



ACRS Role in Design Certification

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**Advisory Committee on Reactor Safeguards
(ACRS)**

Quadripartite Working Group Meeting
October 9-10, 2008
Paris, France

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Design Certification

- Pre-approval by NRC of an essentially complete nuclear plant design including
 - Inspections, tests, analyses, and acceptance criteria (ITAAC)
 - Design Acceptance Criteria (DAC)
- Intended to achieve finality through rulemaking
- Facilitates standardization
- Reduces uncertainty by resolving design issues prior to license application

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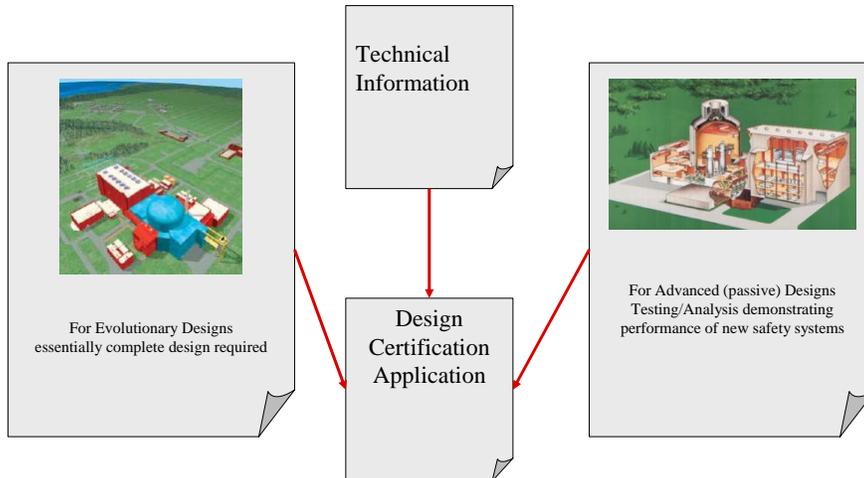
Finality of Design Certification

- ❑ Any modification the NRC imposes on a design certification will be applied to all plants referencing the certified design
- ❑ The Commission may not impose new requirements on a certified design unless:
 - ❑ compliance exceptions, or
 - ❑ adequate protection, and
 - ❑ special circumstances specifically defined in regulations

Finality of Design Certification (continued)

- Puts strong emphasis on the quality of initial review
 - ❑ Must get it right the first time
- Three levels of information Tier1/Tier2/Tier2*
- Changes are possible, however:
 - ❑ Change to Tier1 requires exemption
 - ❑ Change to Tier2* requires NRC approval

Contents of Applications



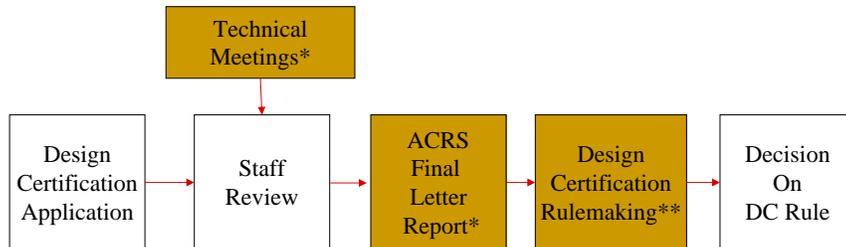
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ACRS Role and Regulatory Responsibility

- **10 CFR 52.53 requires that each application for a standard design certification be referred to the ACRS for a review and report on those portions of the application which concern safety**
 - Provide an independent review of NRC staff safety evaluation
 - Provide an open forum for public participation in the review process

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Design Certification Process



* Opportunity for Public Participation
** Notice and comment

ACRS Review Process

- **Begins early in the regulatory process**
- **Consists of a series of subcommittee meetings with NRC staff and applicant on preliminary safety findings and open issues**
- **Subcommittee develops a proposed ACRS position and reports to the full committee**
- **ACRS position developed after presentations and extensive deliberation by the full Committee**

ACRS Review Process (continued)

- **Interim ACRS letter reports issued as appropriate:**
 - To identify issues of concern
 - Identify items for which additional information, discussions, and clarifications are needed
- **Final ACRS letter report to NRC Chairman provides safety conclusion**

Example of ACRS Review (System 80+)

Subcommittee Meetings

Advanced PWR
6 meetings
April 1990 – February 1993

Joint I&C/Computers
3 meetings
February 1991- March 1992

ABB-CE Standard Plant Design
4 meetings
December 1993 – April 1994

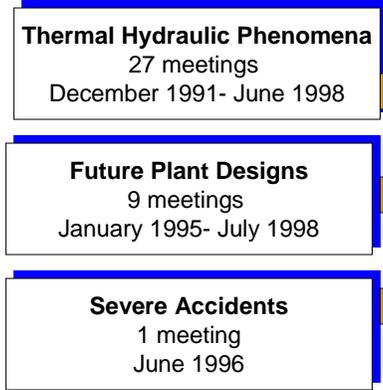
Full Committee Meetings

2 Meetings
September 1992
May 1994

Report to NRC Chairman
May 1994

Example of ACRS Review (AP-600)

Subcommittee Meetings



Full Committee Meetings

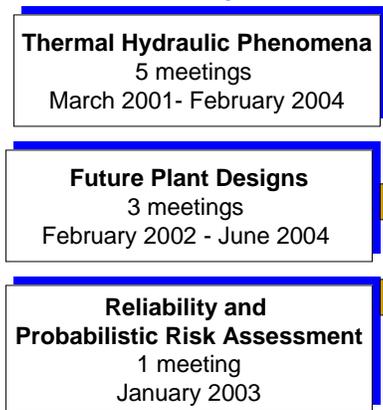
6 Meetings

August	1996
June	1997
February	1998
April	1998
May	1998
July	1998

Report to NRC Chairman
July 1998

Example of ACRS Review (AP1000)

Subcommittee Meetings



Full Committee Meetings

10 Meetings

August	2000
April	2001
March	2002
November	2002
February	2003
April	2003
October	2003
March	2004
June	2004
July	2004

Report to NRC Chairman
July 2004

ACRS Review Process (continued)

- **Focus of ACRS review is primarily on technical issues and their resolution**
- **Safety enhancement features over previous LWR designs are noted in reaching a conclusion on safety of design**
- **The certification application, NRC's safety evaluation, and meeting with both the NRC staff and applicant form the basis for the final report**

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ACRS APPROACH FOR AP1000 (Example)

- **Prior to the start of ACRS review, the Committee agreed on its approach in a strategic planning session**
 - Review all design changes from the AP600 design
 - Review experience with the 14 ft long fuel elements (any increased tendency for bowing/blockage)
 - Review scaling analysis to determine applicability of test and analysis program of AP600
 - Determine if separate effects tests for AP600 are applicable to AP1000
 - Request an uncertainty analysis for the codes used for DBA analyses to assess margins
 - Review critical accident sequences for water level and adequacy of the ADS4 squib valve
 - Determine if GOTHIC code is applicable to AP1000
 - Assess defense-in-depth function of containment [seismic, hydrogen stratification, PRA results for CDF and LERF, aerosol behavior]
 - Review PRA quality and results
 - Review staff's DSER and FSER for determination of compliance with the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations

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Specific Items (AP1000) - examples

- ❑ S4 squib valve. [How can the reliability be justified/demonstrated?]
- ❑ Strainer blockage issue. [How is the design designed to assure long-term cooling?]
- ❑ How are code deficiencies resolved or dispositioned?
- ❑ Scaling issues?
- ❑ In-vessel retention. [What is the potential for failure and subsequent fuel/coolant interaction?]

Illustrative Example (AP1000)

STRAINER BLOCKAGE/LONG-TERM COOLING ASSURANCE:

ISSUE: Barsebäck event in Sweden and other LOCA-like events in U.S. have shown significant debris generation and subsequent tendency to block inlet screens to the recirculation lines that provide for long-term cooling of PWRs. How has AP1000 design dealt with this issue?

RESOLUTION BASIS:

- The screens for AP1000 are more robust than current plants [bigger, higher off floor, significant barriers, known location of the depressurization]
- ITAAC item. This is a generic PWR issue that has not yet been resolved generically. ITAAC will ensure that AP1000 complies with the resolution

ACRS Conclusion

- **All ACRS safety concerns have been addressed**
- **Design is robust and that it can be built and operated without undue risk to the health and safety of the public**