclear Regulatory Commission, “Transmittal of IRWST and CR Screen Related Documents, Enclosure 25, APP-GW-GLR-110 Rev. 0 Proprietary,” 02/26/2010 (ML100640586)

9. Letter to U.S. Nuclear Regulatory Commission, “Transmittal of Technical Report APP-GW-GLR-079 Revision 8 (TR-026), (Proprietary & Non-Proprietary) “AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA” Enclosure 3, 07/20/2010 (ML102170124)

10. Letter to U.S. Nuclear Regulatory Commission, “Submittal of APP-GW-GLE-002 Revision 7 - Impacts to the AP I000™ to Address Generic Safety Issue (GSI) - 191,” 12/13/2010 (ML101970030)

11. Letter to U.S. Nuclear Regulatory Commission, “AP1000 Response to Request for Additional Information (SRP6.2.2),” Enclosure, 07/30/2010 (ML102160216)

In view of the relatively narrow range of conditions explored in the applicant's test program and the significant uncertainties in the results, a site visit was conducted to better understand the AP1000 related results in the context of in-core debris effects found in the PWR Owner's Group (PWROG) experiments. These cover a wide range of conditions and, while not directly applicable to the AP1000, offer valuable insights into the effects of various experimental parameters.

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Sincerely,

/RA/

Said Abdel-Khalik
Chairman

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Letter to the Honorable Gregory B Jaczko, Chairman, NRC, from Said Abdel-Khalik, Chairman, ACRS, dated December 20, 2010

SUBJECT: LONG-TERM CORE COOLING FOR THE WESTINGHOUSE AP1000 PRESSURIZED WATER REACTOR

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New Reactor Licensing Applications

Schedules By Calendar Year

2005 2006 2007 2008 2009 2010 2011 2012 2013 2014 2015 2016 2017 2018

03/24/11

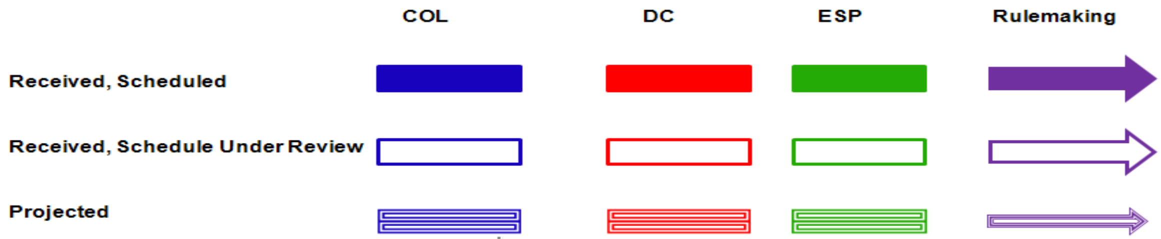
Schedules depict completion of staff safety and environmental reviews. COL and ESP Projects marked as "SCHEDULED" reflect four months additional time to complete mandatory hearings. Issuance of license is dependent upon completion of hearing process as well as design certification rulemaking for the selected design.

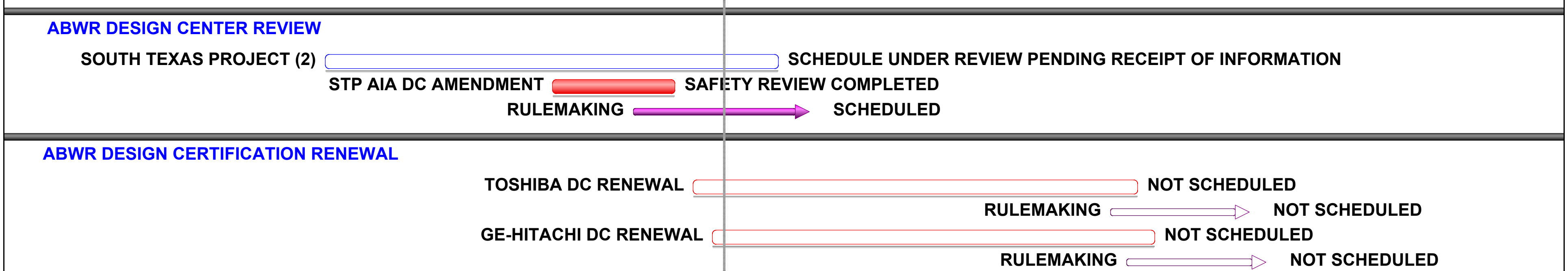
Schedule begin date is reflected as docketing date, or expected docketing date, following staff acceptance review.

Schedules depicted for future activities represent nominal assumed review durations based on submittal time frames in letters of intent from prospective applicants.

Where applicable, actual schedules are used, based on schedules as shown on NRC public web pages. For schedules under review, projected schedules are based on schedules as estimated by the NRC given the latest information the staff has. Schedules for COLs representing design certifications that are under schedule review will be adjusted once DC schedule is finalized.

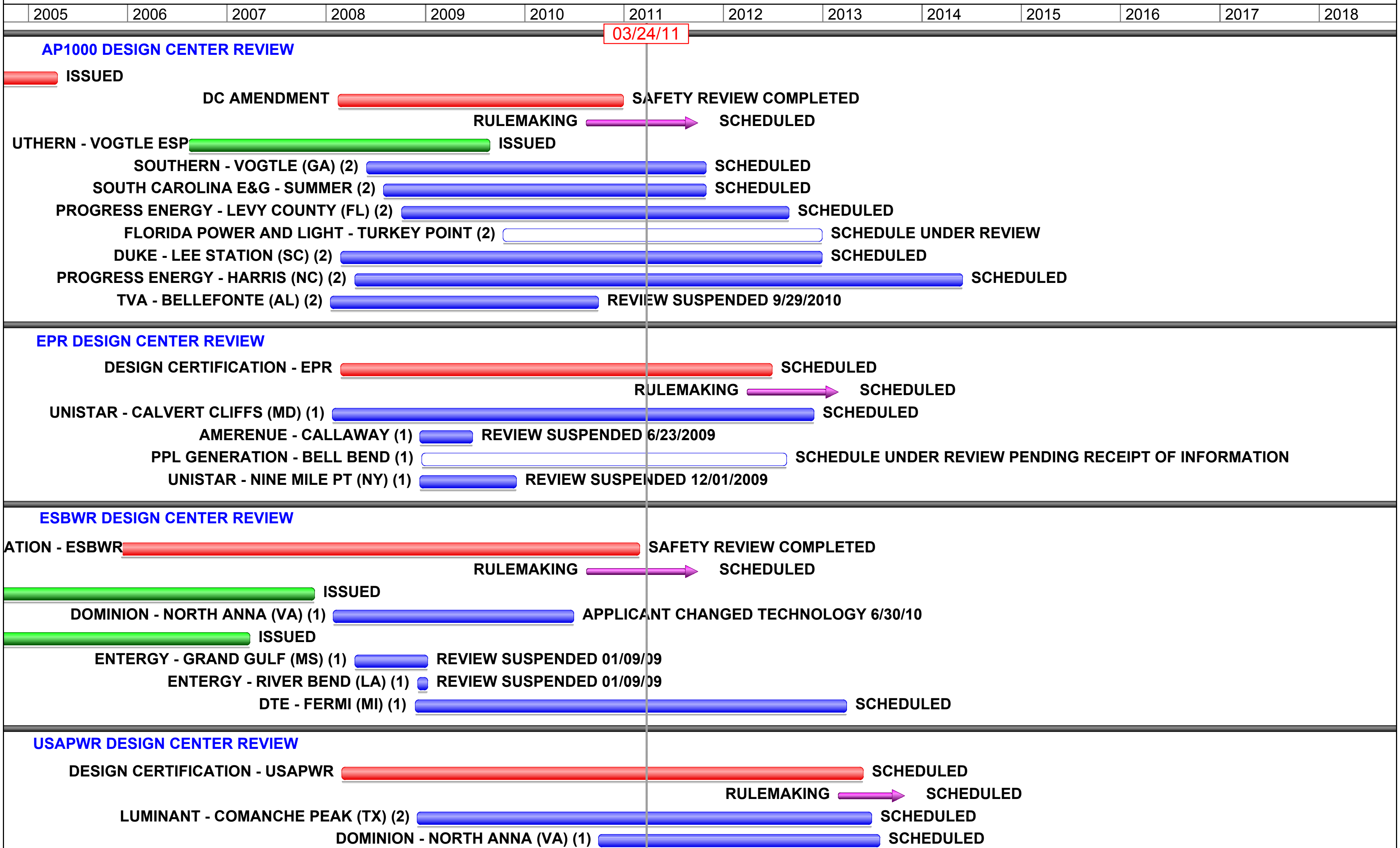
Numbers in () next to COL name indicate number of units/site.





New Reactor Licensing Applications

Schedules By Calendar Year



Thu 03/24/11

New Reactor Licensing Applications
Schedules By Calendar Year