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A Reactor That Can't Melt Down?

By Matthew L. Wald

MORE than a decade after the last order was placed in this country for a commercial nuclear plant, some engineers and utility analysts think it is time to start developing a second generation of reactors that would be meltdown-proof. Their hope is that these reactors would be available when the demand for power picks up and when cost or environmental impact makes fossil fuels unattractive.

A step toward a new reactor technology could come in the next few weeks as the Department of Energy entertains proposals for a new reactor to produce bomb fuel. Proponents of gas-graphite technology largely bypassed in this country are hoping that the Government will construct a graphite plant that could be a model for a new generation of civilian reactors.

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Graphite has had a bad name since the April 1986 accident at the Soviet Union's Chernobyl plant. There, the burning graphite core sent radioactive contaminants into the upper atmosphere, and around the world. But the reactor being proposed for the Department of Energy has a different design, which the manufacturer, GA Technologies Inc., said would not permit the melting of nuclear fuel.

The company says the plant "does not depend on active engineered safety features or human actions for safety of the public or the investment," and that passive features make a melting of the nuclear fuel impossible even if the operators make errors and all the mechanical safety equipment fails.

The reason is the substitution of graphite, which withstands extreme heat, for water, which heat drives off, and for metal, which heat will melt.

In all nuclear reactors, atoms of uranium or plutonium are struck by neutrons and split. This process generates heat and releases more neutrons, which split other atoms, continuing the chain reaction.

To slow the neutrons to the speed at which they are most likely to cause another atom to split, reactors use a medium called a moderator. Civilian reactors in this country, with one exception, use ordinary water (called light water) as the moderator.

The Soviet Union and a few other nations use graphite. The Department of Energy operates one graphite plant, the N-Reactor, in Hanford, Wash., to produce plutonium for bombs.

The design by GA, formerly General Atomics, differs markedly from the ones at Hanford and Chernobyl. A major difference from the Chernobyl plant is that as temperature rises, it tends to choke off the nuclear reaction. At Chernobyl, the opposite was true.

Unlike either the Chernobyl design or the N-Reactor, the GA design does not use metal to encase the nuclear fuel. Instead, GA uses a multilayered ceramic and carbon pellet, little bigger than a grain of sand. Linden Blue, GA's vice chairman, describes these pellets as pressure vessels — the term that in light-water plants is used to describe the multi-ton steel containers in which the water is heated to 600 degrees Fahrenheit.

The pellets are formed into cylinders about the size of a crayon, and the cylinders are placed into blocks of graphite a little bigger than a coffee table.

As the uranium in the pellets is split, the heat and the neutrons flow out of the pellets. But the "daughter atoms" created by the fission, which are the main source of radioactivity, are trapped inside.

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The graphite holds the fuel in place and moderates the neutrons' speed. In normal operation, the graphite blocks absorb the heat of the reaction and give it off to the helium gas that is used as a coolant. The gas is then used to boil water into steam to make electricity.

In an accident, the graphite can withstand very high temperatures without damage, even if the helium ceases to circulate. In contrast, in a light-water reactor, the water that functions as a moderator and coolant can boil away, and the metal used to hold the fuel in place can melt.

The graphite reactor is meltdown-proof, according to the designers, because it cannot reach 3,272 degrees Fahrenheit, the temperature at which the fuel pellets could break open and allow the radiation to escape.

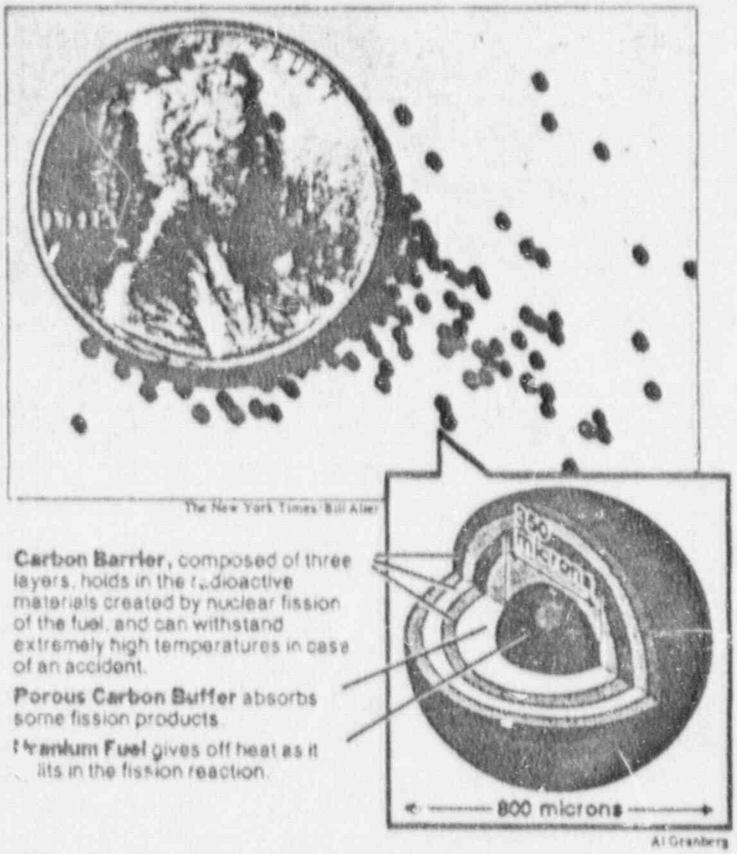
The reactor is designed to be buried in a silo, and even if the helium stops circulating, and if back-up heat-removal systems fail, heat will radiate to the silo walls and into the soil, holding temperatures to a maximum of 2,912 degrees Fahrenheit.

GA's proposal for a military reactor differs little in design from its civilian plant. The plan primarily adds "target elements" of lithium-six, a type of lithium that turns to tritium when it absorbs a neutron. The tritium is a type of hydrogen that is the fuel of H-bombs.

One commercial demonstration graphite plant was built: Fort St. Vrain, in the Colorado town of the same name. The plant is not a good advertisement for the technology, having produced only about 10 percent of the power that would have resulted if it could have run around the clock at 100 percent of capacity. GA says that Fort St. Vrain demonstrated the high quality of the fuel, and the benefits of using helium as the coolant instead of water. But the helium circulator was an untried design, and broke down frequently.

For a new plant, GA says it would use a design proved in Europe, where several graphite-moderated, helium-cooled reactors are operating successfully.

GA is competing with some well-established vendors. The Westinghouse Electric Corporation, which designs light water reactors, and Bechtel, the architect-engineering concern that has built many of them, have signed an agreement to bid for the construction of a light-water reactor or a heavy-water reactor, whichever the Department of Energy chooses.



The Carbon-Coated Fuel Particle

Carbon-coated fuel pellets that nest in heat-tolerant graphite blocks. Penny shown for relative size. Inset shows the cross-section of the fuel particle.

PART 52

EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS

54 FR 15372

Authority: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2133, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, 1246, as amended (42 U.S.C. 5841, 5842, 5846).

General Provisions

§ 52.1 Scope.

This part governs the issuance of early site permits, standard design certifications, and combined licenses for nuclear power facilities licensed under section 103 or 104b of the Atomic Energy Act of 1954, as amended (68 Stat. 919), and Title II of the Energy Reorganization Act of 1974 (88 Stat. 1242).

§ 52.3 Definitions.

As used in this part,

(a) "Combined license" means a combined construction permit and operating license with conditions for a nuclear power facility issued pursuant to Subpart C of this part.

(b) "Early site permit" means a Commission approval, issued pursuant to Subpart A of this part, for a site or sites for one or more nuclear power facilities.

(c) "Standard design" means a design which is sufficiently detailed and complete to support certification in accordance with ... part B of this part, and which is u... ... a multiple number of unit... ... multiple number of sites witho... ...pening or repeating the review.

(d) "Standard design certification", "design certification", or "certification" means a Commission approval, issued pursuant to Subpart B of this part, of a standard design for a nuclear power facility. A design so approved may be referred to as a "certified standard design".

(e) All other terms in this part have the meaning set out in 10 CFR 50.2, or section 11 of the Atomic Energy Act, as applicable.

§ 52.5 Interpretations.

54 FR 15372

Except as specifically authorized by the Commission in writing, no interpretation of the meaning of the regulations in this part by any officer or employee of the Commission other than a written interpretation by the General Counsel will be recognized to be binding upon the Commission.

§ 52.8 Information collection requirements: OMB approval.

(a) The Nuclear Regulatory Commission has submitted the information collection requirements contained in this part to the Office of Management and Budget (OMB) for approval as required by the Paperwork Reduction Act of 1980 (44 U.S.C. 3501, *et seq.*). OMB has approved the information collection requirements contained in this part under control number 3150(b).

(b) The approved information collection requirements contained in this part appear in §§ 52.15, 52.17, 52.29, 52.45, 52.47, 52.57, 52.75, 52.77, and 52.79.

Subpart A—Early Site Permits

§ 52.11 Scope of subpart.

This subpart sets out the requirements and procedures applicable to Commission issuance of early site permits for approval of a site or sites for one or more nuclear power facilities separate from the filing of an application for a construction permit or combined license for such a facility.

§ 52.13 Relationship to Subpart F of 10 CFR Part 2 and Appendix Q of this part.

The procedures of this subpart do not replace those set out in Subpart F of 10 CFR Part 2 or Appendix Q of this part. Subpart F applies only when early review of site suitability issues is sought in connection with an application for a permit to construct certain power facilities. Appendix Q applies only when NRC staff review of one or more site suitability issues is sought separately from and prior to the submittal of a construction permit. A Staff Site Report issued under Appendix Q in no way affects the authority of the Commission or the presiding officer in any proceeding under Subpart F or G of 10 CFR Part 2. Subpart A applies when any person who may apply for a construction permit under 10 CFR Part 50 or for a combined license under 10 CFR Part 52 seeks an early site permit from the Commission separately from an application for a construction permit or a combined license for a facility.

54 FR 15372

§ 52.15 Filing of applications.

(a) Any person who may apply for a construction permit under 10 CFR Part 50, or for a combined license under 10 CFR Part 52, may file with the Director of Nuclear Reactor Regulation an application for an early site permit. An application for an early site permit may be filed notwithstanding the fact that an application for a construction permit or a combined license has not been filed in connection with the site or sites for which a permit is sought.

(b) The application must comply with the filing requirements of 10 CFR 50.30 (a), (b), and (f) as they would apply to an application for a construction permit. The following portions of § 50.4, which is referenced by § 50.30(a)(1), are applicable: paragraphs (a), (b) (1)–(3), (c), (d), and (e).

§ 52.17 Contents of applications.

(a)(1) The application must contain the information required by 10 CFR 50.33 (a)–(d), the first three sentences of § 50.34(a)(1), and, to the extent approval of emergency plans is sought under paragraph (b)(2)(ii) of this section, the information required by § 50.33 (g) and (j), and § 50.34(b)(6)(v). In particular, the application should describe the following:

(i) The number, type, and thermal power level of the facilities for which the site may be used;

(ii) The boundaries of the site;

(iii) The proposed general location of each facility on the site;

(iv) The anticipated maximum levels of radiological and thermal effluents each facility will produce;

(v) The type of cooling systems, intakes, and outflows that may be associated with each facility;

(vi) The seismic, meteorological, hydrologic, and geologic characteristics of the proposed site (see Appendix A to 10 CFR Part 100);

(vii) The location and description of any nearby industrial, military, or transportation facilities and routes; and

(viii) The existing and projected future population profile of the area surrounding the site.

(2) A complete environmental report as required by 10 CFR 51.45 and 51.50 must be included in the application, provided, however, that such environmental report must focus on the environmental effects of construction and operation of a reactor, or reactors, which have characteristics that fall within the postulated site parameters, and provided further that the report need not include an assessment of the benefits (for example, need for power) of the proposed action, but must include an evaluation of alternative sites to determine whether there is any obviously superior alternative to the site proposed.

54 FR 15372

(b)(1) The application must identify physical characteristics unique to the proposed site, such as egress limitations from the area surrounding the site, that could pose a significant impediment to the development of emergency plans.

(2) The application may also either:

(i) Propose major features of the emergency plans, such as the exact sizes of the emergency planning zones, that can be reviewed and approved by NRC in consultation with FEMA in the absence of complete and integrated emergency plans; or

(ii) Propose complete and integrated emergency plans for review and approval by the NRC, in consultation with the Federal Emergency Management Agency, in accord with the applicable provisions of 10 CFR 50.47.

(3) Under paragraphs (b)(1) and (2)(i) of this section, the application must include a description of contacts and arrangements made with local, state, and federal governmental agencies with emergency planning responsibilities. Under the option set forth in paragraph (b)(2)(ii) of this section, the applicant shall make good faith efforts to obtain from the same governmental agencies certifications that: (i) The proposed emergency plans are practicable; (ii) These agencies are committed to participating in any further development of the plans, including any required field demonstrations, and (iii) that these agencies are committed to executing their responsibilities under the plans in the event of an emergency. The application must contain any certifications that have been obtained. If these certifications cannot be obtained, the application must contain information, including a utility plan, sufficient to show that the proposed plans nonetheless provide reasonable assurance that adequate protective measures can and will be taken, in the event of a radiological emergency at the site.

(c) If the applicant wishes to be able to perform, after grant of the early site permit, the activities at the site allowed by 10 CFR 50.10(e)(1) without first obtaining the separate authorization required by that section, the applicant shall propose, in the early site permit, a plan for redress of the site in the event that the activities are performed and the site permit expires before it is referenced in an application for a construction permit or a combined license issued under Subpart C of this part. The application must demonstrate that there is reasonable assurance that redress carried out under the plan will achieve an environmentally stable and aesthetically acceptable site suitable for whatever non-nuclear use may conform with local zoning laws.

54 FR 15372

§ 52.18 Standards for review of applications.

Applications filed under this subpart will be reviewed according to the applicable standards set out in 10 CFR Part 50 and its appendices and Part 100 as they apply to applications for construction permits for nuclear power plants. In particular, the Commission shall prepare an environmental impact statement during review of the application, in accordance with the applicable provisions of 10 CFR Part 51, provided, however, that the draft and final environmental impact statements prepared by the Commission focus on the environmental effects of construction and operation of a reactor, or reactors, which have characteristics that fall within the postulated site parameters, and provided further that the statements need not include an assessment of the benefits (for example, need for power) of the proposed action, but must include an evaluation of alternative sites to determine whether there is any obviously superior alternative to the site proposed. The Commission shall determine, after consultation with the Federal Emergency Management Agency, whether the information required of the applicant by § 52.17(b)(1) shows that there is no significant impediment to the development of emergency plans, whether any major features of emergency plans submitted by the applicant under § 52.17(b)(2)(i) are acceptable, and whether any emergency plans submitted by the applicant under § 52.17(b)(2)(ii) provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

§ 52.19 Permit and renewal fees.

The fees charged for the review of an application for the initial issuance or renewal of an early site permit are set forth in 10 CFR 170.12, together with a schedule for their deferred recovery. There is no application fee.

§ 52.21 Hearings.

An early site permit is a partial construction permit and is therefore subject to all procedural requirements in 10 CFR Part 2 which are applicable to construction permits, including the requirements for docketing in §§ 2.101(a)(1)–(4), and the requirements for issuance of a notice of hearing in §§ 2.104(a), (b)(1)(iv) and (v), (b)(2) to the extent it runs parallel to (b)(1)(iv) and (v), and (b)(3), provided that the designated sections may not be construed to require that the environmental report or draft or final environmental impact statement include an assessment of the benefits of the proposed action. In the hearing, the presiding officer shall also determine whether, taking into consideration the site criteria contained in 10 CFR Part 100, a reactor, or reactors, having characteristics that fall within the parameters for the site can be constructed and operated without undue risk to the health and safety of the public. All hearings conducted on applications for early site permits filed under this part are governed by the procedures contained in Subpart G of Part 2.

54 FR 15372

§ 52.23 Referral to the ACRS.

The Commission shall refer a copy of the application to the Advisory Committee on Reactor Safeguards (ACRS). The ACRS shall report on those portions of the application which concern safety.

§ 52.24 Issuance of early site permit.

After conducting a hearing under § 52.21 of this subpart and receiving the report to be submitted by the Advisory Committee on Reactor Safeguards under § 52.23 of this subpart, and upon determining that an application for an early site permit meets the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations, and that notifications, if any, to other agencies or bodies have been duly made, the Commission shall issue an early site permit, in the form and containing the conditions and limitations, as the Commission deems appropriate and necessary.

§ 52.25 Extent of activities permitted.

(a) If an early site permit contains a site redress plan, the holder of the permit, or the applicant for a construction permit or combined license who references the permit, may perform the activities at the site allowed by 10 CFR 50.10(e)(1) without first obtaining the separate authorization required by that section, provided that the final environmental impact statement prepared for the permit has concluded that the activities will not result in any significant adverse environmental impact which cannot be redressed.

54 FR 15372

(b) If the activities permitted by paragraph (a) of this section are performed at any site for which an early site permit has been granted, and the site is not referenced in an application for a construction permit or a combined license issued under Subpart C of this part while the permit remains valid, then the early site permit must remain in effect solely for the purpose of site redress, and the holder of the permit shall redress the site in accordance with the terms of the site redress plan required by § 52.17(c). If, before redress is complete, a use not envisaged in the redress plan is found for the site or parts thereof, the holder of the permit shall carry out the redress plan to the greatest extent possible consistent with the alternate use.

§ 52.27 Duration of permit.

(a) Except as provided in paragraph (b) of this section, an early site permit issued under this subpart may be valid for not less than ten nor more than twenty years from the date of issuance.

(b)(1) An early site permit continues to be valid beyond the date of expiration in any proceeding on a construction permit application or a combined license application which references the early site permit and is docketed either before the date of expiration of the early site permit, or, if a timely application for renewal of the permit has been filed, before the Commission has determined whether to renew the permit.

(2) An early site permit also continues to be valid beyond the date of expiration in any proceeding on an operating license application which is based on a construction permit which references the early site permit, and in any hearing held under § 52.103 of this part before operation begins under a combined license which references the early site permit.

(c) An applicant for a construction permit or combined license may, at its own risk, reference in its application a site for which an early site permit application has been docketed but not granted.

§ 52.29 Application for renewal.

(a) Not less than twelve nor more than thirty-six months prior to the end of the initial twenty-year period, or any later renewal period, the permit holder may apply for a renewal of the permit. An application for renewal must contain all information necessary to bring up to date the information and data contained in the previous application.

54 FR 15372

(b) Any person whose interests may be affected by renewal of the permit may request a hearing on the application for renewal. The request for a hearing must comply with 10 CFR 2.714. If a hearing is granted, notice of the hearing will be published in accordance with 10 CFR 2.703.

(c) An early site permit, either original or renewed, for which a timely application for renewal has been filed, remains in effect until the Commission has determined whether to renew the permit. If the permit is not renewed, it continues to be valid in certain proceedings in accordance with the provisions of § 52.27(b).

(d) The Commission shall refer a copy of the application for renewal to the Advisory Committee on Reactor Safeguards (ACRS). The ACRS shall report on those portions of the application which concern safety and shall apply the criteria set forth in § 52.31.

§ 52.31 Criteria for renewal.

(a) The Commission shall grant the renewal if the Commission determines that the site complies with the Atomic Energy Act and the Commission's regulations and orders applicable and in effect at the time the site permit was originally issued, and any new requirements the Commission may wish to impose after a determination that there is a substantial increase in overall protection of the public health and safety or the common defense and security to be derived from the new requirements and that the direct and indirect costs of implementation of those requirements are justified in view of this increased protection.

(b) A denial of renewal on this basis does not bar the permit holder or another applicant from filing a new application for the site which proposes changes to the site or the way in which it is used which correct the deficiencies cited in the denial of the renewal.

§ 52.33 Duration of renewal.

Each renewal of an early site permit may be for not less than ten nor more than twenty years.

§ 52.35 Use of site for other purposes.

54 FR 15372

A site for which an early site permit has been issued under this subpart may be used for purposes other than those described in the permit, including the location of other types of energy facilities. The permit holder shall inform the Director of Nuclear Reactor Regulation of any significant uses for the site which have not been approved in the early site permit. The information about the activities must be given to the Director in advance of any actual construction or site modification for the activities. The information provided could be the basis for imposing new requirements on the permit, in accordance with the provisions of § 52.39. If the permit holder informs the Director that the holder no longer intends to use the site for a nuclear power plant, the Director shall terminate the permit.

§ 52.37 Reporting of defects and noncompliance; revocation, suspension, modification of permits for cause.

For purposes of Part 21 and 10 CFR 50.100, an early site permit is a construction permit.

§ 52.39 Finality of early site permit determinations.

(a)(1) Notwithstanding any provision in 10 CFR 50.109, while an early site permit is in effect under §§ 52.27 or 52.33 the Commission may not impose new requirements, including new emergency planning requirements, on the early site permit or the site for which it was issued, unless the Commission determines that a modification is necessary either to bring the permit or the site into compliance with the Commission's regulations and orders applicable and in effect at the time the permit was issued, or to assure adequate protection of the public health and safety or the common defense and security.

(2) In making the findings required for issuance of a construction permit, operating license, or combined license, or the findings required by § 52.103 of this part, if the application for the construction permit, operating license, or combined license references an early site permit, the Commission shall treat as resolved those matters resolved in the proceeding on the application for issuance or renewal of the early site permit, unless a contention is admitted that a reactor does not fit within one or more of the site parameters included in the site permit, or a petition is filed which alleges either that the site is not in compliance with the terms of the early site permit, or that the terms and conditions of the early site permit should be modified.

(i) A contention that a reactor does not fit within one or more of the site parameters included in the site permit may be litigated in the same manner as other issues material to the proceeding.

54 FR 15372

(ii) A petition which alleges that the site is not in compliance with the terms of the early site permit must include, or clearly reference, official NRC documents, documents prepared by or for the permit holder, or evidence admissible in a proceeding under Subpart G of Part 2, which show, prima facie, that the acceptance criteria have not been met. The permit holder and NRC staff may file answers to the petition within the time specified in 10 CFR 2.730 for answers to motions by parties and staff. If the Commission, in its judgment, decides, on the basis of the petition and any answers thereto, that the petition meets the requirements of this paragraph, that the issues are not exempt from adjudication under 5 U.S.C. 554(a)(3), that genuine issues of material fact are raised, and that settlement or other informal resolution of the issues is not possible, then the genuine issues of material fact raised by the petition must be resolved in accordance with the provisions in 554, 556, and 557 which are applicable to determining applications for initial licenses.

(iii) A petition which alleges that the terms and conditions of the early site permit should be modified will be processed in accord with 10 CFR 2.206. Before construction commences, the Commission shall consider the petition and determine whether any immediate action is required. If the petition is granted, then an appropriate order will be issued. Construction under the construction permit or combined license will not be affected by the granting of the petition unless the order is made immediately effective.

(iv) Prior to construction, the Commission shall find that the terms of the early site permit have been met.

(b) An applicant for a construction permit, operating license, or combined license who has filed an application referencing an early site permit issued under this subpart may include in the application a request for a variance from one or more elements of the permit. In determining whether to grant the variance, the Commission shall apply the same technically relevant criteria as were applicable to the application for the original or renewed site permit. Issuance of the variance must be subject to litigation during the construction permit, operating license, or combined license proceeding in the same manner as other issues material to those proceedings.

Subpart B—Standard Design Certifications

§ 52.41 Scope of subpart.

This subpart set out the requirements and procedures applicable to Commission issuance of rules granting standard design certification for nuclear power facilities separate from the filing of an application for a construction permit or combined license for such facility.

54 FR 15372

§ 52.43 Relationship to Appendices M, N, and O of this part.

(a) Appendix M to this part governs the issuance of licenses to manufacture nuclear power reactors to be installed and operated at sites not identified in the manufacturing license application. Appendix N governs licenses to construct and operate nuclear power reactors of duplicate design at multiple sites. These appendices may be used independently of the provisions in this subpart unless the applicant also wishes to use a certified standard design approved under this subpart.

54 FR 10632

(b) Appendix O governs the staff review and approval of preliminary and final standard designs. A staff approval under Appendix O in no way affects the authority of the Commission or the presiding officer in any proceeding under Subpart G of 10 CFR Part 2. Subpart B of Part 52 governs Commission approval, or certification, of standard designs by rulemaking.

54 FR 15372

(c) A final design approval under Appendix O is a prerequisite for certification of a standard design under this subpart. An application for a final design approval must state whether the applicant intends to seek certification of the design. If the applicant does so intend, the application for the final design approval must, in addition to containing the information required by Appendix O, comply with the applicable requirements of Part 52, Subpart B, particularly §§ 52.45 and 52.47.

§ 52.45 Filing of applications.

(a)(1) Any person may seek a standard design certification for an essentially complete nuclear power plant design which is an evolutionary change from light water reactor designs of plants which have been licensed and in commercial operation before the effective date of this rule.

(2) Any person may also seek a standard design certification for a nuclear power plant design which differs significantly from the light water reactor designs described in paragraph (a)(1) of this section or utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions.

(b) An application for certification may be filed notwithstanding the fact that an application for a construction permit or combined license for such a facility has not been filed.

(c)(1) Because a final design approval under Appendix O of this part is a prerequisite for certification of a standard design, a person who seeks such a certification and does not hold, or has not applied for, a final design approval, shall file with the Director of Nuclear Reactor Regulation an application for a final design approval and certification.

54 FR 15372

(2) Any person who seeks certification but already holds, or has applied for, a final design approval, also shall file with the Director of Nuclear Reactor Regulation an application for certification, because the NRC staff may require that the information before the staff in connection with the review for the final design approval be supplemented for the review for certification.

(d) The applicant must comply with the filing requirements of 10 CFR 50.30(a) (1)–(4), and (6) and 50.30(b) as they would apply to an application for a nuclear power plant construction permit. The following portions of § 50.4, which is referenced by § 50.30(a)(1), are applicable to the extent technically relevant: paragraphs (a); (b), except for paragraphs (b); (c); and (e).

§ 52.47 Contents of applications.

(a) The requirements of this paragraph apply to all applications for design certification.

(1) An application for design certification must contain:

(i) The technical information which is required of applicants for construction permits and operating licenses by 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100, and which is technically relevant to the design and not site-specific;

(ii) Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f);

(iii) The site parameters postulated for the design, and an analysis and evaluation of the design in terms of such parameters;

(iv) Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority Generic Safety Issues which are identified in the version of NUREG–0933 current on the date six months prior to application and which are technically relevant to the design;

(v) A design-specific probabilistic risk assessment;

(vi) Proposed tests, inspections, analyses, and acceptance criteria which are necessary and sufficient to provide reasonable assurance that, if the tests, inspections and analyses are performed and the acceptance criteria met, a plant which references the design is built and will operate in accordance with the design certification.

(vii) The interface requirements to be met by those portions of the plant for which the application does not seek certification. These requirements must be sufficiently detailed to allow completion of the final safety analysis and design-specific probabilistic risk assessment required by paragraph (a)(1)(v) of this section;

54 FR 15372

(viii) Justification that compliance with the interface requirements of paragraph (a)(1)(vii) of this section is verifiable through inspection, testing (either in the plant or elsewhere), or analysis. The method to be used for verification of interface requirements must be included as part of the proposed tests, inspections, analyses, and acceptance criteria required by paragraph (a)(1)(vi) of this section; and

(ix) A representative conceptual design for those portions of the plant for which the application does not seek certification, to aid the staff in its review of the final safety analysis and probabilistic risk assessment required by paragraph (a)(1)(v) of this section, and to permit assessment of the adequacy of the interface requirements called for by paragraph (a)(1)(vii) of this subsection.

(2) The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the NRC, and procurement specifications and construction and installation specifications by an applicant. The Commission will require, prior to design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if such information is necessary for the Commission to make its safety determination.

(3) The staff shall advise the applicant on whether any technical information beyond that required by this section must be submitted.

(b) This paragraph applies, according to its provisions, to particular applications:

(1) The application for certification of a nuclear power plant design which is an evolutionary change from light water reactor designs of plants which have been licensed and in commercial operation before the effective date of this rule must provide an essentially complete nuclear power plant design except for site-specific elements such as the service water intake structure and the ultimate heat sink.

54 FR 15372

(2)(i) Certification of a standard design which differs significantly from the light water reactor designs described in paragraph (b)(1) of this section or utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions will be granted only if

(A) (*1*) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof;

(*2*) Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof;

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

(*4*) The scope of the design is complete except for site-specific elements such as the service water intake structure and the ultimate heat sink; or

(B) There has been acceptable testing of an appropriately sited, full-size, prototype of the design over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions. If the criterion in paragraph (b)(2)(i)(A)(*4*) of this section is not met, the testing of the prototype must demonstrate that the non-certified portion of the plant cannot significantly affect the safe operation of the plant.

(ii) The application for final design approval of a standard design of the type described in this subsection must propose the specific testing necessary to support certification of the design, whether the testing be prototype testing or the testing required in the alternative by paragraph (b)(2)(i)(A) of this section.

The Appendix O final design approval of such a design must identify the specific testing required for certification of the design.

(3) An application seeking certification of a modular design must describe the various options for the configuration of the plant and site, including variations in, or sharing of, common systems, interface requirements, and system interactions. The final safety analysis and the probabilistic risk assessment should also account for differences among the various options, including any restrictions which will be necessary during the construction and startup of a given module to ensure the safe operation of any module already operating.

54 FR 15372

§ 52.48 Standards for review of applications.

Applications filed under this subpart will be reviewed for compliance with the standards set out in 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100 as they apply to applications for construction permits and operating licenses for nuclear power plants, and as those standards are technically relevant to the design proposed for the facility.

§ 52.49 Fees for review of applications.

The fees charged for the review of an application for the initial issuance or renewal of a standard design certification are set out in 10 CFR 170.12, together with a schedule for their deferred recovery. There is no application fee.

§ 52.51 Administrative review of applications.

(a) A standard design certification is a rule that will be issued in accordance with the provisions of Subpart H of 10 CFR Part 2, as supplemented by the provisions of this section. The Commission shall initiate the rulemaking after an application has been filed under § 52.45 and shall specify the procedures to be used for the rulemaking.

(b) The rulemaking procedures must provide for notice and comment and an opportunity for an informal hearing before an Atomic Safety and Licensing Board. The procedures for the informal hearing must include the opportunity for written presentations made under oath or affirmation and for oral presentations and questioning if the Board finds them either necessary for the creation of an adequate record or the most expeditious way to resolve controversies. Ordinarily, the questioning in the informal hearing will be done by members of the Board, using either the Board's questions or questions submitted to the Board by the parties. The Board may also request authority from the Commission to use additional procedures, such as direct and cross examination by the parties, or may request that the Commission convene a formal hearing under Subpart G of 10 CFR Part 2 on specific and substantial disputes of fact, necessary for the Commission's decision, that cannot be resolved with sufficient accuracy except in a formal hearing. The staff will be a party in the hearing.

54 FR 15372

(c) The decision in such a hearing will be based only on information on which all parties have had an opportunity to comment, either in response to the notice of proposed rulemaking or in the informal hearing. Notwithstanding anything in 10 CFR 2.790 to the contrary, proprietary information will be protected in the same manner and to the same extent as proprietary information submitted in connection with applications for construction permits and operating licenses under 10 CFR Part 50, provided that the design certification shall be published in Chapter I of this Title.

§ 52.53 Referral to the ACRS.

The Commission shall refer a copy of the application to the Advisory Committee on Reactor Safeguards (ACRS). The ACRS shall report on those portions of the application which concern safety.

§ 52.54 Issuance of standard design certification.

After conducting a rulemaking proceeding under § 52.51 on an application for a standard design certification and receiving the report to be submitted by the Advisory Committee on Reactor Safeguards under § 52.53, and upon determining that the application meets the applicable standards and requirements of the Atomic Energy Act and the Commission's regulations, the Commission shall issue a standard design certification in the form of a rule for the design which is the subject of the application.

§ 52.55 Duration of certification.

(a) Except as provided in paragraph (b) of this section, a standard design certification issued pursuant to this subpart is valid for fifteen years from the date of issuance.

(b) A standard design certification continues to be valid beyond the date of expiration in any proceeding on an application for a combined license or operating license which references the standard design certification and is docketed either before the date of expiration of the certification, or, if a timely application for renewal of the certification has been filed, before the Commission has determined whether to renew the certification. A design certification also continues to be valid beyond the date of expiration in any hearing held under § 52.103 before operation begins under a combined license which references the design certification.

54 FR 15372

(c) An applicant for a construction permit or combined license may, at its own risk, reference in its application a design for which a design certification application has been docketed but not granted.

§ 52.57 Application for renewal.

(a) Not less than twelve nor more than thirty-six months prior to expiration of the initial fifteen-year period, or any later renewal period, any person may apply for renewal of the certification. An application for renewal must contain all information necessary to bring up to date the information and data contained in the previous application. The Commission will require, prior to renewal of certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if such information is necessary for the Commission to make its safety determination. Notice and comment procedures must be used for a rulemaking proceeding on the application for renewal. The Commission, in its discretion, may require the use of additional procedures in individual renewal proceedings.

(b) A design certification, either original or renewed, for which a timely application for renewal has been filed remains in effect until the Commission has determined whether to renew the certification. If the certification is not renewed, it continues to be valid in certain proceedings, in accordance with the provisions of § 52.55.

(c) The Commission shall refer a copy of the application for renewal to the Advisory Committee on Reactor Safeguards (ACRS). The ACRS shall report on those portions of the application which concern safety and shall apply the criteria set forth in § 52.59.

§ 52.59 Criteria for renewal.

(a) The Commission shall issue a rule granting the renewal if the design, either as originally certified or as modified during the rulemaking on the renewal, complies with the Atomic Energy Act and the Commission's regulations applicable and in effect at the time the certification was issued, and any other requirements the Commission may wish to impose after a determination that there is a substantial increase in overall protection of the public health and safety or the common defense and security to be derived from the new requirements and that the direct and indirect costs of implementation of those requirements are justified in view of this increased protection. In addition, the applicant for renewal may request an amendment to the design certification. The Commission shall grant the amendment request if it determines that the amendment will comply with the Atomic Energy Act and the Commission's regulations in effect at the time of renewal. If the amendment request entails such an extensive change to the design certification that an essentially new standard design is being proposed, an application for a design certification shall be filed in accordance with § 52.45 and 52.47 of this part.

(b) Denial of renewal does not bar the applicant, or another applicant, from filing a new application for certification of the design, which proposes design changes which correct the deficiencies cited in the denial of the renewal.

§ 52.61 Duration of renewal.

Each renewal of certification for a standard design will be for not less than ten nor more than fifteen years.

54FR19622

§ 52.63 Finality of standard design certification

54 FR 15372

(a)(1) Notwithstanding any provision in 10 CFR 50.109, while a standard design certification is in effect under § 52.55 or 52.61, the Commission may not modify, rescind, or impose new requirements on the certification, whether on its own motion, or in response to a petition from any person, unless the Commission determines in a rulemaking that a modification is necessary either to bring the certification or the referencing plants into compliance with the Commission's regulations applicable and in effect at the time the certification was issued, or to assure adequate protection of the public health and safety or the common defense and security. The rulemaking procedures must provide for notice and comment and an opportunity for the party which applied for the certification to request an informal hearing which uses the procedures described in § 52.51 of this subpart.

(2) Any modification the NRC imposes on a design certification rule under paragraph (a)(1) of this section will be applied to all plants referencing the certified design, except those to which the modification has been rendered technically irrelevant by action taken under paragraphs (a)(3), (a)(4), or (b) of this section.

54 FR 15372

(3) While a design certification is in effect under § 52.55 or § 52.61, unless (i) a modification is necessary to secure compliance with the Commission's regulations applicable and in effect at the time the certification was issued or to assure adequate protection of the public health and safety or the common defense and security, and (ii) special circumstances as defined in 10 CFR 50.12(a) are present, the Commission may not impose new requirements by plant-specific order on any part of the design of a specific plant referencing the design certification if that part was approved in the design certification. In addition to the factors listed in § 50.12(a), the Commission shall consider whether the special circumstances which § 50.12(a)(2) requires to be present outweigh any decrease in safety that may result from the reduction in standardization caused by the plant-specific order.

(4) Except as provided in 10 CFR 2.758, in making the findings required for issuance of a combined license or operating license, or for any hearing under § 52.103, the Commission shall treat as resolved those matters resolved in connection with the issuance or renewal of a design certification.

(b)(1) An applicant or licensee who references a standard design certification may request an exemption from one or more elements of the design certification. The Commission may grant such a request only if it determines that the exemption will comply with the requirements of 10 CFR 50.12(a). In addition to the factors listed in § 50.12(a), the Commission shall consider whether the special circumstances which § 50.12(a)(2) requires to be present outweigh any decrease in safety that may result from the reduction in standardization caused by the exemption. The granting of an exemption on request of an applicant must be subject to litigation in the same manner as other issues in the operating license or combined license hearing.

(2) Subject § 50.59, a licensee who references a standard design certification may make changes to the design of the nuclear power facility, without prior Commission approval, unless the proposed change involves a change to the design as described in the rule certifying the design. The licensee shall maintain records of all changes to the facility and these records must be maintained and available for audit until the date of termination of the license.

54 FR 15372

(c) The Commission will require, prior to granting a construction permit, combined license, or operating license which references a standard design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if such information is necessary for the Commission to make its safety determinations, including the determination that the application is consistent with the certified design. This information may be acquired by appropriate arrangements with the design certification applicant.

Subpart C—Combined Licenses

§ 52.71 Scope of Subpart.

This subpart sets out the requirements and procedures applicable to Commission issuance of combined licenses for nuclear power facilities.

§ 52.73 Relationship to Subparts A and B.

An application for a combined license under this subpart may, but need not, reference a standard design certification issued under Subpart B of this part or an early site permit issued under Subpart A of this part, or both. In the absence of a demonstration that an entity other than the one originally sponsoring and obtaining a design certification is qualified to supply such design, the Commission will entertain an application for a combined license which references a standard design certification issued under Subpart B only if the entity that sponsored and obtained the certification supplies the certified design for the applicant's use.

§ 52.75 Filing of applications.

Any person except one excluded by 10 CFR 50.38 may file an application for a combined license for a nuclear power facility with the Director of Nuclear Reactor Regulation. The applicant shall comply with the filing requirements of 10 CFR 50.4 and 50.30 (a) and (b), except for paragraph (b)(6) of § 50.4, as they would apply to an application for a nuclear power plant construction permit. The fees associated with the filing and review of the application are set out in 10 CFR Part 170.

§ 52.77 Contents of applications; general information.

The application must contain all of the information required by 10 CFR 50.33, as that section would apply to applicants for construction permits and operating licenses, and 10 CFR 50.33a, as that section would apply to an applicant for a nuclear power plant construction permit. In particular, the applicant shall comply with the requirement of § 50.33a(b) regarding the submission of antitrust information.

§ 52.79 Contents of applications; technical information.

54 FR 15372

(a)(1) In general, if the application references an early site permit, the application need not contain information or analyses submitted to the Commission in connection with the early site permit, but must contain, in addition to the information and analyses otherwise required, information sufficient to demonstrate that the design of the facility falls within the parameters specified in the early site permit, and to resolve any other significant environmental issue not considered in any previous proceeding on the site or the design.

(2) If the application does not reference an early site permit, the applicant shall comply with the requirements of 10 CFR 50.30(f) by including with the application an environmental report prepared in accordance with the provisions of Subpart A of 10 CFR Part 51.

(3) If the application does not reference an early site permit which contains a site redress plan as described in § 52.17(c), and if the applicant wishes to be able to perform the activities at the site allowed by 10 CFR 50.10(e)(1), then the application must contain the information required by § 52.17(c).

(b) The application must contain the technically relevant information required of applicants for an operating license by 10 CFR 50.34. The final safety analysis report and other required information may incorporate by reference the final safety analysis report for a certified standard design. In particular, an application referencing a certified design must describe those portions of the design which are site-specific, such as the service water intake structure and the ultimate heat sink. An application referencing a certified design must also demonstrate compliance with the interface requirements established for the design under § 52.47(a)(1), and have available for audit procurement specifications and construction and installation specifications in accordance with § 52.47(a)(2). If the application does not reference a certified design, the application must comply with the requirements of § 52.47(a)(2) for level of design information, and shall contain the technical information required by §§ 52.47(a)(1) (i), (ii), (iv), and (v) and (3), and, if the design is modular, § 52.47(b)(3).

54 FR 15372

(c) The application for a combined license must include the proposed test, inspections, and analyses which the licensee shall perform and the acceptance criteria therefor which are necessary and sufficient to provide reasonable assurance that, if the tests, inspections and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the combined license. Where the application references a certified standard design, the test, inspections, analyses and acceptance criteria contained in the certified design must apply to those portions of the facility design which are covered by the design certification.

(d) The application must contain emergency plans which provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site.

(1) If the application references an early site permit, the application may incorporate by reference emergency plans, or major features of emergency plans, approved in connection with the issuance of the permit.

(2) If the application does not reference an early site permit, or if no emergency plans were approved in connection with the issuance of the permit, the applicant shall make good faith efforts to obtain certifications from the local and State governmental agencies with emergency planning responsibilities (i) that the proposed emergency plans are practicable, (ii) that these agencies are committed to participating in any further development of the plans, including any required field demonstrations, and (iii) that these agencies are committed to executing their responsibilities under the plans in the event of an emergency. The application must contain any certifications that have been obtained. If these certifications cannot be obtained, the application must contain information, including a utility plan, sufficient to show that the proposed plans nonetheless provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency at the site.

§ 52.81 Standards for review of applications.

Applications filed under this subpart will be reviewed according to the standards set out in 10 CFR Parts 20, 50, 51, 55, 73, and 100 as they apply to applications for construction permits and operating licenses for nuclear power plants, and as those standards are technically relevant to the design proposed for the facility.

§ 52.83 Applicability of Part 50 provisions.

54 FR 15372

Unless otherwise specifically provided in this subpart, all provisions of 10 CFR Part 50 and its appendices applicable to holders of construction permits for nuclear power reactors also apply to holders of combined licenses issued under this subpart. Similarly, all provisions of 10 CFR Part 50 and its appendices applicable to holders of operating licenses also apply to holders of combined licenses issued under this subpart, once the Commission has made the findings required under § 52.103, provided that, as applied to a combined license, 10 CFR 50.51 must require that the initial duration of the license may not exceed 40 years from the date on which the Commission makes the findings required under § 52.103. However, any limitations contained in Part 50 regarding applicability of the provisions to certain classes of facilities continue to apply.

§ 52.85 Administrative review of applications.

A proceeding on a combined license is subject to all applicable procedural requirements contained in 10 CFR Part 2, including the requirements for docketing (§ 2.101) and issuance of a notice of hearing (§ 2.104). All hearings on combined licenses are governed by the procedures contained in Part 2, Subpart G.

§ 52.87 Referral to the ACRS.

The Commission shall refer a copy of the application to the Advisory Committee on Reactor Safeguards (ACRS). The ACRS shall report on those portions of the application which concern safety and shall apply the criteria set forth in § 52.81, in accordance with the finality provisions of this part.

§ 52.89 Environmental review.

If the application references an early site permit or a certified standard design, the environmental review must focus on whether the design of the facility falls within the parameters specified in the early site permit and any other significant environmental issue not considered in any previous proceeding on the site or the design. If the application does not reference an early site permit or a certified standard design, the environmental review procedures set out in 10 CFR Part 51 must be followed, including the issuance of a final environmental impact statement, but excluding the issuance of a supplement under § 51.95(a).

§ 52.91 Authorization to conduct site activities.

(a)(1) If the application references an early site permit which contains a site redress plan as described in § 52.17(c) the applicant is authorized by § 52.25 to perform the site preparation activities described in 10 CFR 50.10(e)(1).

(2) If the application does not reference an early site permit which contains a redress plan, the applicant may not perform the site preparation activities allowed by 10 CFR 50.10(e)(1) without first submitting a site redress plan in accord with § 52.79(a)(3) and obtaining the separate authorization required by 10 CFR 50.10(e)(1). Authorization must be granted only after the presiding officer in the proceeding on the application has made the findings and determination required by 10 CFR 50.10(e)(2) and has determined that the site redress plan meets the criteria in § 52.17(c).

54 FR 15372

(3) Authorization to conduct the activities described in 10 CFR 50.10(e)(3)(i) may be granted only after the presiding officer in the combined license proceeding makes the additional finding required by 10 CFR 50.10(e)(3)(ii).

(b) If, after an applicant for a combined license has performed the activities permitted by paragraph (a) of this section, the application for the license is withdrawn or denied, and the early site permit referenced by the application expires, then the applicant shall redress the site in accord with the terms of the site redress plan. If, before redress is complete, a use not envisaged in the redress plan is found for the site or parts thereof, the applicant shall carry out the redress plan to the greatest extent possible consistent with the alternate use.

§ 52.93 Exemptions and variances.

(a) Applicants for a combined license under this subpart, or any amendment to a combined license, may include in the application a request, under 10 CFR 50.12, for an exemption from one or more of the Commission's regulations, including any part of a design certification rule. The Commission shall grant such a request if it determines that the exemption will comply with the requirements of 10 CFR 50.12(a) or 52.63(b)(1) if the exemption includes any part of the design certification rule.

(b) An applicant for a combined license, or any amendment to a combined license, who has filed an application referencing an early site permit issued under this subpart may include in the application a request for a variance from one or more elements of the permit. In determining whether to grant the variance, the Commission shall apply the same technically relevant criteria as were applicable to the application for the original or renewed site permit. Issuance of the variance must be subject to litigation during the combined license proceeding in the same manner as other issues material to that proceeding.

54 FR 15372

§ 52.97 Issuance of combined licenses.

(a) The Commission shall issue a combined license for a nuclear power facility upon finding that the applicable requirements of 10 CFR 50.40, 50.42, 50.43, 50.47, and 50.50 have been met, and that there is reasonable assurance that the facility will be constructed and operated in conformity with the license, the provisions of the Atomic Energy Act, and the Commission's regulations.

(b) The Commission shall identify in the license the tests, inspections, and analyses that the licensee shall perform and the acceptance criteria therefor which are necessary and sufficient to provide reasonable assurance that, if the tests, inspections, and analyses are performed and the acceptance criteria met, the facility has been constructed and will be operated in conformity with the license, the provisions of the Atomic Energy Act, and the Commission's regulations. Any modification to, addition to, or deletion from the terms of a combined license, including any modification to, addition to, or deletion from the tests, inspections, analyses, or related acceptance criteria contained in such license, is a proposed amendment to such license. There shall be an opportunity for a hearing on the proposed amendments, and any hearing held must be completed before operation of the facility.

§ 52.99 Inspection during construction.

After issuance of a combined license, the NRC staff shall assure that the required inspections, tests, and analyses are performed and that the prescribed acceptance criteria are met. Holders of combined licenses shall comply with the provisions of 10 CFR 50.70 and 50.71. At appropriate intervals during construction, the NRC staff shall publish in the Federal Register notices of the successful completion of inspections, tests, and analyses.

§ 52.101 Pre-operational antitrust review.

If, before the Commission makes the findings required uner § 52.103, the Commission, after consultation with the Attorney General, determines that significant changes in the licensee's activities or proposed activities have occurred subsequent to the previous review by the Attorney General and the Commission in connection with the issuance of the combined license, the antitrust review required by section 105c(2) of the Atomic Energy Act must be completed prior to commencement of commercial operation of the facility. Upon completion of this review, the Director of Nuclear Reactor Regulation may impose any additional license conditions as authorized by section 105c of the Atomic Energy Act.

54 FR 15372

§ 52.103 Operation under a combined license.

(a) Not less than 180 days before loading of fuel into the reactor, the holder of the combined license shall, in writing, notify the Commission of the expected dates of both fuel loading and criticality. The Commission shall publish notice of these dates in the Federal Register. The Federal Register notice must also advise persons whose interests may be affected by facility operation of their rights under paragraph (b) of this section.

(b)(1) Not later than 60 days after publication of the notice required by paragraph (a) of this section, any person whose interest may be affected by facility operation may file one or both of the following in writing:

(i) A petition which shows, prima facie, that one or more of the acceptance criteria in the combined license have not been met and, as a result, there is good cause to modify or prohibit operation; or

(ii) A petition to modify the terms and conditions of the combined license.

(2)(i) A good cause petition filed under paragraph (b)(1)(i) of this section will be granted by the Commission only if it includes, or clearly references, official NRC documents, documents prepared by or for the combined license holder, or evidence admissible in a proceeding under Subpart G of Part 2, which show, prima facie, that the acceptance criteria have not been met. The combined license holder and NRC staff may file answers to the petition within the time specified in 10 CFR 2.730 for answers to motions by parties and staff. If the Commission in its judgment decides, on the basis of the petitions and any answers thereto, that the petition meets the requirements of this paragraph, that the issues raised by the petition are not exempt from adjudication under 5 U.S.C. 554(a)(3), that genuine issues of material fact are raised, and that settlement or other informal resolution of the issues is not possible, then the genuine issues of material fact raised by the petition must be resolved in accordance with the provisions in 5 U.S.C. 554, 556, and 557 which are applicable to determining applications for initial licenses. In such cases, the notice of hearing from the Commission must specify the procedures to be followed. Matters exempt from adjudication under 5 U.S.C. 554(a)(3) may be decided by the Commission solely on the basis of the showing of good cause and any responsive pleadings.

54 FR 15372

(ii) A petition to modify the terms and conditions of the combined license will be processed as a request for action in accord with 10 CFR 2.206. The petitioner shall file the petition with the Secretary of the Commission. Before the licensed activity allegedly affected by the petition (fuel loading, low power testing, etc.) commences, the Commission shall consider the petition and determine whether any immediate action is required. If the petition is granted, then an appropriate order will be issued. Fuel loading and operation under the combined license will not be affected by the granting of the petition unless the order is made immediately effective.

(c) Prior to fuel loading, the Commission shall find that the acceptance criteria in the combined license have been met and that, accordingly, the facility has been constructed and will operate in conformity with the Atomic Energy Act and the Commission's regulations. If the combined license is for a modular design, each reactor module may require a separate finding as construction proceeds.

Appendices A–L [Reserved]

Appendix M—Standardization of Design; Manufacture of Nuclear Power Reactors; Construction and Operation of Nuclear Power Reactors Manufactured Pursuant to Commission License

Section 101 of the Atomic Energy Act of 1954, as amended, and § 50.10 of this chapter require a Commission license to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, use, import, or export any production or utilization facility. The regulations in Part 50 require the issuance of a construction permit by the Commission before commencement of construction of a production or utilization facility, and the issuance of an operating license before operation of the facility. The provisions of Part 50 relating to the facility licensing process are, in general, predicated on the assumption that the facility will be assembled and constructed on the site at which it is to be operated. In those circumstances, both facility design and site-related issues can be considered in the initial construction permit stage of the licensing process.

However, under the Atomic Energy Act, a license may be sought and issued authorizing the manufacture of facilities but not their construction and installation at the sites on which the facilities are to be operated. Prior to the "commencement of construction", as defined in § 50.10(c) of this chapter of a facility (manufactured pursuant to such a Commission license) on the site at which it is to operate—that is preparation of the site and installation of the facility—a construction permit that, among other things, reflects approval of the site on which the facility is to be operated, must be issued by the Commission. This appendix sets out the particular requirements and provisions applicable to such situations where nuclear power reactors to be manufactured pursuant to a Commission license and subsequently installed at the site pursuant to a Commission construction permit, are of the type described in § 50.22 of this chapter. It thus codifies one approach to the standardization of nuclear power reactors.

54 FR 15372

1. Except as otherwise specified in this appendix or as the context otherwise indicates, the provisions in Part 50 applicable to construction permits, including the requirement in § 50.58 of this chapter for review of the application by the Advisory Committee on Reactor Safeguards and the holding of a public hearing, apply in context, with respect to matters of radiological health and safety, environmental protection, and the common defense and security, to licenses pursuant to this Appendix M to manufacture nuclear power reactors (manufacturing licenses) to be operated at sites not identified in the license application.

2. An application for a manufacturing license pursuant to this Appendix M must be submitted, as specified in § 50.4 of this chapter and meet all the requirements of §§ 50.34(a) (1)–(9) and 50.34a (a) and (b) of this chapter except that the preliminary safety analysis report shall be designated as a "design report" and any required information or analyses relating to site matters shall be predicated on postulated site parameters which must be specified in the application. The application must also include information pertaining to design features of the proposed reactor(s) that affect plans for coping with emergencies in the operation of the reactor(s).

3. An applicant for a manufacturing license pursuant to this Appendix M shall submit with his application an environmental report as required of applicants for construction permits in accordance with Subpart A of Part 51 of this chapter, provided, however, that such report shall be directed at the manufacture of the reactor(s) at the manufacturing site, and, in general terms, at the construction and operation of the reactor(s) at a hypothetical site or sites having characteristics that fall within the postulated site parameters. The related draft and final environmental impact statement prepared by the Commission's regulatory staff will be similarly directed.

4. (a) Sections 50.10 (b) and (c), 50.12(b), 50.23, 50.30(d), 50.34(a)(10), 50.34a(c), 50.35 (a) and (c), 50.40(a), 50.45, 50.55(d), 50.56 of this chapter and Appendix J of Part 50 do not apply to manufacturing licenses. Appendices E and H of Part 50 apply to manufacturing licenses only to the extent that the requirements of these appendices involve facility design features.

(b) The financial information submitted pursuant to § 50.33(f) of this chapter and Appendix C of Part 50 shall be directed at a demonstration of the financial qualifications of the applicant for the manufacturing license to carry out the manufacturing activity for which the license is sought.

5. The Commission may issue a license to manufacture one or more nuclear power reactors to be operated at sites not identified in the license application if the Commission finds that:

54 FR 15372

(a) The applicant has described the proposed design of and the site parameters postulated for the reactor(s), including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features of components incorporated therein for the protection of the health and safety of the public.

(b) Such further technical or design information as may be required to complete the design report and which can reasonably be left for later consideration, will be supplied in a supplement to the design report.

(c) Safety features or components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted a research and development program reasonably designed to resolve any safety questions associated with such features of components; and

(d) On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved before any of the proposed nuclear power reactor(s) are removed from the manufacturing site and (ii) taking into consideration the site criteria contained in Part 100 of this chapter, the proposed reactor(s) can be constructed and operated at sites having characteristics that fall within the site parameters postulated for the design of the reactor(s) without undue risk to the health and safety of the public.

(e) The applicant is technically and financially qualified to design and manufacture the proposed nuclear power reactor(s).

(f) The issuance of a license to the applicant will not be inimical to the common defense and security or to the health and safety of the public.

(g) On the basis of the evaluations and analyses of the environmental effects of the proposed action required by Subpart A of Part 51 of this chapter and paragraph 3 of this Appendix, the action called for is the issuance of the license.

Note. When an applicant has supplied initially all of the technical information required to complete the application, including the final design of the reactor(s), the findings required for the issuance of the license will be appropriately modified to reflect that fact.

6. Each manufacturing license issued pursuant to this appendix will specify the number of nuclear power reactors authorized to be manufactured and the latest date for the completion of the manufacture of all such reactors. Upon good cause shown, the Commission will extend such completion date for a reasonable period of time.

7. The holder of a manufacturing license issued pursuant to this Appendix M shall submit to the Commission the final design of the nuclear power reactor(s) covered by the license as soon as such design has been completed. Such submittal shall be in the form of an application for amendment of the manufacturing license.

54 FR 15372

8. The prohibition in § 50.10(c) of this chapter against commencement of construction of a production or utilization facility prior to issuance ... a construction permit applies to the transport of a nuclear power reactor(s) manufactured pursuant to this appendix from the manufacturing facility to the site at which the reactor(s) will be installed and operated. In addition, such nuclear power reactor(s) shall not be removed from the manufacturing site until the final design of the reactor(s) has been approved by the Commission in accordance with paragraph 7.

9. An application for a permit to construct a nuclear power reactor(s) which is the subject of an application for a manufacturing license pursuant to this Appendix M need not contain such information or analyses as have previously been submitted to the Commission in connection with the application for a manufacturing license, but shall by §§ 50.34(a) and 50.34a of this chapter, sufficient information to demonstrate that the site on which the reactor(s) is to be operated falls within the postulated site parameters specified in the relevant manufacturing license application.

10. The Commission may issue a permit to construct a nuclear power reactor(s) which is the subject of an application for a manufacturing license pursuant to this Appendix M if the Commission (a) finds that the site on which the reactor is to be operated falls within the postulated site parameters specified in the relevant application for a manufacturing license and (b) makes the findings otherwise required by Part 50. In no event will a construction permit be issued until the relevant manufacturing license has been issued.

11. An operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this Appendix M may be issued by the Commission pursuant to § 50.57 and Subpart A of Part 51 of this chapter except that the Commission shall find, pursuant to § 50.57(a)(1), that construction of the reactor(s) has been substantially completed in conformity with both the manufacturing license and the construction permit and the applications therefor, as amended, and the provisions of the Act, and the rules and regulations of the Commission. Notwithstanding the other provisions of this paragraph, no application for an operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this Appendix M will be docketed until the application for an amendment to the relevant manufacturing license required by paragraph 7 has been docketed.

54 FR 15372

12. In making the findings required by this part for the issuance of a construction permit or an operating license for a nuclear power reactor(s) that has been manufactured under a Commission license issued pursuant to this appendix, or an amendment to such a manufacturing license, construction permit, or operating license, the Commission will treat as resolved those matters which have been resolved at an earlier stage of the licensing process, unless there exists significant new information that substantially affects the conclusion(s) reached at the earlier stage or other good cause.

Appendix N—Standardization of Nuclear Power Plant Designs: Licenses To Construct and Operate Nuclear Power Reactors of Duplicate Design at Multiple Sites

Section 101 of the Atomic Energy Act of 1954, as amended, and § 50.10 of this chapter require a Commission license to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, use, import or export any production or utilization facility. The regulations in Part 50 require the issuance of a construction permit by the Commission before commencement of construction of a production or utilization facility, except as provided in § 50.10(e) of this chapter, and the issuance of an operating license before the operation of the facility.

The Commission's regulations in Part 2 of this chapter specifically provide for the holding of hearings on particular issues separately from other issues involved in hearings in licensing proceedings (§ 2.761a, Appendix A, section I(c)), and for the consolidation of adjudicatory proceedings and of the presentations of parties in adjudicatory proceedings such as licensing proceedings (§§ 2.715a, 2.716).

This appendix sets out the particular requirements and provisions applicable to situations in which applications are filed by one or more applicants for licenses to construct and operate nuclear power reactors of essentially the same design to be located at different sites.[1]

1. Except as otherwise specified in this appendix or as the context otherwise indicates, the provisions of Part 50, applicable to construction permits and operating licenses, including the requirement in § 50.58 of this chapter for review of the application by the Advisory Committee on Reactor Safeguards and the holding of public hearings, apply to construction permits and operating license subject to this Appendix N.

2. Applications for construction permits submitted pursuant to this Appendix must include the information required by §§ 50.33, 50.33a, 50.34(a) and 50.34a (a) and (b) of this chapter, and be submitted as specified in § 50.4 of this chapter. The applicant shall also submit the information required by § 51.50 of this chapter.

For the technical information required by §§ 50.34(a) (1) through (5) and (8) and 50.34a (a) and (b) of this chapter, reference may be made to a single preliminary safety analysis of the design[2] which, for the purposes of § 50.34(a)(1) includes one set of site parameters postulated for the design of the reactors, and an analysis and evaluation of the reactors in terms of such postulated site parameters. Such single preliminary safety analysis shall also include information pertaining to design features of the proposed reactors that affect plans for coping with emergencies in the operation of the reactors, and shall describe the quality assurance program with respect to aspects of design, fabrication, procurement and construction that are common to all of the reactors.

54 FR 15372

3. Applications for operating licenses submitted pursuant to this Appendix N shall include the information required by §§ 50.33, 50.34 (b) and (c), and 50.34a(c) of this chapter. The applicant shall also submit the information required by § 51.53 of this chapter. For the technical information required by §§ 50.34(b) (2) through (5) and 50.34a(c), reference may be made to a single final safety analysis of the design.

Appendix O—Standardization of Design: Staff Review of Standard Designs

This appendix sets out procedures for the filing, staff review and referral to the Advisory Committee on Reactor Safeguards of standard designs for a nuclear power reactor of the type described in § 50.22 of this chapter or major portions thereof.

1. Any person may submit a proposed preliminary of final standard design for a

[1] If the design for the power reactor(s) proposed in a particular application is not identical to the others, that application may not be processed under this appendix and Subpart D of Part 2 of this chapter.

[2] As used in this appendix, the design of a nuclear power reactor included in a single referenced safety analysis report means the design of those structures, systems and components important to radiological health and safety and the common defense and security.

nuclear power reactor of the type described in § 50.22 to the regulatory staff for its review. Such a submittal may consist of either the preliminary or final design for the entire reactor facility or the preliminary or final design of major portions thereof.

2. The submittal for review of the standard design must be made in the same manner and in the same number of copies as provided in §§ 50.4 and 50.30 of this chapter for license applications.

3. The submittal for review of the standard design shall include the information described in §§ 50.33 (a) through (d) of this chapter and the applicable technical information required by §§ 50.34 (a) and (b), as appropriate, and 50.34a of this chapter (other than that required by §§ 50.34(a) (8) and (10), 50.34(b)(1), (6) (i), (ii), (iv), and (v) and 50.34(b) (7) and (8)). The submittal shall also include a description, analysis and evaluation of the interfaces between the submitted design and the balance of the nuclear power plant. With respect to the requirements of §§ 50.34(a)(1) of this chapter, the submittal for review of a standard design shall include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of such postulated site parameters. The information submitted pursuant to § 50.34(a)(7) of this chapter shall be limited to the quality assurance program to be applied to the design, procurement and fabrication of the structures, systems, and components for which design review has been requested and the information submitted pursuant to § 50.34(a)(9) of this chapter shall be limited to the qualifications of the person submitting the standard design to design the reactor or major portion thereof. The submittal shall also include information pertaining to design features that affect plans for coping with emergencies in the operation of the reactor or major portion thereof.

54 FR 15372

4. Once the regulatory staff has initiated a technical review of a submittal under this appendix, the submittal will be referred to the Advisory Committee on Reactor Safeguards (ACRS) for a review and report.

5. Upon completion of their review of a submittal under this appendix, the regulatory staff shall publish in the Federal Register a determination as to whether or not the preliminary or final design is acceptable, subject to such conditions as may be appropriate, and make available in the Public Document Room an analysis of the design in the form of a report. An approved design shall be utilized by and relied upon by the regulatory staff and the ACRS in their review of any individual facility license application which incorporates by reference a design approved in accordance with this paragraph unless there exists significant new information which substantially affects the earlier determination or other good cause.

6. The determination and report by the regulatory staff shall not constitute a commitment to issue a permit or license, or in any way affect the authority of the Commission, Atomic Safety and Licensing Appeal Panel, Atomic Safety and Licensing Board Panel, and other presiding officers in any proceeding under Subpart G of Part 2 of this chapter.

7. Information requests to the approval holder regarding an approved design shall be evaluated prior to issuance to ensure that the burden to be imposed on respondents is justified in view of the potential safety significance of the issue to be addressed in the requested information. Each such evaluation performed by the NRC staff shall be in accordance with 10 CFR 50.54(f) and shall be approved by the Executive Director for Operations or his or her designee prior to issuance of the request.

Appendix P [Reserved]

Appendix Q—Pre-Application Early Review of Site Suitability Issues

This appendix sets out procedures for the filing, Staff review, and referral to the Advisory Committee on Reactor Safeguards (ACRS) of requests for early review of one or more site suitability issues relating to the construction and operation of certain utilization facilities separately from and prior to the submittal of applications for construction permits for the facilities. The appendix also sets out procedures for the preparation and issuance of Staff Site Reports and for their incorporation by reference in applications for the construction and operation of certain utilization facilities. The utilization facilities are those which are subject to § 51.20(b) of this chapter and are of the type specified in § 50.21(b) (2) or (3) or § 50.22 of this chapter or are testing facilities. This appendix does not apply to proceedings conducted pursuant to Subpart F or Part 2 of this chapter.

54 FR 15372

1. Any person may submit information regarding one or more site suitability issues to the Commission's Staff for its review separately from and prior to an application for a construction permit for a facility. Such a submittal shall be accompanied by any fee required by Part 170 of this chapter and shall consist of the portion of the information required of applicants for construction permits by §§ 50.33 (a)–(c) and (e) of this chapter, and, insofar as it relates to the issue(s) of site suitability for which early review is sought, by §§ 50.34(a)(1) and 50.30(f) of this chapter, except that information with respect to operation of the facility at the projected initial power level need not be supplied.

2. The submittal for early review of site suitability issue(s) must be made in the same manner and in the same number of copies as provided in §§ 50.4 and 50.30 of this chapter for license applications. The submittal must include sufficient information concerning range of postulated facility design and operation parameters to enable the Staff to perform the requested review of site suitability issues. The submittal must contain suggested conclusions on the issues of site suitability submitted for review and must be accompanied by a statement of the bases or the reasons for those conclusions. The submittal must also list, to the extent possible, any long-range objectives for ultimate development of the site, state whether any site selection process was used in preparing the submittal, describe any site selection process used, and explain what consideration, if any, was given to alternative sites.

3. The staff shall publish a notice of docketing of the submittal in the Federal Register, and shall send a copy of the notice of docketing to the Governor or other appropriate official of the State in which the site is located. This notice shall identify the location of the site, briefly describe the site suitability issue(s) under review, and invite comments from Federal, State, and local agencies and interested persons within 120 days of publication or such other time as may be specified, for consideration by the staff in connection with the initiation or outcome of the review and, if appropriate by the ACRS, in connection with the outcome of their review. The person requesting review shall serve a copy of the submittal on the Governor or other appropriate official of the State in which the site is located, and on the chief executive of the municipality in which the site is located or, if the site is not located in a municipality, on the chief executive of the county. The portion of the submittal containing information requested of applicants for construction permits by §§ 50.33 (a)–(c) and (e) and 50.34(a)(1) of this chapter will be referred to the ACRS for a review and report. There will be no referral to the ACRS unless early review of the site safety issues under § 50.34(a)(1) is requested.

54 FR 15372

4. Upon completion of review by the staff and, if appropriate by the ACRS, of a submittal under this appendix, the staff shall prepare a Staff Site Report which shall identify the location of the site, state the site suitability issues reviewed, explain the nature and scope of the review, state the conclusions of the staff regarding the issues reviewed and state the reasons for those conclusions. Upon issuance of a Staff Site Report, the staff shall publish a notice of the availability of the report in the Federal Register and shall place copies of the report in the Commission's Public Document at 2120 L Street NW., Lower Level (Room LL–6), Washington, DC 20037, and in a Local Public Document Room(s) located near the site identified in the Staff Site Report. The staff shall also send a copy of the report to the Governor or other appropriate official of the State in which the site is located, and to the chief executive of the municipality in which the site is located or, if the site is not located in a municipality, to the chief executive of the county.

5. Any Staff Site Report prepared and issued in accordance with this appendix may be incorporated by reference, as appropriate, in an application for a construction permit for a utilization facility which is subject to § 51.20(b) of this chapter and is of the type specific in § 50.21(b) (2) or (3) or § 50.22 of this chapter or is a testing facility. The conclusions of the Staff Site Report will be reexamined by the staff where five years or more have elapsed between the issuance of the Staff Site Report and its incorporation by reference in a construction permit application.

6. Issuance of a Staff Site Report shall not constitute a commitment to issue a permit or license, to permit on-site work under § 50.10(e) of this chapter, or in any way affect the authority of the Commission, Atomic Safety and Licensing Appeal Panel, Atomic Safety and Licensing Board Panel, and other

presiding officers in any proceeding under Subpart F and/or G of Part 2 of this chapter.

7. The staff will not conduct more than one review of site suitability issues with regard to a particular site prior to the full construction permit review required by Subpart A of Part 51 of this chapter. The staff may decline to prepare and issue a Staff Site Report in response to a submittal under this appendix where it appears that, (a) in cases where no review of the relative merits of the submitted site and alternative sites under Subpart A of Part 51 of this chapter is requested, there is a reasonable likelihood that further staff review would identify one or more preferable alternative sites and the staff review of one or more site suitability issues would lead to an irreversible and irretrievable commitment of resources prior to the submittal of the analysis of alternative sites in the Environmental Report that would prejudice the later review and decision on alternative sites under Subpart F and/or G of Part 2 and Subpart A of Part 51 of this chapter; or (b) in cases where, in the judgment of the staff, early review of any site suitability issue or issues would not be in the public interest, considering (1) the degree of likelihood that any early findings on those issues would retain their validity in later reviews, (2) the objections, if any, of cognizant state or local government agencies to the conduct of an early review on those issues, and (3) the possible effect on the public interest of having an early, if not necessarily conclusive, resolution of those issues.

54 FR 15372

PART 52

EARLY SITE PERMITS; STANDARD DESIGN CERTIFICATIONS; AND COMBINED LICENSES FOR NUCLEAR POWER REACTORS

STATEMENTS OF CONSIDERATION

54 FR 15372
Published 4/18/89
Effective 5/18/89

10 CFR Parts 2, 50, 51, 52, and 170

RIN 3150-AC61

Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission is now adding a new part to its regulations which provides for issuance of early site permits, standard design certifications, and combined construction permits and operating licenses with conditions for nuclear power reactors. The new part sets out the review procedures and licensing requirements for applications for these new licenses and certifications. The final action is intended to achieve the early resolution of licensing issues and enhance the safety and reliability of nuclear power plants.

EFFECTIVE DATE: May 18, 1989.

ADDRESS: Documents relative to this final rule may be examined and copied for a fee at the NRC Public Document Room, 2120 L Street NW, Washington, DC.

FOR FURTHER INFORMATION CONTACT: Steven Crockett, Attorney, Office of the General Counsel, telephone (301) 492-1600, on procedural matters, or Jerry Wilson, Office of Nuclear Regulatory Research, telephone (301) 492-3729, on technical matters, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

SUPPLEMENTARY INFORMATION:

I. Background

The Commission has long sought nuclear power plant standardization and the enhanced safety and licensing reform which standardization could make possible. For more than a decade, the Commission has been adding provisions to 10 CFR Part 50 and Part 2 that allow for limited degrees of standardization, and for as many years, the Commission has been proposing legislation to Congress on the subject. The Commission was frequently asked by Members of Congress to what extent legislation on the subject was necessary, and in doing the analysis necessary to reply to these questions, the Commission came to believe that much of what it sought could be accomplished within its current statutory authority. Thus the Commission embarked on standardization rulemaking.

The rulemaking process has been lengthy and highly public. A year and a half ago, the Commission announced its intent to pursue standardization rulemaking in its Policy Statement on Nuclear Power Plant Standardization (52 FR 34884, September 15, 1987). The Policy Statement set forth the principles that would guide the rulemaking and provided for a forty-five-day comment period on the Policy Statement. On October 20, 1987, about mid-way through the comment period the NRC staff held a public workshop on the Policy Statement. During the Workshop, the staff presented a detailed outline of the proposed rule and answered preliminary questions about it. A transcript of the workshop may be found in the Commission's public document room, Gelman Building, 2120 L Street, NW, Washington, DC. After a lengthy internal consideration of the comments received on the Policy Statement and the outline of the rule presented at the Workshop, and after public briefings of the Commission and the Advisory Committee on Reactor Safeguards (ACRS), the Commission issued a proposed rule (53 FR 32060; August 23, 1988) and provided for a sixty-day comment period. The comment period was extended to 75 days on October 24, 1988 (53 FR 41609). Mid-way through that period the NRC staff again held a public workshop, this time on the text of the proposed rule.[1]

During the second, 75-day comment period, the Commission received over 70 sets of comments, ranging from one-page letters to multi-paged documents, one of which included an annotated rewrite of the whole rule. The commenters included the Department of Energy (DOE), agencies and offices in the states of Connecticut, Indiana, New York, and North Carolina, the Nuclear Utility Management and Resources Council (NUMARC), the American Nuclear Energy Council, Westinghouse, General Electric, Combustion Engineering, Stone & Webster, the U.S. Chamber of Commerce, the Union of Concerned Scientists (UCS), the Nuclear Information and Resource Service (NIRS), the Ohio Citizens for Responsible Energy (OCRE), the Maryland Nuclear Safety Coalition, and several utilities, corporations, public interest groups, and individuals. All the comments may be viewed in the agency's public document room.

The Commission has carefully considered all the comments and wishes to express its sincere appreciation of the often considerable efforts of the commenters. While the broad outlines, and even many of the details, of the proposed rule remained unchanged in the final rule, few sections of the proposed rule have escaped revision in light of the comments, and some have been thoroughly revised. In the remainder of this section of this final rule preamble, the Commission makes two general responses to comments and then summarizes both the comments and its responses to them. In Section II of this final rule preamble, the Commission responds to comments on the chief issues raised by the comments. While Section II often touches on the broad policies which lie behind the rule, readers wishing to know more about those broad policies may consult the statement of considerations which was published with the proposed rule. In Section III, which proceeds section-by-section through the final rule, the Commission notes minor changes and offers some minor clarifications of the meaning of some provisions. For a complete record of the differences

[1] Given this lengthy and public process, the Commission is unpersuaded by commenters on the proposed rule who claim that the public was not given enough time to consider the rule. For example, the Nuclear Information Resource Service (NIRS) says that given the importance of the rule, one "would think that the NRC would encourage the widest possible public participation on this rule, perhaps even by making special efforts to solicit comment." That is, of course, precisely what the Commission did.

between the proposed rule and the final rule, readers may consult the comparative text of the final rule, which is available in the agency's public document room.

Two General Responses to Comments

Before summing up the comments and the Commission's responses to them, the Commission wishes to make clear what it has not tried to do in this rulemaking. First, although this is an important rulemaking, it does not resolve all the safety, environmental, and political issues facing nuclear power. The Commission received urgings to undertake deep reforms before issuing this final rule. The Commission was, for instance, urged to streamline the hearing procedures in 10 CFR Part 2, Subpart G, restructure the utilities' liabilities under the Price-Anderson Act, decide once and for all what safety criteria shall be applied to all future plants, solve the problem of nuclear waste, turn all health and safety regulation—not just the NRC's—over to the states, reconsider whether economic considerations should ever enter into safety decisions, conduct local running referenda on whether a given nuclear power plant should be built, and have Congress directly review designs. In sum, the Commission was urged to do everything before it did anything.

However, the Commission has stuck to the simple aim in this rulemaking of providing procedures for the standardization of nuclear power plants and more generally for the early resolution of safety and environmental issues in licensing proceedings. The Commission has declined to tie the fate of this rulemaking to the progress of the agency's many other ongoing efforts, such as revision of the agency's hearing procedures, implementation of the Policy Statement on Safety Goals (51 FR 30028, August 21, 1986), development of techniques of analysis of risk and cost, and preparation for the licensing of a high-level waste repository. The final rule necessarily touches on substance whenever it sets forth requirements for the technical content of applications for early site permits, design certifications, or combined licenses, or discusses the applicability of existing standards to new designs and new situations. But even here, the Commission has avoided establishing new safety or environmental standards, although the Commission may choose to adopt additional safety standards applicable to new designs prior to the advent of design certifications.

Second, many saw this rule as the occasion for arguments over the future viability of nuclear power in the United States. On the one hand, the Commission is vigorously accused of promoting the nuclear industry and shutting local governments and individual citizens out of the licensing process. On the other hand, the Commission is told that the licensing process is "the reason" for "the loss of the nuclear option", and that reform of that process is the "sine qua non" of the viability of that option.

Certainly, the Commission hopes that this rule will have a beneficial effect on the licensing process. In other words, the Commission hopes that effort has not been wasted on a rule which will never be used. But the Commission is not out to secure, single-handedly, the viability of the industry or to shut the general public out. The future of nuclear power depends not only on the licensing process but also on economic trends and events, the safety and reliability of the plants, political fortunes, and much else. The Commission's intent with this rulemaking is only to have a sensible and stable procedural framework in place for the consideration of future designs, and to make it possible to resolve safety and environmental issues before plants are built, rather than after.

Summary of the Comments and the Commission's Responses

The comments on the proposed rule are characterized both by their broad agreement that standardization and early resolution of licensing issues are desirable, and by their often deep differences on what kinds of designs should be certified, how they should be certified, and what consequences certification should have for the licensing process.

As to what kinds of designs should be certified, except for the very few who opposed any licensing of any nuclear power plant, no commenter opposes the certification of designs which differ significantly from the designs which have been built thus far, but some, UCS, for instance, say that only "advanced" designs should be certified, and many, including UCS, DOE, and Westinghouse, say that only designs for whole plants should be certified.

While not withholding certification from incomplete designs or designs which are not advanced, the final rule has moved a long way from the position the Commission took in the legislative proposal it made shortly before this rulemaking began. There, certification was held out only for evolutionary light water designs, but was permitted for the design of any "major portion" of a plant. The final rule provides for certification of advanced designs and permits certification of designs of less than full scope only in highly restricted circumstances.

As to how designs should be certified, most commenters think the Commission has authority to certify either by rule or by license. However, some commenters see advantages in certification by license. OCRE, for instance, says that certification by license is more appropriate, and some industry commenters think that more protections are available to the holder of a design license than are available to the "holder" of a design rule. Some commenters prefer certification by license because they believe that a hearing on a license has to be a formal adjudication.

The final rule reflects the Commission's long-standing preference for certification by rulemaking (see the old 10 CFR Part 50, Appendix O, paragraph 7), and for certification hearing procedures which, while they permit formal procedures when needed, do not assume that formal procedures are the best means for resolving every safety issues.

Finally, the deepest differences among the commenters concern the consequences of standardization and other devices for early resolution of licensing issues for the licensing process. One commenter believes that, once a plant is built under a combined license, there need be no hearing at all before operation begins. Several of these commenters characterize the proposed rule's provision for an opportunity for a hearing just before operation as the old two-step licensing process under a different name. Others believe not only that there should be such a hearing but also that resolution of issues in earlier proceedings does not entail any restriction on the issues which may be raised in the hearing after construction. Many of these commenters attribute to the Commission an intent to do away with public participation in the licensing process.

The Commission has given more consideration to this issue than to any other procedural question raised by the proposed rule. As a result, the proposed rule's provisions on hearings just before operation have been revised in the final rule (the revised provisions are discussed in more detail below). However, the final rule still provides for an opportunity for a hearing on limited issues before operation under a combined license. But the mere fact of this opportunity does not mean that the rule is hiding the old two-step process under a different name. By far the greater part of the issues which in the past have been considered in operating license hearings would, under the new rule, be considered at the combined license stage or in a certification proceeding, including the bulk of emergency planning issues. Similarly, the mere fact that any hearing prior to operation would be limited does not mean that the Commission is attempting to remove the public from the licensing process. The rule does not prevent the public from participating in the resolution of any operating license issue. It simply moves the bulk of the issues up front in the licensing process to the design certification, early site permit, and combined license parts of the process.

II. The Principal Issues

1. Requirements for Applications for Design Certification

Because design certification is the key procedural device in Part 52 for bringing about enhanced safety and early resolution of licensing issues, the Commission begins its discussion of the principal issues with responses to comments on the proposed rule's requirements for applications for certification.

a. "Advanced" Designs

The proposed rule provided for certification both of evolutionary light-water designs, that is, improved versions of the light-water designs now in operation, and of "advanced" designs, that is, designs which differ significantly from the evolutionary light-water designs, or which incorporate, to a greater extent than evolutionary light-water designs do, simplified, inherent, passive, or other innovative means to accomplish their safety functions (the distinction between evolutionary light-water designs and advanced designs is discussed at greater length below). The proposed rule required that some advanced designs could not be certified until full-scale prototypes of them were built and tested. While agreeing with the requirement for prototype testing of some advanced designs, several commenters, UCS prominent among them, say that certification should be held out only to advanced designs. UCS argues that without such a limitation on the designs which could be offered up for certification, the proposed rule would discriminate against the development of advanced designs of greater safety, because, given the choice between seeking certification of a familiar design and seeking certification of a design which the Commission might require to be tested in a full-scale prototype, an applicant would choose to avoid having to build a prototype.

As is noted above, the rule, unlike the legislative proposals which preceded it, provides for certification of advanced designs. However, it also provides for certification of evolutionary light-water designs. The Commission's legislative proposals on standardization have always focused on these designs, on the grounds that the light-water designs now in operation provide a high degree of protection to public health and safety. Moreover, the Commission does not believe that the requirement in some cases for a prototype is such a burden. Whatever burden having to test a prototype may be, the burden may be lessened by agreements of cost-sharing among utilities and other organizations, and by licensing the prototype for commercial operation. It is well to remember also that, under the rule, prototype testing is required only for certification or an unconditional final design approval, if at all. A final design approval under 10 CFR Part 52, Appendix O (formerly in Part 50) can be granted subject to conditions requiring prototype testing. See 10 CFR Part 52, Appendix O, paragraph 5. Moreover, a licensed prototype may be replicated.

b. Requirement to Address Unresolved Safety Issues and Safety Goals

Several commenters object to the proposed rule's requirement that applicants for certification propose technical resolutions of Unresolved Safety Issues and high- and medium-priority Generic Safety Issues. This requirement, and similar ones relating to probabilistic risk assessments and the Commission's Three Mile Island requirements for new plants, 10 CFR 50.34(f), were announced in the Commission's Severe Accident Policy Statement (50 FR 32138; August 8, 1985) and in the Commission's Policy Statement on Standardization (52 FR 34884; September 15, 1987). Some commenters call it "inappropriate" to impose this burden on applicants. Others say that no resolution of one of these issues should be imposed on a design unless the resolution had passed a cost-benefit test.

The Commission believes that it is not inappropriate to require that an applicant for certification show either that a particular issue is not relevant to the design proffered in the application, or that the applicant has in hand a design-specific resolution of the issue (the applicant is of course not required to propose a generic resolution of the issue). As to cost-benefit tests, the Commission will of course apply them to the resolution of safety issues where the resolutions are being imposed on existing plants and adequate protection is already secured. See 10 CFR 50.109 and *UCS* v. *NRC*, 824 F.2d 108 (D.C. Cir. 1987). However, initial certification does not involve backfitting. Designers will, of course, strive for a cost-effective design, but the Commission declines to incorporate a cost-benefit test in the standards for certification.

c. Requirements on Scope of Design and on Prototypes

In the statement of considerations accompanying the proposed rule, the Commission noted that the proposed rule permitted certification of incomplete designs only in limited cases, while the legislation the Commission had proposed to the 100th Congress had been less stringent about scope of design. The Commission invited comment on whether the final rule should return to the policy reflected in the proposed legislation. DOE, Westinghouse, and UCS, among others argue that only designs of complete power plants—excluding site-specific elements of course—should be certified. NUMARC, however, advocates a return to the policy of the legislation proposed to the 100th Congress. One engineering firm argues that requiring complete designs would limit market forces that could contribute to standardization.

The final rule is even more stringent about completeness of design than the proposed rule was. The final rule's provisions on scope, see § 52.47, reflect a policy that certain designs, especially designs which are evolutions of light-water designs now in operation, should not be certified unless they include all of a plant which can affect safe operation of the plant except its site-specific elements. See § 52.47(b). Examples of designs which are evolutions of currently operating light-water designs are General Electric's ABWR, Westinghouse's SP/90, and Combustion Engineering's System 80+. Full-scope may also be required of certain advanced designs, namely, the "passive" light-water designs such as General Electric's SBWR and Westinghouse's AP600. Considerations of safety, not market forces, constitute the basis for the final rule's requirement that these designs be full-scope designs. Long experience with operating light-water designs more than adequately demonstrates the adverse safety impact which portions of the balance of plant can have on the nuclear island. Given this experience, certification of these designs must be based on a consideration of the whole plant, or else the certifications of those designs will lack that degree of finality which should be the mark of certification.

However, the Commission has not adopted UCS's position that no design of incomplete scope could ever be certified. There is no reason to conclude that there could never be a design which protects the nuclear island against adverse effects caused by events in the balance of plant. The final rule therefore provides the opportunity for certification of designs of less than complete scope, if they belong to the class of advanced designs. See § 52.47(b). Examples of designs in this class include the passive light-water designs mentioned above and non-light-water designs such as General Electric's PRISM, Rockwell's SAFR, and General Atomic's MHTGR. But here too the rule sets a high standard. Certification of an advanced design of incomplete scope will be given only after a showing, using a full-scale prototype, that the balance of plant cannot significantly affect the safe operation of the plant.

Standardization along these lines may indeed limit some market forces, particularly those which encourage a highly differentiated range of products. However, the final rule's requirements on scope in no way limit innovative arrangements among vendors and architect-engineers for bringing new designs before the Commission.

The final rule is clearer than the proposed rule was in identifying those designs which cannot be certified without a program of testing. For purposes of determining which designs must undergo a testing program to be certified, the rule distinguishes between all advanced designs—be they passive light-water or non-light water—and evolutionary light-water designs. Some testing may be required of all advanced designs. Passive light-water designs are to some extent also evolutions of the light-water designs now licensed, but they have design features which are not present on plants licensed and operating in the United States. Therefore the rule requires that the maturity of the passive light-water designs be demonstrated through a combination of experience, appropriate tests, or analyses, but most likely not through prototype testing. See § 52.47(b)(2). While analyses may be relied upon by the staff to demonstrate the acceptability of a particular safety feature which evolved from previous experience or to justify the acceptability of a scale model test, it is very unlikely that an advanced design would be certified solely on the basis of analyses. Prototype testing is likely to be required for certificaton of advanced non-light-water designs because these revolutionary designs use innovative means to accomplish their safety functions, such as passive decay heat removal and reactivity control, which have not been licensed and operated in the United States. See id.

d. Certification by Rulemaking

The proposed rule provided for design certification by rulemaking. Here the proposed rule was in accord with the old 10 CFR Part 50, Appendix O, paragraph 7 (this paragraph is now being replaced by Subpart B of Part 52). However, in the notice of proposed rulemaking, the Commission invited comments on whether certification should be by license rather than rule. Although the Commission expressed some doubts on the matter, commenters generally agree that the Commission has the authority to license designs. Some industry commenters and some public interest groups alike go further and argue that certification by license is preferable. Industry commenters arguing this position believe that the rights and obligations which attach to a license are clearer than those which attach to a rule. For instance, a license is possessed by some entity and, under Commission law, cannot be transferred without that entity's consent. Some public interest groups prefer certification by license because they believe that the hearing on a license would have to be a formal adjudication.

The Commission continues to believe that certification by rule is preferable to certification by license. As DOE says, a design certification will, like a rule, have generic application. Moreover, certification by rulemaking leaves the Commission free to adapt hearing procedures to the requirements of the subject matter, rather than rely exclusively on formal adjudicatory devices even when they are not useful (hearing procedures are more fully discussed below). Finally, certification by rulemaking permits the Commission to consider reactor designs submitted by foreign corporations. However, the Commission will give priority to designs for which there is a demonstrated interest in the United States. The Commission will review other designs as resources permit.

For the reasons just given, the final rule retains provisions for certification by rulemaking. Westinghouse suggests also adding provisions for certification by license, leaving it to the applicant to choose between certification by license and certification by rulemaking. The Commission, however, prefers rulemaking and sees no advantage to providing such an option.

NUMARC, while supporting certification by rule, suggests adding provisions analogous to existing provisions in 10 CFR Part 50 for transfer or revocation of a license. See 10 CFR 50.80 and 50.100. However, a rule certifying a design does not, strictly speaking, belong to the designer. Therefore, such a rule cannot be transferred or revoked by adjudicatory enforcement. Applying § 50.80, in particular, to a rule certifying a design would be akin to giving the vendor of the design a patent, but the Commission has no authority to issue patents.

Nonetheless, the vendor whose design is certified by rule is not without protection. Section 52.63(a), the Administrative Procedure Act, and, ultimately, judicial review protect the vendor against arbitrary amendment or recission of the certification rule, and the law of patents and trade secrets protects the vendor against unlawful use of the design. In order to give the vendor more opportunity to treat elements of the design as trade secrets, the final rule provides that proprietary information contained in an application for design certification shall be given the same treatment that such information would be given in a proceeding on an application for a construction permit or an operating license under 10 CFR Part 50. See § 52.51. Moreover, an applicant referencing a design certification and seeking to use a designer other than the designer which achieved the certification would have to comply with §§ 52.63(c) and 52.73, and the other designer would have to pay a portion of the cost of review of the application for certification. See 10 CFR 170.12 (d) and (e), as amended in this document.

e. Applicability of Existing Standards

With one exception, the proposed rule did not say what safety standards would be applied to a design proffered for certification, or even precisely what existing information requirements applicants would have to meet.[8] In its lengthy and highly detailed comments, NUMARC proposes adding to the rule a large number of highly specific cross-references to Part 50, and a statement that no other portions of Part 50 apply.

The final rule provides that the standards set out in 10 CFR Part 20, Part 50 and its appendices, and Parts 73 and 100 will apply to the new designs where those standards are technically relevant to the design proposed for the facility. See new § 52.48. Application of Parts 20, 50, 73, and 100 to the certification of new

[8] The proposed rule did state that an application for certification would have to demonstrate that the design complied with the technically relevant portions of the Commission's Three Mile Island requirements set forth in 10 CFR 50.34(f). See § 52.47(a), 53 FR 32073 (proposed rule).

designs, as reflected in § 52.48, should go a long way toward establishing the regulatory standard that new designs must meet, and thereby provide the regulatory stability that is an essential prerequisite to realizing the benefits of standardization. The Commission recognizes that new designs may incorporate new features not addressed by the current standards in Parts 20, 50, 73 or 100 and that, accordingly, new standards may be required to address any such new design features. Therefore, the NRC staff shall, as soon as practicable, advise the Commission of the need for criteria for judging the safety of designs offered for certification that are different from or supplementary to current standards in 10 CFR Parts 20, 50, 73, and 100. The Commission shall consider the NRC staff's views and determine whether additional rulemaking is needed or appropriate to resolve generic questions that are applicable to multiple designs. The objective of such rulemaking would be to incorporate any new standards in Part 50 or Part 100, as appropriate, rather than to develop such standards in the context of the Commission's review and approval of individual applications for design certifications. On the other hand, new design features that are unique to a particular design would be addressed in the context of a rulemaking proceeding for that particular design.

f. Hearings on Applications for Design Certifications

Like the proposed rule, the final rule provides for notice and comment rulemaking on an application for a design certification, together with an opportunity for an informal hearing on an application for a design certification. The rule also permits the use of more formal procedures where they are the only procedures available for resolving a given issue properly. See § 52.51. UCS and others argue that any hearing on certification should be a formal adjudication. In particular, UCS argues that the certification proceeding will be dealing with adjudicative, as opposed to legislative, facts and therefore should be fully adjudicatory. UCS characterizes adjudicative facts as "uniquely related to activities of the parties that are at issue" and legislative facts as "facts about industry practices, economic impact, scientific data, and other information about which the parties have no special information."

UCS' argument proves too much. If the facts to be considered in a certification proceeding are wholly adjudicative, then, because those facts are like the facts considered in any rulemaking on safety issues, every such rulemaking must be a formal adjudication. However, this conclusion is clearly not the law; therefore, the facts in a certification proceeding are not wholly adjudicatory. Moreover, if such facts must be categorized at all, they are more "legislative" than "adjudicative", as UCS defines those terms, for while they are "related to activities of the parties", they are not uniquely so, and they are facts about "industry practices, scientific data", engineering principles, and the like.

Several commenters also argue that the certification proceeding should be a formal adjudication because cross-examination is an unsurpassed means for discovering the truth. Again, the argument proves too much, namely, that every rulemaking, indeed every species of lawmaking, should be formal adjudication. Part 52 does not assume the superiority, or even the usefulness, of formal procedures for resolving every issue; but it does provide for their use where they are the only means available for resolving an issue properly.

g. Fees for Review of Applications

The final rule adheres to the fee policy embodied in the proposed rule. An applicant for design certification does not have to pay an application fee, but the applicant will have to pay the full cost of the NRC review of the application, although not until the certification is referenced in an application for a construction permit or combined license, or, failing that, not until the certification expires. The details of the scheme of deferral of the fees appear in conforming amendments to the recently amended 10 CFR Part 170 (53 FR 52632; December 29, 1988).

UCS asserts that the provision for deferral of fees for NRC review is "unconscionable". To the contrary, the Commission believes that there is nothing "unconscionable" about deferral of fees for a program whose aim is to enhance safety.

Some industry commenters assert that the requirement for payment of the full cost of NRC review presents an "insurmountable disincentive" to the development of certified designs. Some industry commenters propose putting a ceiling on fees for certification review, in order to help vendors better estimate the costs of developing and certifying a design. The Commission fully recognizes that it will be difficult for a vendor to estimate the costs of taking a design through to certification. However, a ceiling on fees only displaces the burden of that uncertainty from the vendor to the public. In recent years, the NRC has been obliged by statute to charge fees which return to the Federal Treasury a portion of the costs incurred in regulation. Deferral of fees is more in line with the policies behind those statutes than is putting the burden of uncertainty on the public.

h. Finality

Standardization has the double aim of enhancing safety and making it possible to resolve design issues before construction. Of these two aims, enhanced safety is the chief, because pre-construction resolution of design issues could be achieved simply through combined construction permits and operating licenses with conditions. Achievement of the enhanced safety which standardization makes possible will be frustrated if too frequent changes to either a certified design or the plants referencing it are permitted.

The proposed rule put forward principally three means of preventing a continual regression from standardization. First, the proposed rule required that any amendment proffered by the "holder" of a certification be considered in a notice and comment rulemaking and granted if the amendment complied with the Atomic Energy Act and the Commission's regulations. Second, the proposed rule prohibited the licensee of a plant built according to a certified design from making any change to any part of the plant which was described in the certification unless the licensee had been granted an exemption under 10 CFR 50.12 from the rule certifying the design. Third, the proposed rule stated that the Commission would not backfit a certified design or the plants built according to it unless a backfit were necessary to assure compliance with the applicable regulations or to assure adequate protection of public health and safety. See § 52.63 of the proposed rule, 53 FR 32074, col. 3, to 32075, col. 2. The Commission invited comment on whether the amendment and exemption standards were stringent enough, and on whether the backfitting standard gave certifications a reasonable degree of finality. See 53 FR 32067, col. 2.

The comments focus on the standard of amending the certification, one group of comments wanting to make it harder for the "holder" of a certification to get an amendment, and another group wanting to make it easier. Several commenters say that the proposed rule wrongly makes it easier for the designer to amend the certified design than it is for the Commission to backfit the design. To correct this perceived imbalance, UCS, among others, proposes that no amendment be granted unless it constitutes a safety enhancement, and that any amendment granted be backfitted on all plants built according to the design being amended. OCRE proposes that, at a minimum, no amendment should be granted which would entail a decrease in safety. On the other side, NUMARC proposes virtually the same standard as a maximum: Any amendment which has no safety impact should be granted. DOE in effect argues that the Commission does not have authority to ask for more than OCRE's minimum, because this type of amendment would be proposed for economic, plant efficiency, or other business reasons and the NRC has no expertise or authority in areas involving business judgments. The law firm of Bishop, Cook, Purcell, and Reynolds, representing several utilities, proposes a backfitting standard more stringent than the one in the proposed rule: The Commission should not impose backfits on a design for the sake of compliance with applicable regulations unless the lack of compliance has an adverse impact on safety. Going even further in the same vein, the U.S. Chamber of Commerce proposes that even where the lack of compliance has an adverse impact on safety, the backfit should have to pass muster under a cross-benefit analysis.

The final rule places a designer on the same footing as the Commission or any other interested member of the public. No matter who proposes it, a change will not be made to a design certification while it is in effect unless the change is necessary to bring the certification into compliance with Commission regulations applicable and in effect when the certification was issued, or to assure adequate protection of public health and safety. See § 52.63(a)(1). Thus, the final rule cannot be said to make it easier for a designer to amend a certification then for the Commission to backfit the design. But more important, the final rule thus provides greater assurance that standardization and the concomitant safety benefits will be preserved.

The Commission is not adopting Bishop, Cook's suggestion that compliance be required only when non-compliance would have an adverse impact on safety. Licensees seeking relief from a design certification, who believe that non-compliance would have no adverse impact on safety, should request an exemption under 10 CFR 50.12. Neither is the Commission adopting the suggestion of the U.S. Chamber of Commerce that cost-benefit analysis be used to determine whether to impose backfits on designs to bring them into compliance with applicable regulations. The Atomic Energy Act allows the Commission to consider costs only in deciding whether to establish or whether to enforce through backfitting safety requirements that are not necessary to provide adequate protection. See *UCS* v. *NRC*, 824 F.2d 108, 120 (1987).

The final rule, like the proposed rule, permits applicants for combined licenses issued under the rule, and licensees of a plant built according to a certified design, to request an exemption under 10 CFR 50.12 from a rule certifying a design. Among the comments on the appropriateness of using § 50.12 in the standardization context were NIRS' comment that § 50.12 permitted exemptions at a "whim" and DOE's suggestion that no exemptions should be granted at all. Out of respect for the unforeseen, the Commission has decided to adhere to § 50.12, but the final rule does require that, before an exemption can be granted, the effect which the exemption might have on standardization and its safety benefits must be considered.

As a further guard against a loss of standardization, the final rule, again like the proposed rule, also prohibits a licensee of a plant built according to a certified design from making any change to any part of the plant which is described in the certification unless the licensee has been granted an exemption under 10 CFR 50.12 from the rule certifying the design. Because the certification is a rule, 10 CFR 50.12, not 50.59, is the standard for determining whether the licensee may make changes to the certified portion of the design of the plant without prior approval from the NRC. NUMARC says that, given the practicalities of construction and the limited resources of the NRC staff, licensees need the flexibility afforded by § 50.59. However, the Commission believes that the certifications themselves and § 50.12 will provide the necessary flexibility with respect to the certified portion of the plant (or at least as much flexibility as is consistent with achieving the safety benefits of standardization), while § 50.59 will continue to apply to the uncertified portion. How much flexibility § 50.12 will provide depends in large part on how much detail is present in a design certification, and just how much is present will be an issue which will have to be resolved in each certification rulemaking. The Commission does expect, however, that there will be less detail in a certification than in an application for certification, and that a rule certifying a design is likely to encompass roughly the same design features that § 50.59 prohibits changing without prior NRC approval. Moreover, the level of design detail in certifications should afford licensees an opportunity to take advantage of improvements in equipment.

The comments on the proposed rule raise two other important finality issues, both connected with backfitting. The first bears on the criteria for renewal of a design certification. The proposed rule provided that the Commission would grant a request for renewal of a design certification if the design complied with regulations in effect at renewal and any more stringent safety requirements which would bring about a substantial increase in safety at a cost justified by the increase (strictly speaking, the backfit rule would not apply at renewal, but the proposal nonetheless incorporated the backfit rule's cost-benefit standards). See § 52.59(a), 53 FR 32074, col. 3. Bishop, Cook, among others, proposes that the standard for renewal be compliance with regulations in effect not at renewal but rather at the time the certification was originally issued, together with any other more stringent requirements which are justified under the backfit rule. The proposed rule's criteria were in fact equivalent to Bishop, Cook's in their impact on a given design certification, but they differed in their impact on the timing of some backfit analyses, the proposed rule providing that some would be done in rulemakings while the given certification was in effect. However, the final rule adopts Bishop, Cook's proposal because it more clearly says that imposition of more stringent requirements on a design during a renewal proceeding will be governed by backfit standards.

The second of the other important finality issues raised by the comments concerns the finality of 10 CFR Part 52, Appendix O (formerly in Part 50) final design approvals (FDAs) already in effect on the effective date of this rule. Section 52.47(a)(2) of the proposed rule stated that holders of FDAs in effect on the effective date of the rule might have to submit more information to the staff in connection with the review for certification. NUMARC proposes adding a "grandfather" clause which would prohibit the Commission from imposing, during the certification proceeding, any change on that part of the design which is covered by an already effective FDA unless the change meets the criteria of the backfit rule.

Adoption of NUMARC's proposal would not only entail a significant change in the force of an FDA, it would also extend the range of application of the backfit rule. Under existing NRC regulations, an FDA binds the staff in a licensing proceeding but not in a certification proceeding; and even in a licensing proceeding, the staff may, on the grounds of significant new information or other good cause, reconsider an earlier determination. See 10 CFR Part 52, Appendix O, paragraph 5. Moreover, the FDA does not bind the Commission or the Commission's adjudicatory panels. Id. at paragraph 6. The backfit rule applies to any proposal which would require the holder of an FDA to meet a new standard in order to remain in possession of the FDA, see 10 CFR 50.109(a)(1), but the backfit rule does not change the force an FDA has in a licensing proceeding or certification proceeding.

NUMARC's proposal, however, would bind both the staff and the Commission in a certification proceeding and would add a cost-benefit test to the tests which must be met before a determination made in an FDA could be reconsidered. NUMARC's proposal thus would effectively amend both the backfit rule and the cited paragraphs of Appendix O: It would, in effect, turn any existing FDA into a partial certification. Here the Commission would rather adhere to the finality provisions in the existing regulations, including Appendix O and the backfit rule. The Commission believes that, in this situation, these provisions adequately balance the need for finality with the need for flexibility to deal with unforeseen safety advances or risks.

2. Early Site Permits

What design certification is to the early resolution of design issues, the early site permit is to the early resolution of site-related issues. Both the certification and the permit make it possible to resolve important licensing issues before a construction permit proceeding. They in effect make possible the banking of designs and sites, thereby making the licensing of a given plant more efficient. However, some commenters question whether the Commission should issue early site permits. The Attorney General of New York, for instance, sees no need for early site permits and questions whether there could be grounds adequate to support approval of a site for twenty years, the term of early site permits under the proposed rule (the final rule provides that permits will have terms of between ten and twenty years). He points out that under the NRC's current regulations, NRC early decisions on site suitability issues raised in connection with a construction permit generally remain effective for only five years. See 10 CFR 2.606 and 10 CFR Part 52, Appendix Q (formerly in Part 50), paragraph 5. The Connecticut Siting Council strongly suggests that the State of Connecticut would be unable to participate in an NRC hearing on an application for an early site permit unless the application proposed a "specific" nuclear power plant. Finally, one commenter is concerned that land approved under an early site permit might never be used for a nuclear power plant, and thus development of the land for a non-nuclear use would have been needlessly delayed.

The Commission believes that early site permits can usefully serve as vehicles for resolving most site issues before large commitments of resources are made. Moreover, the Commission believes that a term of ten to twenty years for early site permits will make early site permits more useful for early resolution of site issues than would the five-year term in 10 CFR 2.606 and 10 CFR Part 52, App. Q, because the longer term will require less frequent reassessments of issues than would the shorter term. The five-year term is a function not of the reliability of the information available to make the decisions, but rather of the fact that the decisions made under those provisions may only resolve isolated site issues [8] and anticipate site utilization in the very near term. The Commission is confident that there will be information adequate to support site approvals lasting up to 20 years. After all, the Commission licenses plants and their sites for operation for periods of up to twice twenty years. Where adequate information is not available, early site permits will not be issued.

The Commission is also confident that enough information on reactor design will be available in an early site permit proceeding to permit sound judgments about environmental impacts and thus to enable state and local agencies such as the Connecticut Siting Council to participate effectively in an early site permit proceeding. The Council says that for it to meaningfully participate in a decision on an application for an early site permit, the application would have to contain "projected emission, discharges, site impacts, safety factors, and exact operational parameters * * * proposed for a site". It is just such information which both the proposed rule and the final rule would require of applicants for early site permits. See § 52.17(a).

Last, although the Commission acknowledges the possibility that non-nuclear development of a site would be postponed when a site is reserved for a nuclear plant and then a plant never built there, the Commission believes that such a possibility does not loom very large. Persons are not likely to go to the expense of applying for an early site permit unless there is a good prospect that the site will be used for a nuclear power plant. Moreover, it may be that many of the sites for which early site permits might be sought are already set aside for use by utilities; thus, even though non-nuclear development of the site might be postponed, non-utility uses of the site would not be. Last, even during the period in which an early site permit is in effect, non-nuclear uses of the site are not prohibited altogether. See § 52.35.

The comments on the proposed rule raise two other important issues concerning the rule's provisions on early site permits. The first issue concerns the division of authority between the Federal government and local governments over the siting of nuclear power facilities. The New York State Energy Office is concerned that the proposed rule leaves the impression that only an early site permit from the NRC is necessary to set aside land for a nuclear power plant. To the contrary, the rule does not, indeed, could not, change the division of authority between the Federal government and the states over the siting of nuclear plants. An early site permit constitutes approval of a site only under the Federal statutes and regulations administered by the Commission, not under any other applicable laws.

The last important issue raised by the comments on early site permits concerns the proposed rule's requirement that the application contain a plan for redress of the site in the event that the site preparation work and similar work ~~and similar work~~ allowed by 10 CFR 50.10(e)(1) is performed and the site permit expires before it is referenced in an application for a construction permit or combined license issued under the rule. The proposed rule required that the plan provide reasonable assurance that redress carried out under the plan would achieve a "self-maintaining, environmentally stable, and aesthetically acceptable site" which conformed to local zoning laws. The only important difference between the proposed and final rules on this subject is that the final rule requires such a plan only of applicants who wish to perform

[8] Thus, the Commission declines to follow the suggestion of the engineering firm of Stone & Webster that partial early site permits be issued. It is not likely that resolutions of isolated site issues could have the degree of finality which a permit lasting ten to twenty years must have.

the activities allowed by 10 CFR 50.10(e)(1). NUMARC says that this requirement is "inherently unworkable" and would involve the Commission in matching redress against a variety of local zoning laws.

To the contrary, the rule's provisions on site redress, including the provision on zoning, are modeled on the redress requirements imposed on the Clinch River Breeder Reactor project. See In the Matter of the U.S. Department of Energy, et al. (Clinch River Breeder Reactor Plant), LBP-83-7, 21 NRC 507 (1983). Moreover, the Commission has long required that applicants' environmental reports discuss compliance with local laws, including zoning laws. See 10 CFR 51.45(d). Apparently, NUMARC is not opposed to redress per se, for NUMARC's proposed revision of § 52.25 of the proposed rule speaks of the possibility that redress of adverse environmental impacts might be necessary. The Commission is only requiring that such redress follow the precedent established at Clinch River and proceed according to a plan incorporated in the early site permit. Containing a redress plan, the permit itself will constitute assurance that, if site preparation activities are carried out but the site never used for a nuclear power plant, the site will not be left in an unacceptable condition.

3. Combined Licenses

a. The Commission's Authority to Issue Combined Licenses

There are two important questions in connection with the proposed rule's provisions on combined construction permits and operating licenses with conditions. The first is whether the Commission has the authority to issue combined licenses. The second is whether, in cases where all design issues are resolved before construction begins, there should be a hearing after construction is complete, and if so, what issues should be considered at the hearing.

Comments on whether the Commission has the authority to issue combined licenses tend to mirror the commenters' views on what kind of hearing should be held after construction is complete. In other words, the discussion of this issue tends to be result-oriented. Thus, many who believe that there should be a hearing after construction, and that it should be as full a hearing as operating license hearings often are, argue that the Commission has no authority to issue combined licenses. They claim that section 185 of the Atomic Energy Act mandates a two-step licensing process (for the text of section 185, see below). They often cite *Power Reactor Development Co. v. International Union of Electrical Workers*, 367 U.S. 396 (1961) as support for this interpretation of section 185. To these arguments, those who believe that there should be no hearing, or else only a highly restricted hearing, after construction is complete reply that section 161h of the Atomic Energy Act gives the Commission authority to combine a construction permit and an operating license in a single license (for the text of section 161h, see below).

A closer look at section 161h and 185 shows that section 161h clearly gives the Commission authority to combine a construction permit and operating license in a single license and that section 185 is not inconsistent with section 161h. Section 161h says, in pertinent part, that the Commission has the authority to "consider in a single application one or more of the activities for which a license is required by this Act [and] combine in a single license one or more of such activities . . ." 42 U.S.C. 2201. The plain language of this section clearly applies to the combining of construction permits and operating licenses, for both construction and operation of nuclear power facilities are "activities for which a license is required by this Act", namely by sections 101 and 185 of the Act, see 42 U.S.C. 2231 and 2235, and section 103a of the Act makes any license to operate a commercial nuclear power facility "subject to such conditions as the Commission may by rule or regulation establish . . ." See 42 U.S.C. 2233. Had Congress intended that construction permits and operating licenses for commercial nuclear power plants be excluded from the language of section 161h, surely Congress would have said so right in that section, for the plain language of that section invites their inclusion, and they are the most important licenses issued under the Act.

Section 185 is not to the contrary. Section 185 says, in pertinent part,

> CONSTRUCTION PERMITS.—All applicants for licenses to construct . . . utilization facilities shall . . . be initially granted a construction permit. . . . Upon the completion of the construction . . . of the facility, upon the filing of any additional information needed to bring the original application up to date, and upon finding that the facility authorized has been constructed and will operate in conformity with the application as amended and in conformity with the provisions of this Act and of the rules and regulations of the Commission, and in the absence of any good cause being shown to the Commission why the granting of a license would not be in accordance with the provisions of this Act, the Commission shall thereupon issue a license to the applicant. . . .

42 U.S.C. 2235. To be sure, the section speaks in terms of a construction permit's being issued first, and then a license (presumably an operating license). However, the contrast between the two licenses is not fundamental to the section. The substance of the section is clearly indicated by the title of the section and by the list of findings the Commission must make. The section may be paraphrased thus: A construction permit is not a grant of authority to operate once construction is complete; before operation begins, the original application must be brought up to date, and the Commission must make certain affirmative findings. Thus the critical matter is not the separation of the two licenses, but the need for specific findings before operation. With this substance, both the proposed rule and the final rule are entirely in accord (the pertinent provisions of the final rule will be described in more detail below).

Moreover, in differentiating between a "construction permit" and a later "license", section 185 is not taking exception to section 161h. Section 185 does not say, for instance, "Notwithstanding anything in section 161h to the contrary, applicants shall be granted initially only a construction permit." By speaking of a separate issuance of a license after completion of construction, section 185 simply conforms itself to the simplest case, in which the licenses are in their elementary, uncombined states, and avoids having to make an already long section longer in order to acknowledge the case which section 161h makes possible. Moreover, section 185 acknowledges section 161h implicitly when it speaks not of a separate application for an operating license but simply of an updating of the original application. Therefore, neither the proposed rule nor the final rule can be faulted for not providing for a separate issuance of an operating license.

This interpretation of section 185 is confirmed by the legislative history of the section. In 1954, when Congress was considering proposed amendments to the Atomic Energy Act of 1946, representatives of the industry complained that the proposed section 185 required that construction of a facility be completed "under a mere construction permit, without any assurance at that stage that there will be issued any license to . . . operate it after it has met all the specifications of the construction permit." Atomic Energy Act of 1954: Hearings on S. 3323 and H.R. 8862 before the Joint Committee on

Atomic Energy, 83rd Congress, 2d Session, 113 (May 10, 1954). These representatives proposed instead that power facility applicants should be able to obtain a single license covering all aspects of their activities—construction, possession of fuel, and operation—and that the license should contain the conditions the applicant would have to meet before operation of a constructed facility could begin. Id. at 113 and 118. On this proposal, the following colloquy took place:

Representative HINSHAW. That seems to me to be reasonable, that you should put all the conditions into 1 license that can be put into 1 license. That would be fair enough.

Chairman COLE. Would you mind my interruption? Why cannot that be done under the terms of the bill as it is now?

Mr. McQUILLEN [representing Detroit Edison]. I think it undoubtedly would be so operated.

Chairman COLE. Of course it would.

Id. at 119. Chairman Cole said this even though neither of the draft bills before the Committee contained the text of what is now section 161h. Twelve days later, as if to put the matter beyond all doubt, the Committee incorporated the present text of section 161h into both bills. The final rule provides for just such a single license, with conditions, as was discussed in this colloquy.

Power Reactor Development Co. v. *Electrical Workers*, 367 U.S. 396 (1961), is not to the contrary. The issue in that case was not whether the Commission had the authority to combine a construction permit with an operating license with conditions, but whether the Commission could postpone the ultimate safety findings until construction was complete. The Court ruled that the Commission could, and found support for its conclusion in section 185, which showed, the Court said, that "Congress contemplated a step-by-step procedure." 367 U.S. at 405. But the Court did not say, "section 185 mandates a separate issuance of an operating license, notwithstanding section 161h." The interpretation of section 161h of the Act was not at issue.

b. Hearings After Construction Is Complete

The first issue concerning hearings after completion of construction under a combined license is whether there should be such hearings at all. Most commenters, whatever their affiliation, believe that there should be the opportunity for such hearings. They disagree only over how limited the hearings should be. DOE argues that there should be no such hearings at all. As the principal support for its argument, DOE cites the section of the Administrative Procedure Act (APA) which says, in effect, that adjudication is not required in cases in which the agency decision rests "solely on inspections, tests, or elections". See 5 U.S.C. 554(a)(3). Under Part 52's provisions of combined licenses, a combined license will contain the tests, inspection, and analyses, and acceptance criteria therefor, which are necessary and sufficient to provide reasonable assurance that the facility has been constructed and will operate in conformity with the license and the Act. See § 52.97. DOE's argument amounts to the claim that the kind of tests and inspections spoken of in Part 52 is the same as the kind of tests and inspections spoken of in the APA.

The Commission agrees that findings which rest solely on the results of tests and inspections should not be adjudicated, and the final rule so provides. See § 52.103. However, not every finding the Commission must make before operation begins under a combined license will necessarily always be based on wholly self-implementing acceptance criteria and therefore encompassed within the APA exception. The Commission does not believe that it is prudent to decide now, before the Commission has even once gone through the process of judging whether a plant built under a combined license is ready to operate, that every finding the Commission will have to make at that point will be cut-and-dried—proceeding according to highly detailed "objective criteria" entailing little judgment and discretion in their application, and not involving questions of "credibility, conflicts, and sufficiency", questions which the Court in *UCS* v. *NRC*, 735 F.2d 1437 (D.C. Cir. 1984), held were marks of issues which should be litigated at least under the facts of that case. Indeed, trying to assure that the tests, inspections, and related acceptance criteria in the combined license are wholly self-implementing may well only succeed in introducing inordinate delay into the hearing on the application for a combined license.

Thus, the question becomes whether the rule should provide an opportunity for a post-construction hearing on the issues which are not excepted from adjudication by the APA. Whether the Commission could or should go further under its governing statutes we leave to future consideration and experience; this rule adopts an approach within the bounds of our legal authority which sets reasonable limits on any post-construction hearing. In this regard, every commenter who believes there should be such an opportunity for hearing also believes that an issue in the hearing should be whether construction has been completed in accord with the terms of the combined license, and the final rule so provides. Also, under section 185 of the Atomic Energy Act, the Commission must find, prior to facility operation, that the facility has been constructed and will operate in conformity with the application and the rules and regulations of the Commission. This statutory finding, in the context of Subpart C of this rule, translates into two separate but related regulatory findings: that compliance with the acceptance criteria in the combined license will provide reasonable assurance that the facility has been constructed and will operate in accordance with the Commission's requirements, and that the acceptance criteria have in fact been satisfied. The former finding will be made prior to issuance of the combined license, and will necessarily be the subject of any combined license hearing under section 189a of the Act. The latter finding cannot by its nature be made until later, after construction is substantially complete, and therefore cannot by its nature be the subject of any hearing prior to issuance of the combined license. Thus, to the extent that an opportunity for hearing should be afforded prior to operation, it should be confined to the single issue that cannot have been litigated earlier—whether the acceptance criteria are satisfied. No commenter has offered any legal argument to the contrary.[4]

Commenters disagree greatly on whether any other issue should be considered in a hearing. The proposed rule provided that intervenors could contend that significant new information showed that some modification to the site or the design was necessary to assure adequate protection. To this, NUMARC responds that "no one could seriously consider ordering a new plant with the licensing uncertainties it would face." NUMARC proposes a complete rewrite of § 52.103, elements of which are discussed below. Several industry commenters point to the "added burdens" that applicants would be assuming under the proposed rule as grounds for severely limiting the issues for hearing. Rockwell International, for instance, claims that, with the hearing

[4] Section 185 also says that, prior to operation, there must be an "absence of good cause being shown to the Commission why the granting of the license would not be in accordance with the provisions of the Act." We think that this implicit opportunity to show "good cause" is satisfied by affording an opportunity for hearing on all findings that will be made prior to facility operation.

under § 52.103, there will be four public hearings for each plant.

Public interest groups also take a dim view of the proposed rule's limitations on the hearing, though their reasons are not the industry's. UCS says that a licensing proceeding without uncertainty is a sham. OCRE goes further and asserts that the uncertainty should be distributed equally: "In a perfectly fair proceeding, [the] chance [of winning] would be 50%." The Maryland Nuclear Safety Coalition counts only two hearings for each plant. NIRS says that many problems with the current generation of reactors were cured under the full two-step licensing process.

This latter group of commenters appears to be opposed to any limitation on the post-construction hearing, for not one of them proposes a concrete alternative to the proposed rule's provisions on the hearing. UCS does say that the hearing should encompass "all issues that are material to the NRC's approval of an operating license for the plant", but that statement is either so general as to be just another way to put the question of what issues should be encompassed, or it is the claim that, when it comes time to determine whether the plant has been built in conformity with the terms of the combined license, all the operating license issues resolved before construction should be treated as if they had never been resolved. Many commenters do in fact seem to be making such a claim, for they contend against any limits on the post-construction hearing at the same time that they support the idea that design issues should be resolved before construction.

There have to be substantial limits on the issues that can be raised after construction. A licensing proceeding without any uncertainty in result may be a sham, but the bulk of the uncertainty should be addressed and resolved prior to, not after, construction. Part 52 does not remove uncertainty, it simply reallocates it to the beginning of the licensing process. The alternative apparently offered by opponents of limits on the post-construction hearing is, in effect, to double the uncertainty by considering every design issue twice.[8] To the extent that these commenters offer any practical arguments in favor of this approach, they are not persuasive. Rockwell International may engage in some double-counting when it asserts that there are four public hearings for each plant, but when the Maryland Nuclear Safety Coalition says that the public can debate licensing issues only in an early site permit hearing and after construction, and therefore needs another hearing on design issues, it inexplicably simply ignores the mandatory public hearing on the application for the combined license and the opportunity for a public hearing on an application for a design certification. Moreover, contrary to NIRS, shortcomings in certain plants were not discovered because the licensing proceedings consisted of two steps but rather because design issues had to be resolved and construction made to conform to design before operation began. Part 52 provides for no less.

The final rule adopts a straightforward approach to limiting the issues in any post-construction hearing on a combined license. As a matter of logic, every conceivable contention which could be raised at that stage would necessarily take one of two general forms. It would allege either that construction had not been completed—and the plant would not operate—in conformity with the terms of the combined license, or that those terms were themselves not in conformity with the Atomic Energy Act and pertinent Commission requirements. The final rule makes issues of conformity with the terms of the combined license part of any post-construction hearing, unless those issues are excepted from adjudication by the APA exception for findings which are based solely on the results of tests and inspections. The final rule does not attempt to say in advance what issues might fall under that exception. The comments are nearly unanimous in the opinion that issues of conformity with the combined license are properly encompassed in any post-construction hearing. Moreover, this limited opportunity for hearing is consistent with the Commission's belief that, even if section 185 did not speak at all to the need for a conformity finding, the Commission itself would need to make such a finding prior to operation in order to conclude, in the language of section 103, that operation is not inimical to the health and safety of the public. The final rule also provides that issues of whether the terms of the combined license are themselves inadequate are to be brought before the Commission under the provisions of 10 CFR 2.206. This approach to issues concerning the inadequacy of the combined license is well-founded in the discretion afforded the Commission under section 185 of the Act to determine what constitutes "good cause" for not permitting operation, and in the analogy which this approach has with the way construction permits are treated in operating license proceedings. Contentions alleging inadequacies in a construction permit are not now admissible in an operating license proceeding. Similarly, under the final rule, contentions alleging inadequacies in a combined license are not admissible in a post-construction hearing. Moreover, as we noted, this approach fully satisfies applicable law.

III. Other Issues

These are taken up section by section. Not discussed are most of the many changes made to the proposed rule for the sake of clarity, brevity, consistency, specificity, and the like. Worth noting, however, is that this Federal Register notice moves Appendices M, N, O, and Q of Part 50 to Part 52, so that, except for Subpart F of 10 CFR Part 2, all of the Commission's regulations on standardization and early resolution of licensing issues will be in one part of 10 CFR Chapter I. Readers are reminded that a comparative text showing all deletions from, and additions to, the proposed rule is available in the NRC's public document room.

1. Early Site Permits

At the suggestion of NUMARC and others, § 52.17 now gives applicants for early site permits the option of submitting partial or complete emergency plans, for final approval. Also, the section requires a redress plan only of applicants who wish to be able to perform the site preparation work and similar work allowed under 10 CFR 50.10(e)(1). Last, incorporating suggestions by UCS and others, the section says what factors should be considered in determining whether the area surrounding the site is "amenable" to emergency planning. To avoid suggesting that the Commission is adopting new emergency planning standards, § 52.17 abandons the proposed language of "amenability to emergency planning" in favor of language drawn from existing regulations on emergency planning.

Section 52.18 now makes clear that need for power is not a consideration at the early site permit stage.

In a number of places—§§ 52.23, 52.53, 52.87, and portions of other sections—the rule provides explicitly for ACRS

[8] Even according to OCRE's notion of a "perfectly fair" proceeding, in which perfect fairness could be achieved by replacing judges with tosses of coins, design issues should not be resolved twice. If they were, intervenors would have two 50% chances to win—that is, to prevent operation of the plant—on design issues. But two even chances are equivalent to a 75% chance overall (e.g., the chance of coming up heads once in two tosses of a coin is 3 out of 4), and a proceeding in which one party has a 75% chance of winning is not, according to OCRE, "perfectly fair".

review of issues to make clear that, even though the Atomic Energy Act does not, in terms, give the ACRS a role in the granting of early site permits, design certifications, or combined licenses, the ACRS is to have the same role with respect to these devices that it does with respect to construction permits, operating licenses, and the like. Wherever the ACRS is spoken of in Part 52, the intention is that the ACRS review the pertinent issues according to the standards specified therein.

As in the proposed rule, § 52.25 provides that the holder of an early site permit which contains a site redress plan, or the applicant for a construction permit or combined license which references such an early site permit, may perform the activities at the site allowed by 10 CFR 50.10(e)(1) without first obtaining the separate authorization required by § 50.10. The New York State Energy Office appears to take this to mean that the holder of the permit may perform the work without NRC approval. To the contrary, the early site permit which contains a redress plan is itself NRC approval. The law firm of LeBoeuf, Lamb, Leiby & MacRae, representing several utilities, argues that recent case law, especially NRDC v. EPA, 859 F.2d 156 (D.C. Cir. 1988), calls into question the Commission's limitations on non-safety related construction before issuance of a permit. LeBoeuf, Lamb concludes that § 52.25 and related portions of Part 52 should be deleted and the limitations in § 50.10 reviewed in the light of the case law. The Office of the General Counsel is undertaking a review and will recommend to the Commission if any changes to these sections are warranted. In the meantime, the Commission has decided to keep Part 52's provisions on site work intact and consistent with the related provisions in Part 50.

Section 52.27 now contains some of the material which appeared in § 52.29 of the proposed rule. OCRE objects to the provision in § 52.27 which treats an early site permit as valid beyond the date of expiration in proceedings based on applications which have referenced the early site permit. OCRE argues that this provision allows clever applicants to avoid new site requirements by referencing an early site permit just before it expir[illegible] bottom, this is really an argument that early site permits should have shorter durations. The Commission is confident that the [illegible]ency will be able to make site [illegible]s which will retain their [illegible]er the durations provided for in [illegible] rule. However, the final rule [illegible]rovide that the duration of an original permit can be fixed at a term shorter than twenty years. See § 52.27(a).

In its comment on § 52.31, LeBoeuf, Lamb suggests that at renewal, the burden should be on the Commission to show why an early site permit should not be renewed, but that a given permit should be renewed only once, and for not more than ten years. The final rule retains the provisions of the proposed rule, because they provide more flexibility to both the Commission and holders of permits.

Much of the discussion in Sections II.1.f. and II.3.b. above on the finality of design certifications and hearings after construction is relevant to the provisions in § 52.39 on the finality of early site permits. Section 52.39 now states that, except in certain limited circumstances, issues resolved in a proceeding on an early site permit shall be treated as resolved in any later proceeding on an application which references the early site permit. One of the circumstances involves petitions under 10 CFR 2.206 that the terms of the early site permit should be modified; § 52.39(a)(2)(iii) assumes that the Commission shall resolve the issues raised by the petition in accordance with the standard in paragraph (a)(1) of the same section.

2. Design Certifications

In the proposed rule, § 52.45 contained material on scope of design and testing of prototypes. This material now appears, in modified form, in § 52.47. The phrase "essentially complete nuclear power plant," which is used in 52.45, is defined as a design which includes all structures, systems, and components which can affect safe operation of the plant except for site-specific elements such as the service water intake structure and the ultimate heat sink. Therefore, those portions of the design that are either site specific (such as the service water intake structure or the ultimate heat sink) or include structures, systems and components which do not affect the safe operation of the facility (such as warehouses and sewage treatment facilities) may be excluded from the scope of design. In addition, an essentially complete design is a design that has been finalized to the point that procurement specifications and construction and installation specifications can be completed and made available for audit if it is determined that they are required for Commission review in accordance with the requirements of § 52.47(a). Procurement specifications would have to identify the equipment and material performance requirements and include the necessary codes, standards, and other acceptance and performance criteria to which the equipment and materials will be fabricated and tested. Construction and installation specifications would have to identify the criteria and methods by which systems, structures and components are erected or installed in the facility and include acceptance, performance, inspection, and testing requirements and criteria.

In § 52.47, the provisions on testing of prototypes have been reworded to avoid suggesting a presumption that designs of the affected class could be certified only after successful testing of a prototype. One individual and the U.S. Metric Association urged that the r[illegible] require that technical information in applications be in metric units. T[illegible] NRC staff believes there is much merit in [illegible]is proposal, but because the public has no[illegible] had an opportunity to comment on it, it is not incorporated in the final rule. The NRC staff is considering proposing an amendment to Part 52 on the subject for Commission review.

On §§ 52.53, 52.55, and 52.63, see the remarks in Section III.1. above on §§ 52.23, 52.27, and 52.39, respectively. Also, § [illegible] of the proposed rule set ten years as the [illegible]uration of certifications. The fi[illegible] [illegible]xtends the duration to fifteen years, t[illegible] permit more operating experience with a given design to accumulate [illegible] the certification comes up for re[illegible]wal or ceases to be available to applicants for combined licenses. In addition, § 52.63(a)(3) now limits Commission-ordered modifications of design-certified elements of a specific plant to situations in which the modification is necessary for adequate protection and special circumstances as defined in 10 CFR 50.12(a) are present. This double requirement does not mean that if a specific plant presents an undue risk but no special circumstances are present the plant will not be modified. Rather, the modification will take place through modification of the certified design itself, as provided for elsewhere in the same section.

Theoretically, it would be possible for an applicant whose application referenced a certified design to select designer(s) other than the designer(s) which had achieved certification of the standard design. Section 52.63(c) makes clear that such an applicant might be required to provide information which is normally contained in procurement specifications and construction and installation specifications and which is consistent with the certified design and available for audit by the NRC staff.

Also, § 52.73 requires a demonstration that the new designer is qualified to supply the design. Last, the new designer would have to pay a portion of the cost of the review of the application for certification. See 10 CFR 170.12(d) and (e), as amended in this document. It is expected, as a practical matter, that applicants referencing a certified design would select the designer which had achieved certification of the standard design.

3. Combined Licenses

Section 52.73 now provides that the entity that obtained certification for a design must be the entity that supplies the design to an applicant for a combined lic[illegible] referencing the design, unless it is demonstrated that another entity is qualified to supply the design. This provision was added because an entity supplying the design should be qualified to do so; the entity which obtained the certification will have demonstrated its qualifications by obtaining the certification.

The last sentence of § 52.75 of the proposed rule now appears in § 52.79 of the final rule.

DOE proposes redrafting § 52.79 to require that no application for a combined license be considered unless it references a certified design. The final rule does not contain this restriction because there may be circumstances in which a combined license would properly utilize a non-standard design, and because such a restriction would mean, among other things, that every prototype would have to be licensed in a fully two-step process. In connection with § 52.79's provisions on submission of complete emergency plans, NIRS somehow concludes that Subpart C's provisions on emergency planning "extend", to the detriment of state and local governments, the "realism" doctrine set forth in 10 CFR 50.47 and recently affirmed in Commonwealth of Massachusetts v. NRC, 856 F.2d 378 (1st Cir. 1988). Apparently, NIRS believes that to settle emeregency planning issues before construction is to "extend" the doctrine. To the contrary, although Subpart C assumes the "realism" doctrine, as it is entitled to do, it does not extend it. The doctrine remains precisely what it is in § 50.47. Moreover, the Commission's aim in drafting Subpart C's provisions on emergency planning has been to follow to the maximum feasible extent the National Governors' Association's Recommendation, at its 79th annual meeting, in 1987, that ". . . emergency plans should be approved by the NRC before it issues the construction permit for any new nuclear power plant."

Section 52.83 now provides that the initial term of a combined license shall not exceed forty years from the date on which the Commission makes the findings required by § 52.103(c).

On § 52.87, see the discussion in Section III.1. on § 52.23.

NUMARC proposed removing from § 52.89 any reference to design certifications, on the grounds that environmental impact statements should not be prepared in connection with certification rulemakings. The references in this section to design certifications are not meant to imply that environmental impact statements must be prepared in connection with design certifications.

Section 52.99 has been reworded to reflect more clearly that the inspection carried out during construction under a combined license will be based on the tests, inspections, analyses, and related acceptance criteria proposed by the applicant, approved by the staff, and incorporated in the combined license. Several industry commenters proposed adding to this section a requirement that the staff prepare a review schedule in connection with each combined license. However, such a requirement would be largely duplicative of a long-standing staff practice under which the staff prepares an annual inspection plan which allocates resources according to the priorities among all pending inspection tasks. The annual plan should assure the timeliness of staff review of construction under a combined license. Section 52.99 envisions a "sign-as-you-go" process in which the staff signs off on inspection units and notice of the staff's sign-off is published in the Federal Register. UCS says that it is "totally inappropriate" for the Commission, while construction is going on, to sign off on inspections and thus put matters beyond dispute which might otherwise be raised after construction is complete. However, UCS has misunderstood the Commission's role in the inspection process. While construction is going on, only the staff signs off on inspections. The Commission makes no findings with respect to construction until construction is complete. Section 52.99 has been modified to make this point more clearly.

UCS and other commenters object to the section in § 52.103 of the proposed rule which provided interested persons thirty days after notice of proposed authorization of operation in which to request a hearing on the specified grounds. Yet the thirty-day requirement was drawn from section 189a of the Act. Neither the Act nor Part 52 imagines that it would be acceptable for interested persons to wait until notice is received before they examine the record of construction. These time periods are like the sixty-day limit in the Hobbs Act, 28 U.S.C. 2344, for petitions for direct judicial review of an agency rule. These limits assume that the petitioner is familiar with the fundamentals of the record before the limited period begins. The limited period is then provided for consideration of options, consultation with other interested persons, and drafting of pleadings. In any event, the final rule provides sixty days, in consideration of the pleading standard § 52.103 imposes on petitioners. Moreover, as noted above, to assist interested persons in becoming familiar with the construction record, § 52.99 now provides that notice of staff approvals of construction will be published periodically in the Federal Register. Any hearing held under § 52.103(b)(2)(1) will use informal procedures to the maximum extent practicable and permissible under law. In particular, the Commission intends to make use of the provisions in 5 U.S.C. 554, 556, and 557 which are applicable to determining applications for initial licenses. Under § 52.103(b)(2)(ii), the NRC staff will review the § 2.206 petition and make appropriate recommendations to the Commission concerning the petition. The Commission itself will issue a decision granting or denying the petition in whole or in part.

Finally, Urenco, Inc., is concerned that the last subsection of § 52.103 not be taken to suggest that the Commission would have to make separate findings for each of the numerous "modules" of a gaseous diffusion facility. The issue of how the modules of a gaseous diffusion facility should be licensed is beyond the scope of this rulemaking; § 52.103 therefore cannot suggest that the Commission would have to make separate findings for each of the modules of such a facility.

IV. Replicate Plant Concept

In the notice of proposed rulemaking, the Commission published a revised policy statement on replication of plants and invited comment on the revised policy. See 53 FR 32067, col. 3, to 32068, col. 1. Several industry commenters remarked that the statement's requirement that the application for replication be submitted within five years of the date of issuance of the staff safety evaluation report for the base plant effectively made replication unavailable for the short term. They recommended removing the restriction.

or at least lengthening it. The Commission has decided to retain this restriction. The five-year figure is in fact already a lengthening of the analogus figure in the immediately preceding version of the policy statement. The restriction is a reflection of the Commission's belief that applications which reach back further than a given number years probably ought to be considered as custom-plant applications.

Policy on Replication

The replicate plant concept involves an application by a utility for a license to construct or operate one or more nuclear power plants of essentially the same design as one already licensed.

The design of the plant already licensed (termed the base plant design) may be replicated at both the construction permit and operating license stages, and in applications for combined construction permits and operating licenses in a one-step licensing process. Replication of an approved base plant design at the construction permit stage is a prerequisite for its replication at the operating license stage. Although replication of the base plant design at the operating license stage is not mandatory, that is, the operating license application may be submitted as a custom plant application, it is strongly recommended.

An application for a replicate plant must demonstrate compliance with the four licensing requirements for new plant designs as set forth in the Commission's Severe Accident Policy Statement (50 FR 32138; August 8, 1985).

Each application proposing to replicate a previously licensed plant will be subjected to a qualification review to determine the acceptability of the base plant for replication and to define specific matters that must be addressed in the application for the replicate plant. A further requirement for qualification is that the application for a replicate plant must be submitted within five years of the date of issuance of the staff safety evaluation report for the base plant. The qualification review will consider the following information:

(1) The arrangement made with the developers of the base plant design for its replication;

(2) The compatibility of the base plant design with the characteristics of the site proposed for the replicate plant;

(3) A description of any changes to the base plant design, with justification for the changes;

(4) The status of any matters identified for the base plant design in the safety evaluation report, or subsequently identified by the ACRS or during the public hearings on the base plant application as requiring later resolution;

(5) Identification of the major contractors, with justification for the acceptability of any that are different than those used by the base plant applicant; and

(6) A discussion of how the replicate plant design will conform to any changes to the Commission's regulations which have become effective since the issuance of the license for the base plant.

Environmental Impact—Categorical Exclusion

The final rule amends the procedures currently found in Part 50 and its appendices for the filing and reviewing of applications for construction permits, operating licenses, early site reviews, and standard design approvals. As such they meet the eligibility criteria for the categorical exclusion set forth in 10 CFR 51.22(c)(3). That section applies to "[a]mendments to . . . Part [] 50 . . . which relate to (i) procedures for filing and reviewing applications for licenses or construction permits or other forms of permission. . . ." As the Commission explained in promulgating this exclusion, "[a]lthough amendments of this type affect substantive parts of the Commission's regulations, the amendments themselves relate solely to matters of procedure. [They] . . . do not have an effect on the environment." 49 FR 9352, 9371, col. 3 (March 12, 1984) (final environmental protection regulations).[6] Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with these final rules.[7]

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 *et seq.*). These requirements have been submitted to the Office of Management and Budget (OMB) for any review appropriate under the Act. The effective date of this rule provides for the ninety days required for OMB review of the information collection requirements contained in the rule.

Public reporting burden for this collection of information is estimated to average 22,000 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing the reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Records and Reports Management Branch, Division of Information Suppport Services, Office of Information and Resources Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555; and to the Paperwork Reduction Project (3150-0000), Office of Management and Budget, Washington, DC 20503.

Regulatory Analysis

As presently constituted, the American population of nuclear power reactors consists largely of one-of-a-kind designs. Experience has shown that the highly individualistic character of this population has consumed enormous resources in the processes of design, construction, and safety review. Because, typically, design of a plant was not complete when construction of it began, many safety questions were not resolved until late in the licensing proceeding for that plant. The late resolution of questions introduced great uncertainty into proceedings, because the process of resolution often entailed lengthy safety reviews, construction delays, and backfits. Moreover, the low incidence of duplication among designs has meant that experience gained in the construction and operation of a given plant has often not been useful in the construction and operation of any other plant, and has made the generic

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[6] It makes no substantive difference for the purpose of the categorical exclusion that the amendments are in a new Part 52 rather than in Part 50. The amendments are, in fact, amendments to the Part 50 procedures and could have been placed in that part.

[7] The requirements concerning testing of full-size prototypes of advanced reactors, see § 52.47, may appear not to fit into the category excluded by § 51.22(c)(3), since to comply with the requirements, an applicant may have to build and test a prototype plant, an act clearly with an environmental impact. Nonetheless, § 52.47 is eligible for exclusion under § 51.22(c)(3). Unlike, for instance, the promulgation of a safety rule which applies to operating plants, the formal action of promulgating § 52.47 has only a potential impact on the environment. That impact becomes actual only if a designer chooses to pursue certification of a certain kind of advanced design. Under the present circumstances, no meaningful environmental assessment or impact statement can be made. Cf. 49 FR at 9372, cols. 2–3 (entering into an agreement with a State under Section 274 of the Atomic Energy Act has no immediate or measurable environmental impact and therefore warrants a categorical exclusion). The issuance of the construction permit and operating license for a prototype plant would, of course, be a major federal action with a significant impact on the environment, and would entail the preparation of an environmental impact statement. Cf. id., col. 3 (the States must prepare detailed environmental analyses before they license certain activities).

resolution of continuing safety issues more complicated.

In the face of this experience with a population of unique plants, there have long been fundamentally only three alternatives for Commission action, the last two of them not mutually exclusive: either make no effort to bring about an increased degree of standardization, or propose legislation on standardization, or enact by rulemaking as much of a scheme for promoting standardization as the Commission's current statutory authority permits. The Commission has for some time concluded against the first alternative, having decided that a substantial increase in standardization would enhance the safety and reliability of nuclear power plants and require fewer resources in safety reviews of plants, and that the Commission should have in place provisions for the review of standardized designs and other devices for assuring early resolution of safety questions. The Commission has therefore pursued standardization both by proposing legislation—without success—and by promulgating rules, in particular Appendices M, N, and O to Part 50 (now Part 52) of 10 CFR. Lacking legislation on standardization, the Commission believes that the most suitable alternative for encouraging further standardization is to fill out and expand the Commission's regulatory scheme for standardization and early resolution of safety issues.

Therefore, the Commission now promulgates a new set of regulations, to be placed in a new part in 10 CFR, Part 52. This new part facilitates the early resolution of safety issues by providing for pre-construction-permit approval of power plant sites, Commission certification of standardized designs, and the issuance of licenses which combine permission to construct a plant with permission to operate it once construction of it has been successfully completed. Ideally, a future applicant will reference an approved site and a certified design in an application for a combined license, thus obviating the need for an extensive review of the application and construction. The provision in Part 52 for Commission certification of designs has the additional objective of encouraging the use of standardized designs, thereby adding to the benefits of early resolution the safety benefits of accumulated experience and the economic benefits of economies of scale and transferable experience.

Quantification of the costs and benefits of this rulemaking is probably not possible. Much depends on the extent to which the industry pursues standardization. Clearly, if the Commission and the industry spend the resources necessary to certify a score of designs and then no applicant references any of them, those resources will have been largely wasted. On the other hand, it is just as clear that if a score of plants uses a single certified design, there will have been a great saving of the resources of the industry, the agency, and the interested public alike. To be added to the uncertainties surrounding the industry's response, there are also uncertainties concerning the costs of the certification process, and the costs of developing the designs themselves, especially the advanced designs, which may require testing of prototypes. However, if the industry finds it in its interest to proceed with the development of nuclear power, there is every reason to expect that the safety and economic benefits of standardization will far outweigh the upfront costs of design and Commission certification: Review time for applications for licenses will be drastically reduced, the public brought into the process before construction, construction times shortened, economies of scale created, reliability of plant performance increased, maintenance made easier, qualified vendor support made easier to maintain, and, most important, safety enhanced.

Thus, the rationale for proceeding with this rulemaking: There is no absolute assurance that certified designs will in fact be used by the utilities; however, it is certain that if the reasonably expected benefits of standardization are to be gained, then the Commission must have the procedural mechanisms in place for review of applications for early site approvals, design certifications, and combined licenses. The most fundamental choice is, of course, the industry's, to proceed or not with standardization, according to its own weighing of costs and benefits. But the Commission must be ready to perform its review responsibilities if the industry chooses standardization.

Regulatory Flexibility Act Certification

The final rule will not have a significant impact on a substantial number of small entities. The final rule will reduce the procedural burden on NRC licensees by improving the reactor licensing process. Nuclear power plant licensees do not fall within the definition of small businesses in section 3 of the Small Business Act, 15 U.S.C. 632, the Small Business Size Standards of the Small Business Administration in 13 CFR Part 121, or the Commission's Size Standards published at 50 FR 50241 (Dec. 9, 1985). The impact on intervenors or potential intervenors will be neutral. For the most part, the final rule will affect the timing of hearings rather than the scope of issues to be heard. For example, many site and design issues will be considered earlier, in connection with the issuance of an early site permit or standard design certification, rather than later, in connection with a facility licensing proceeding. Similarly, a combined licensed proceeding will include consideration of many of the issues that would ordinarily be deferred until the operating license proceeding. Thus, the timing rather than the cost of participating in NRC licensing proceedings will be affected. Intervenors may experience some increased preparation costs if they seek to reopen previously decided issues because of the increased showing that will be required. Once a hearing commences, however, an intervenor's costs should be decreased because the issues will be more clearly defined than under existing practice. Therefore, in accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that the final rule will not have a significant economic impact on a substantial number of small entities and that, therefore, a regulatory flexibility analysis need not be prepared.

Backfit Analysis

This rule does not modify or add to the systems, structures, components, or design of a facility; or the design approval or manufacturing license for a facility; or the procedures or organization required to construct or operate a facility. However, it could be argued that this rule modifies and adds to the procedures or organization required to design a facility, since the rule adds to, or else at least spells out, the requirements for applicants for design certifications. Moreover, the rule, at the very least, substantially modifies the expectations of anyone who had hoped to apply for a design certification under the previously existing section 7 of Appendix O, particularly of any such who presently hold preliminary or final design approvals under that Appendix.

Nonetheless, the Commission believes that the backfit rule does not apply to this rule and, therefore, that no backfit analysis pursuant to 10 CFR 50.109(c) is required for this rule. The backfit rule was not intended to apply to every action which substantially changes settled expectations. Clearly, the backfit rule would not apply to a rule which would impose more stringent requirements on all future applicants for construction permits, even though such a rule arguably might have an adverse impact on a person who was considering applying for a permit but had not done so yet. In this latter case, the backfit rule protects the construction permit holder, not the prospective applicant, or even the present applicant. The final rule below is of the character of such a hypothetical rule. The final rule arguably imposes more stringent requirements for design certification and thereby may have an adverse impact on some persons. However, the effects of the final rule will be largely prospective, and the rule does not require any present holder of a design approval (no person holds a design certification) to meet new standards in order to remain in possession of such an approval.

List of Subjects

10 CFR Part 2

Administrative practice and procedure, Antitrust, Byproduct material, Classified information, Environmental protection, Nuclear Materials, Nuclear power plants and reactors, Penalty, Sex discrimination, Source material, Special nuclear material, Waste treatment and disposal.

10 CFR Part 50

Antitrust, Classified information, Fire protection, Incorporation by reference, Intergovernmental relations, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, Reporting and recordkeeping requirements.

10 CFR Part 51

Administrative practice and procedure, Environmental impact statement, Nuclear materials, Nuclear power plants and reactors, Reporting and recordkeeping requirements.

10 CFR Part 52

Administrative practice and procedure, Antitrust, Backfitting, Combined license, Early site permit, Emergency planning, Fees, Inspection, Limited work authorization, Nuclear power plants and reactors, Probabilistic risk assessment, Prototype, Reactor siting criteria, Redress of site, Reporting and recordkeeping requirements, Standard design, Standard design certification.

10 CFR Part 170

Byproduct material, Nuclear materials, Nuclear power plants and reactors, Penalty, Source material, Special nuclear material.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 552 and 553, the Commission is adding to 10 CFR Chapter I a new Part 52 and adopting amendments to 10 CFR Parts 2, 50, 51, and 170:

54 FR 19632
Published 5/8/89

10 CFR Part 52

RIN 3150-AC61

Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors

Correction

In rule document 89-8832 beginning on page 15372 in the issue of Tuesday, April 18, 1989, make the following corrections:

§ 52.43 [Corrected]

1. On page 15390, in the first column, in § 52.43(b), in the fourth line, "is" should read "in".

§ 52.64 [Corrected to read § 52.63]

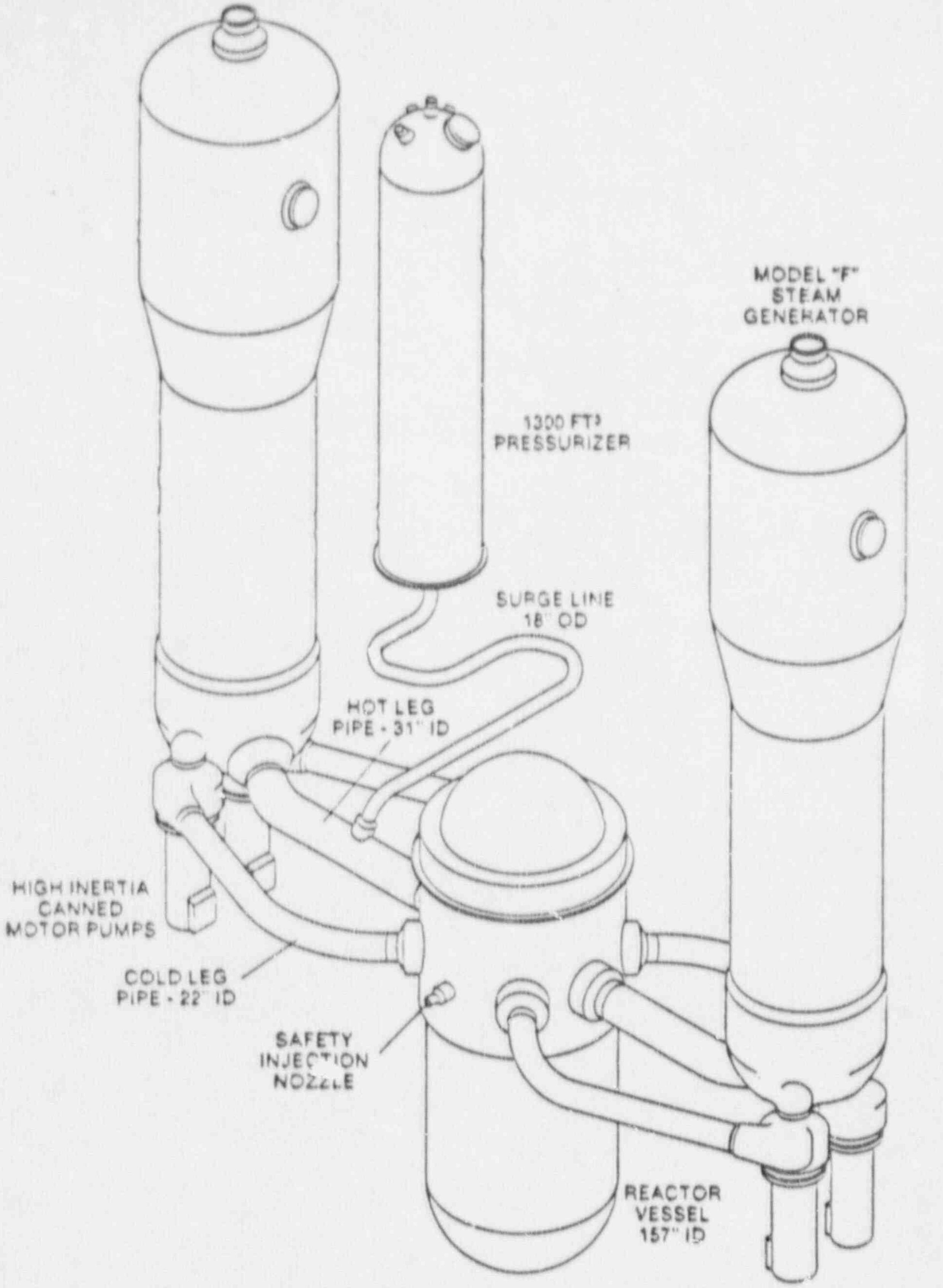
2. On page 15392, in the second column, § 52.64 Finality of standard design certification should read § 52.63 Finality of standard design certification.

PRESENTATION TO U.S. NRC

AUGUST 24, 1989

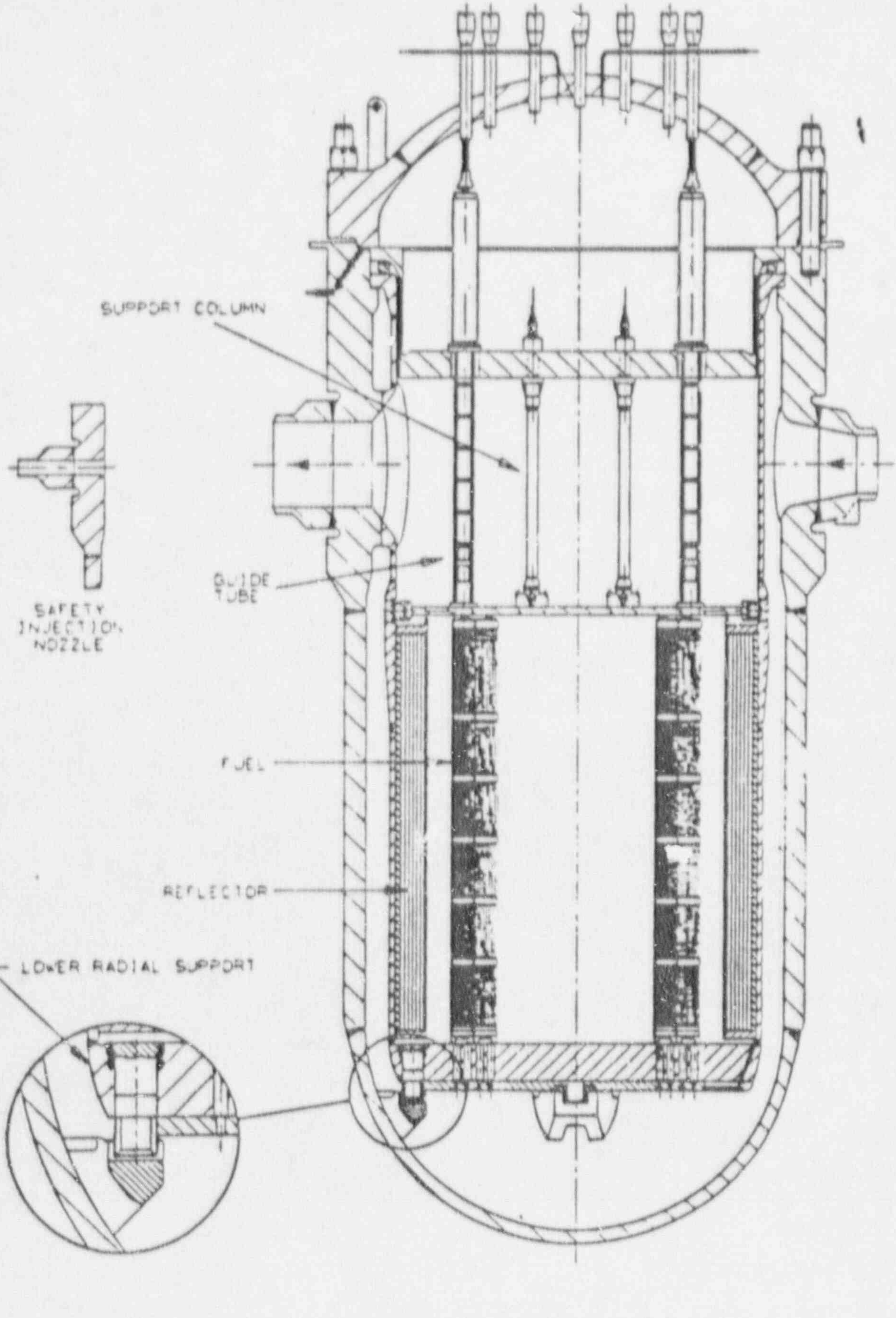
REACTOR COOLANT SYSTEM

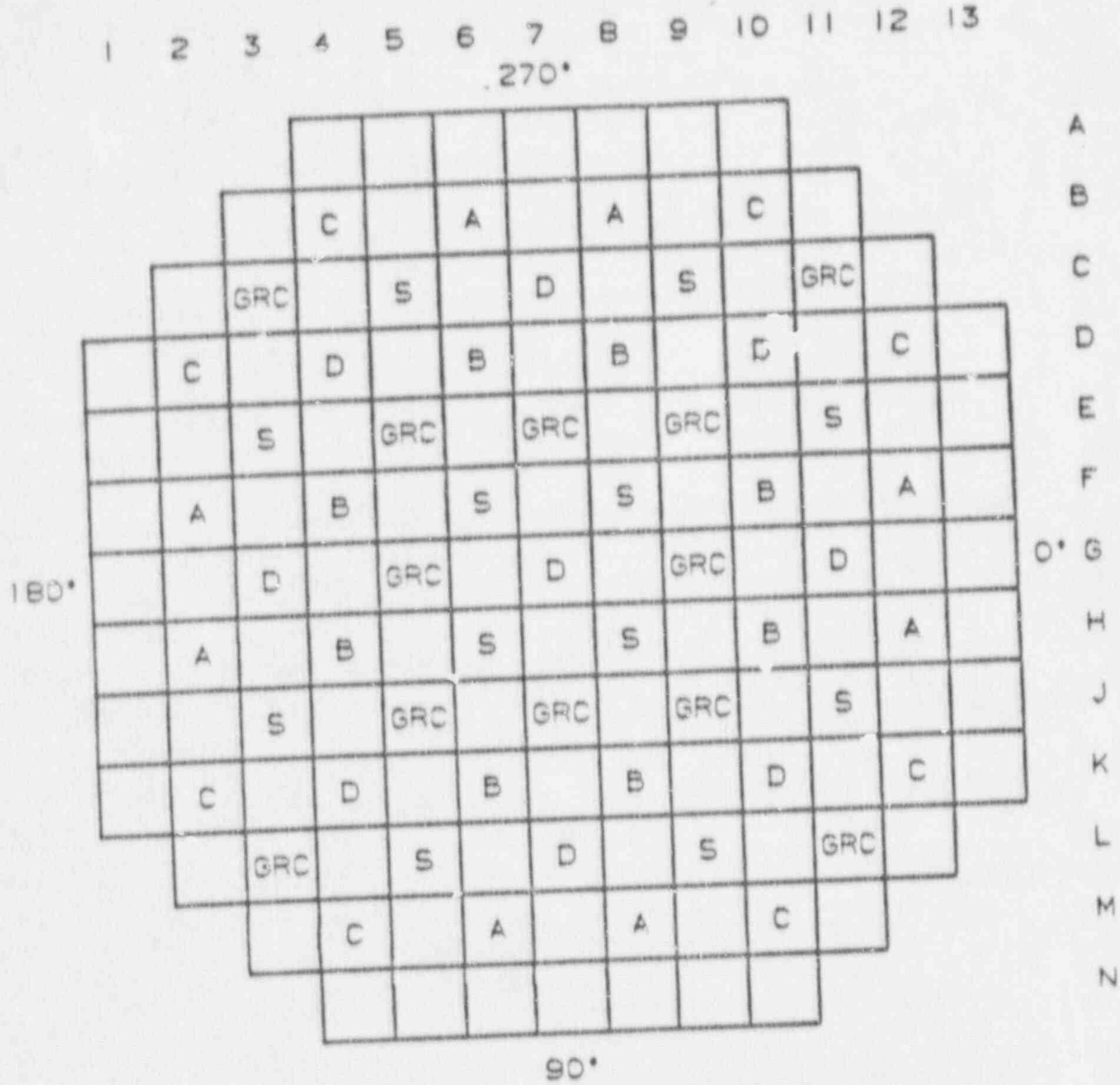
S. N. TOWER



AP600 Reactor Coolant System

. . 600 REACTOR GENERAL LAYOUT





BANK	BANK TYPE	NUMBER OF RODS
D	CONTROL	9
C	CONTROL	8
B	CONTROL	8
A	CONTROL	8
S	SHUTDOWN	12
GRC	GRAY	12

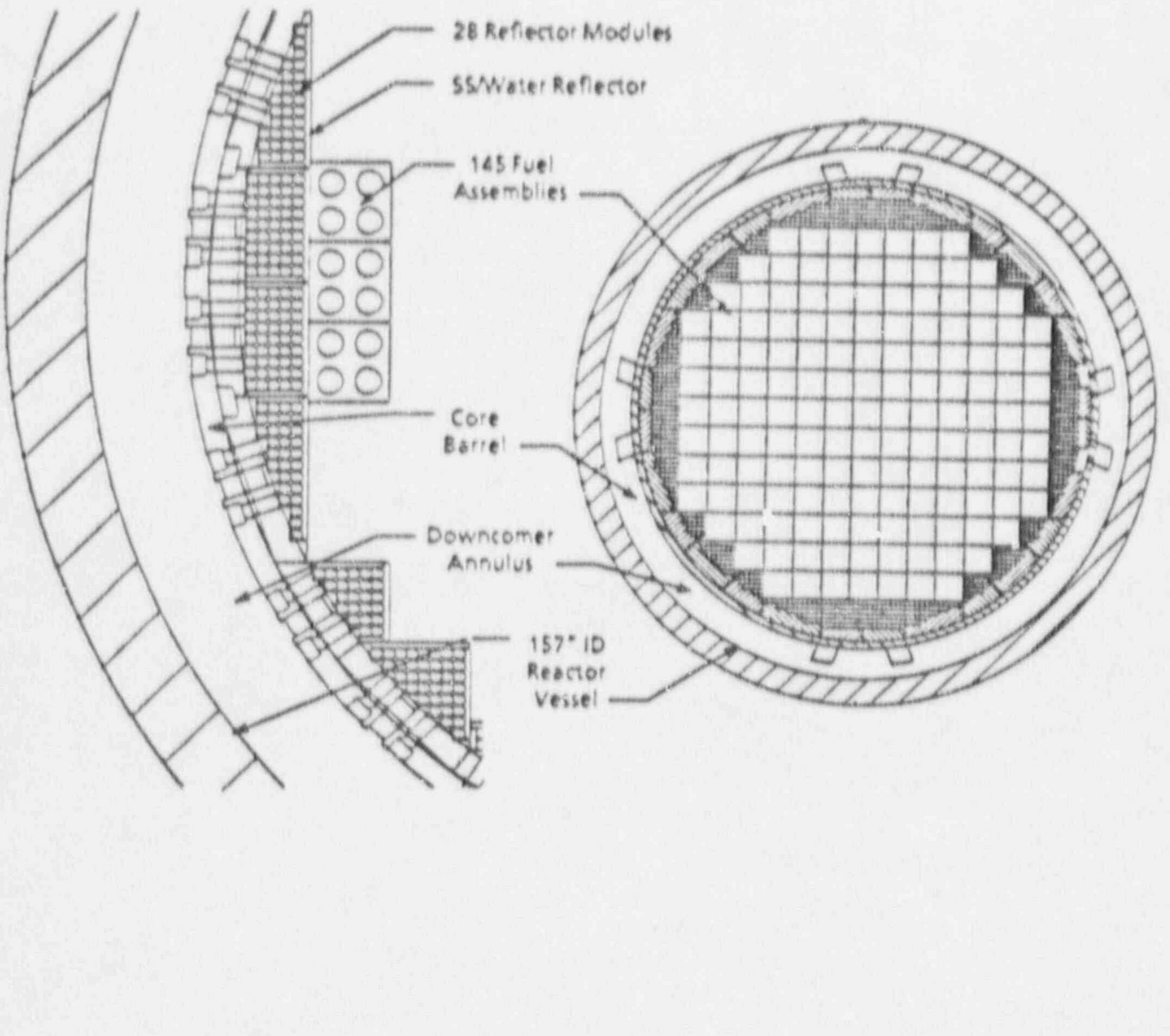
AP600 Rod Cluster Locations

CORE PARAMETERS

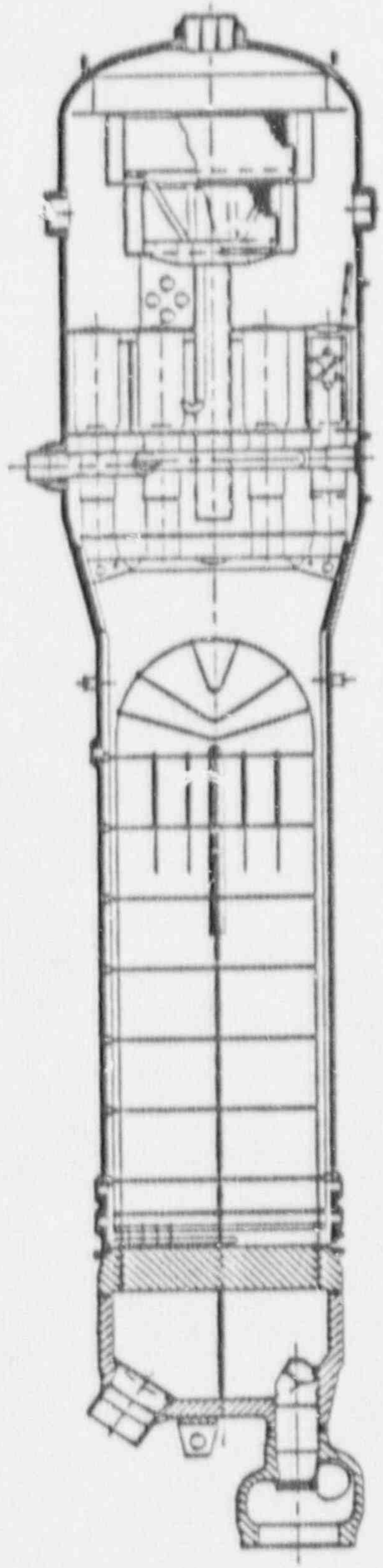
Parameter	Value
Core power, MWt	1812
No. of fuel assemblies	145
No. of fuel rods	38,280
Fuel assembly pitch, in.	8.466
Inter-assembly gap, in.	0.050 at Zr mid-grid
Active core length, in.	144
Core loading, MTU	61.02
Geometric fuel density, percent of theoretical	94.5
Average linear power, kw/ft	3.84
Average power density, kw/liter	73.89
Average specific power, kw/kg	29.7
Fuel rod heat transfer area, ft^2	43,294
Average heat flux, BTU/hr-ft^2	139,133
Water volume fraction	0.6096
UO_2 volume fraction	0.2726
Zircaloy-4 volume fraction	0.1022
Water-to-fuel ratio	2.237
Fraction of reactor power generated in-core	0.974
Design value of F_Q	2.40
Design value of $F_{\Delta H}$	1.55
Design value of F_Z (cosine)	1.55

COMPARISON OF RESOURCE REQUIREMENTS
(18 Month Equilibrium Fuel Cycles)

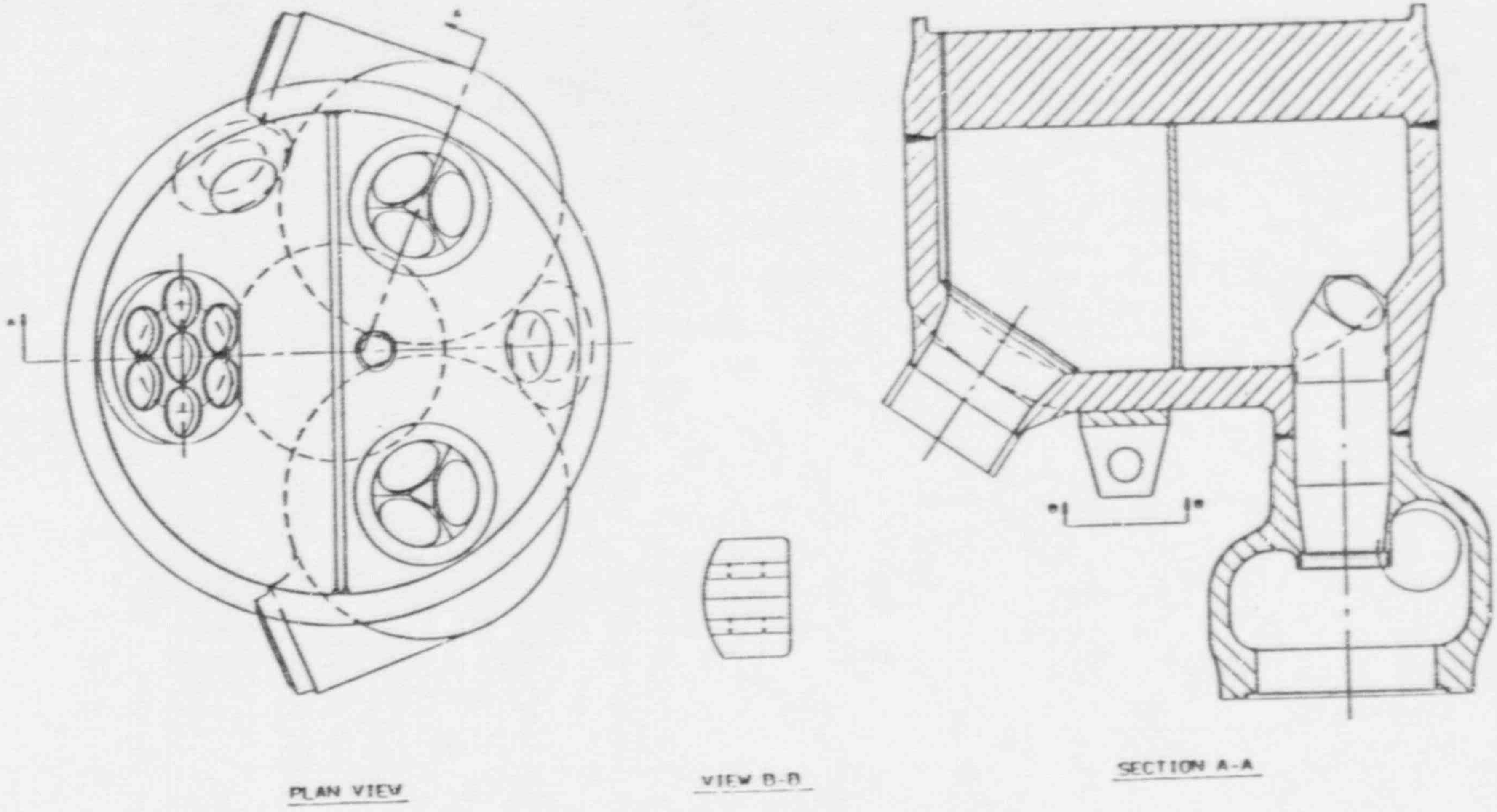
	Proposed Design		Standard 2-Loop Design
Core Power (MWt)	1812		1876
Core Loading (MTU)	61.02		49.45
Energy Output/Cycle (GWD)	843.8		843.8
No. of Feed Assemblies Required/ Cycle	48		49
Cycle Burnup (MWD/MTU)	13829		17064
Discharge Burnup (MWD/MTU)	~42000		~42000
Required Feed Fuel Enrichment (w/o U-235)	3.6		4.6
Ore Requirement (lbs U_3O_8/GWD)	414		537
(Decrease Relative to Standard Design (%))		(22.9)	
Separative Work Requirement (SWU/GWD)	134		190
(Decrease Relative to Standard Design (%))		(29.5)	



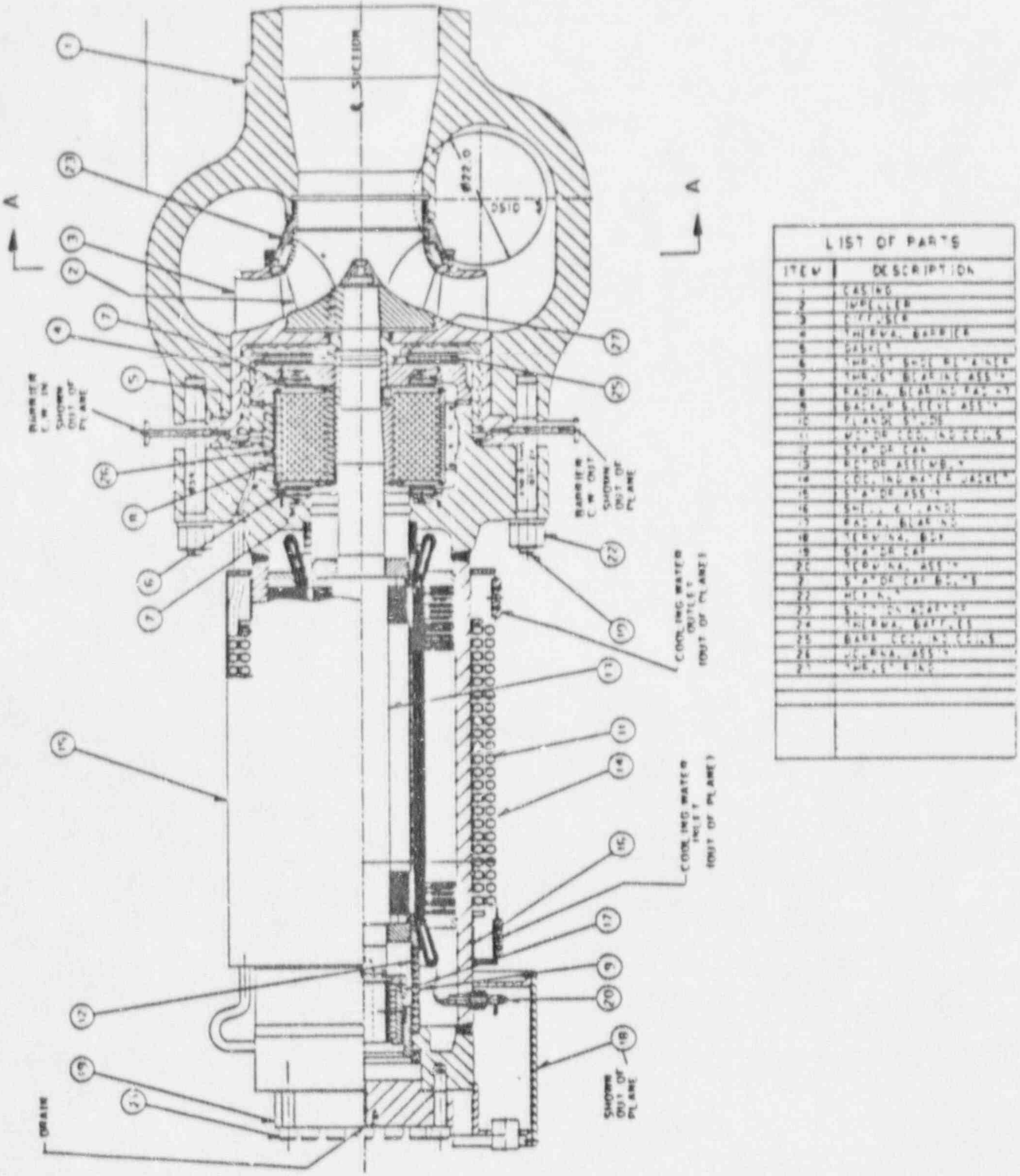
AP600 Reactor Vessel and Core Cross Section



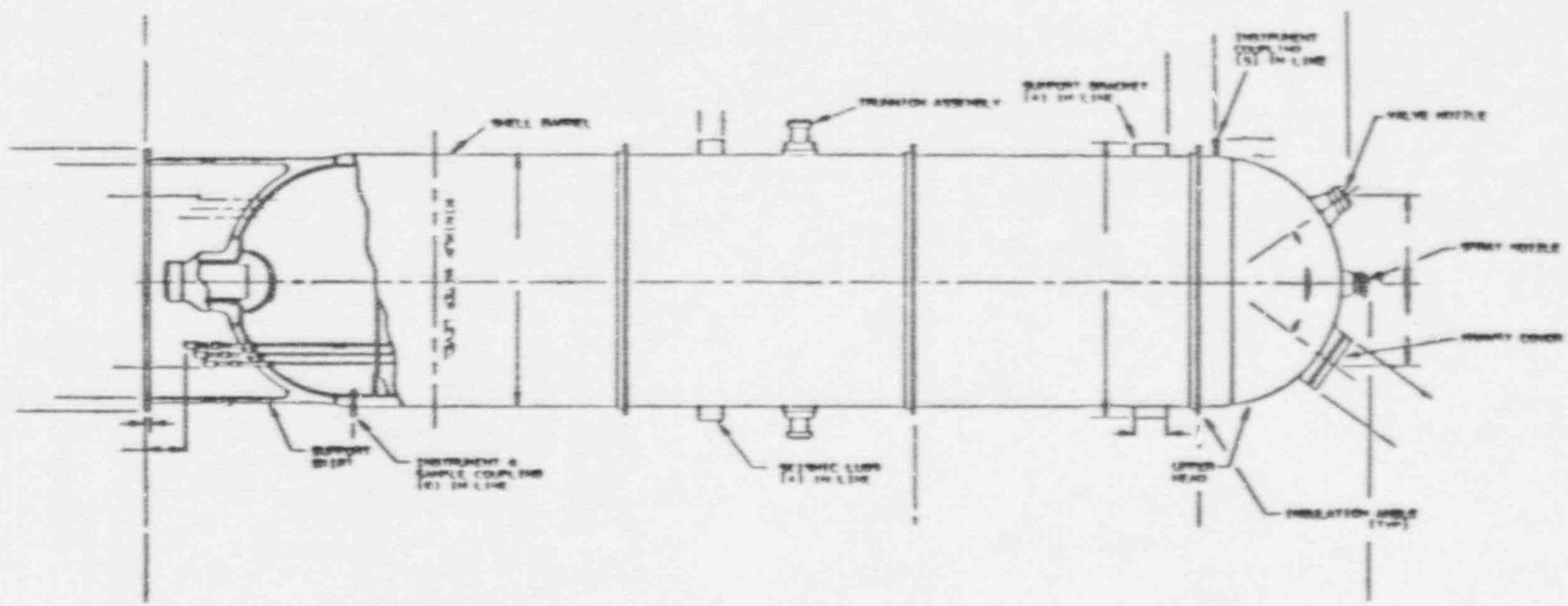
AP600 General Arrangement of the Steam Generator



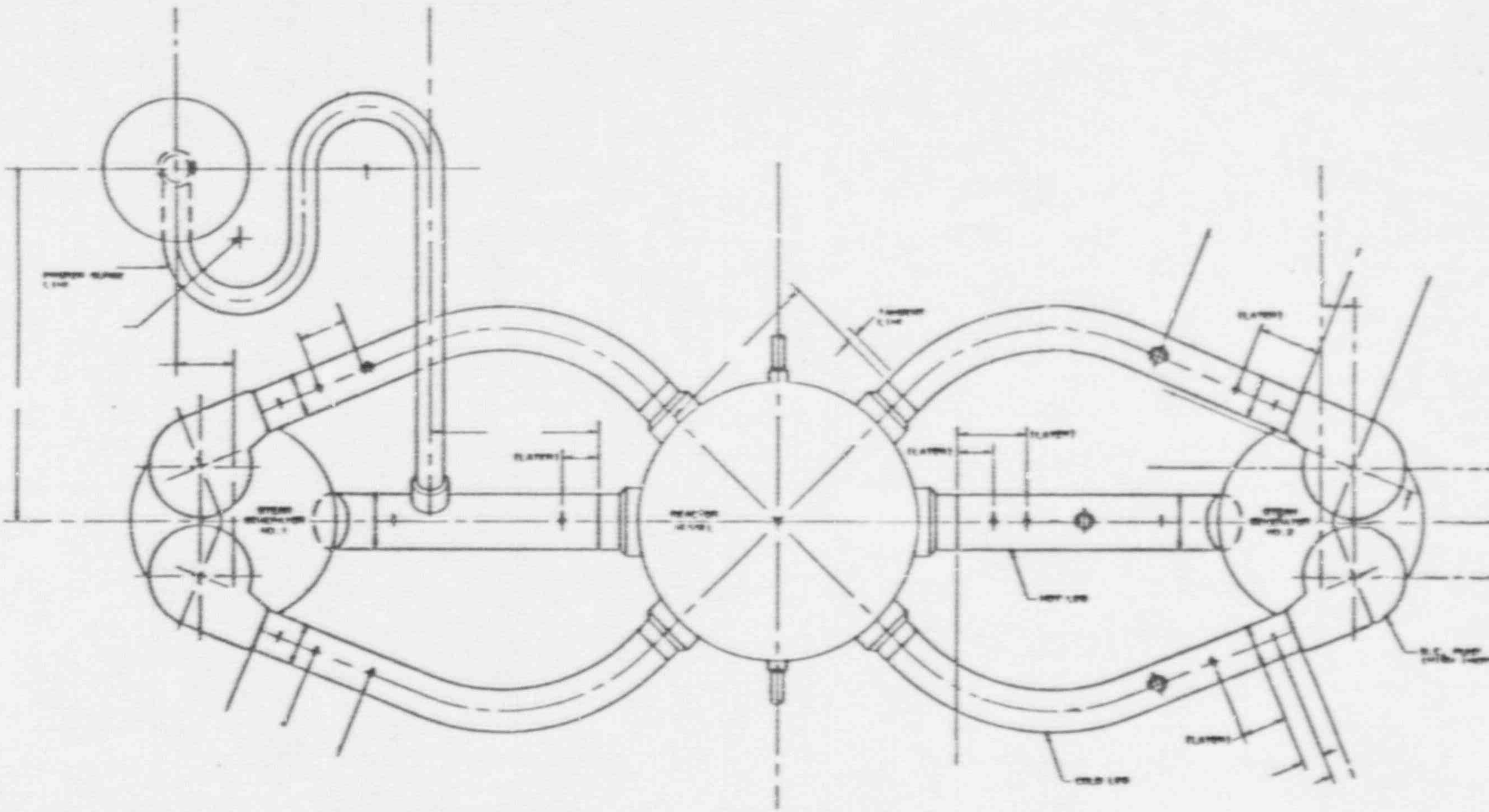
AP600 Steam Generator Pump/Channel Head Arrangement



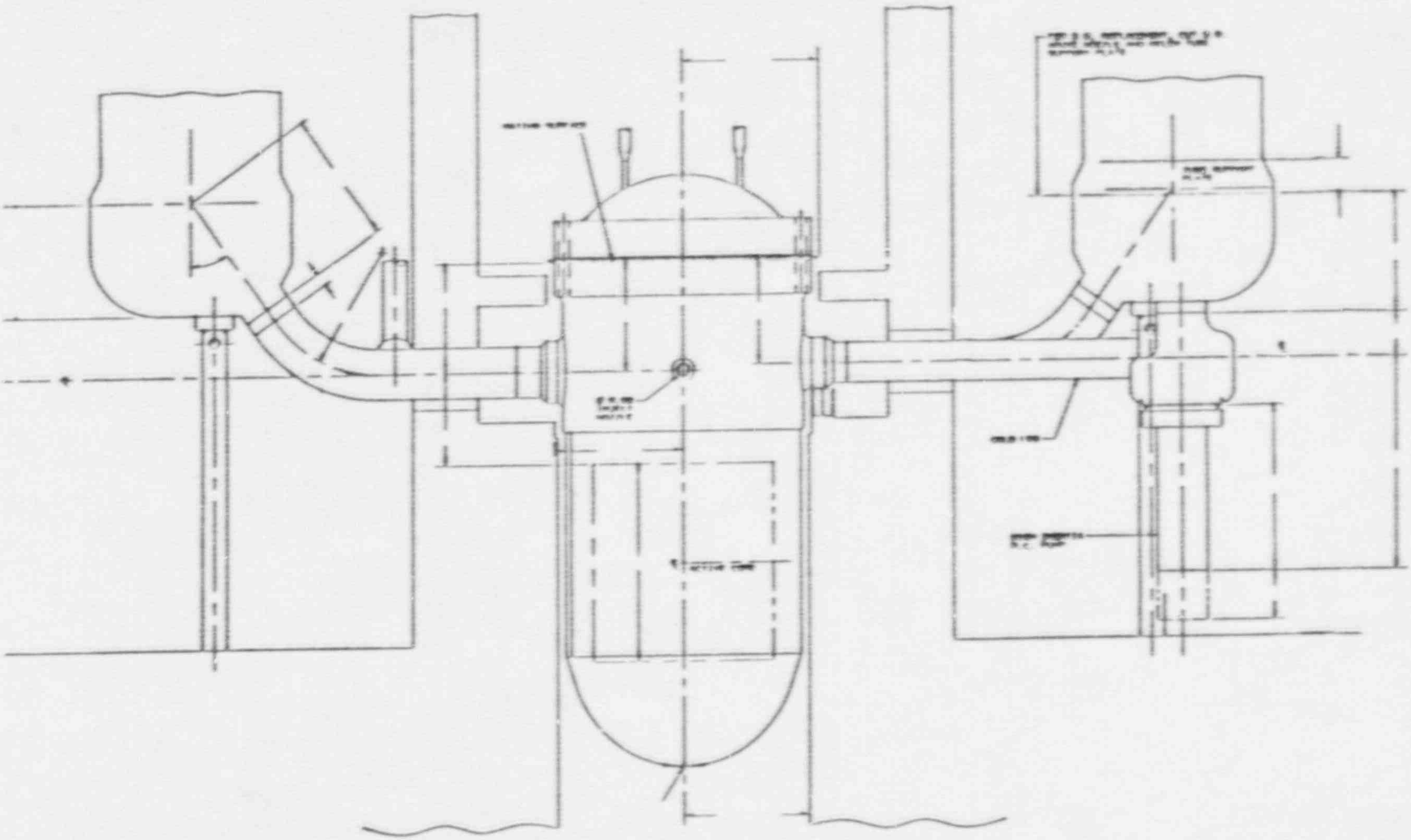
AP600 High Inertia Reactor Coolant Pump (5000 lb ft²)



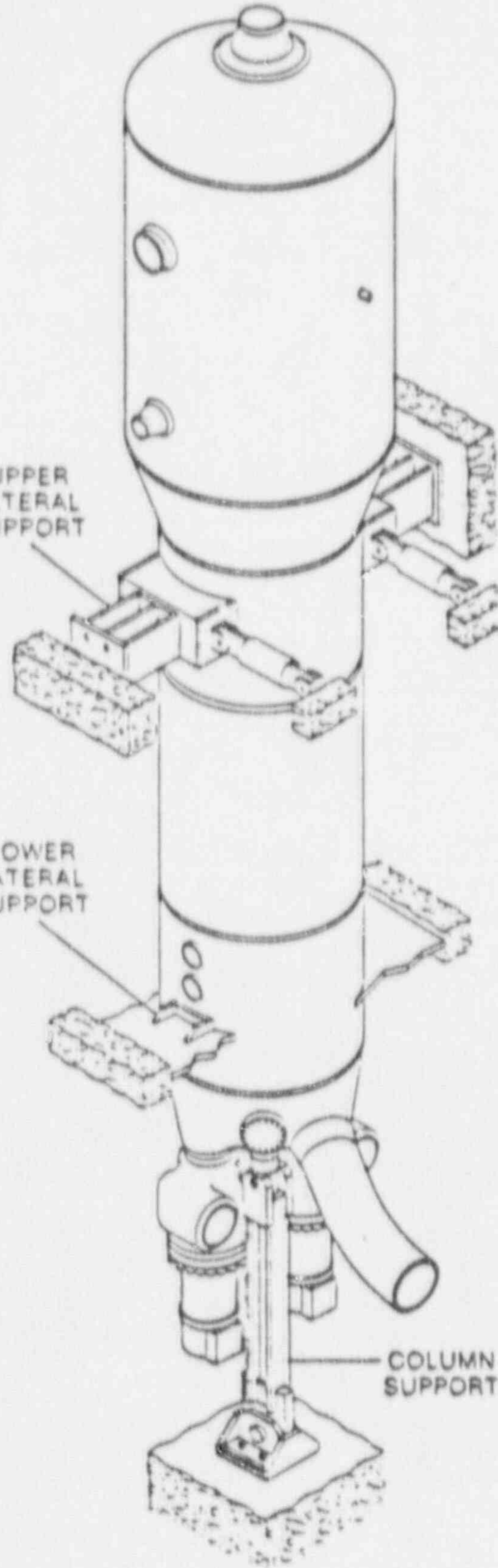
AP600 Pressurizer Vessel



AP600 Reactor Coolant Loop Piping (Plan View)



AP600 Reactor Coolant Loop Piping (Elevation View)



AP600 Steam Generator Supports

AP600 SYSTEM DESIGN
PASSIVE SAFETY SYSTEMS

PRESENTED TO NRC

AUGUST 24, 1989

TERRY SCHULZ
WESTINGHOUSE

TLS08/21/1989

AP600 SYSTEMS DESIGN APPROACH

- o SYSTEMATIC APPROACH - CRITERIA / GOALS

- o AGGRESSIVELY SIMPLIFY SYSTEMS
  - DESIGN, PROCUREMENT, CONSTRUCTION, OPERATION, INSPECTION, MAINTENANCE

- o EMPLOY PASSIVE SAFETY SYSTEMS
  - SIMPLIFIED SAFETY SYSTEMS
  - NON-SAFETY NORMAL SYSTEMS

- o OPTIMIZE / INTEGRATE SYSTEMS WITH ....
  - SAFETY AND RISK ANALYSIS
  - PLANT ARRANGEMENT
  - MODULARIZED CONSTRUCTION

AP600 - OPERATOR ACTION OBJECTIVES

- o PLANT OBJECTIVE IS TO PROVIDE CORE COOLING FOLLOWING DBA EVENTS (CH 15) FOR AN INDEFINITE TIME WITH NO OPERATOR ACTION

  - SAFETY GRADE, SINGLE FAILURE

- o SOME OPERATOR ACTIONS AND EXTERNAL ASSISTANCE WILL BE REQUIRED AFTER 3 DAYS TO MEET OTHER NRC REQUIREMENTS, SUCH AS MONITORING.

  - TEMPORARY EQUIPMENT MAY BE BROUGHT TO THE PLANT AFTER 3 DAYS AS LONG AS IT IS READILY AVAILABLE AND REASONABLE QUANITIES ARE NEEDED.

  - THE CONSEQUENCES OF FAILURE TO PROVIDE THE TEMPORARY EQUIPMENT SHOULD NOT LEAD TO CORE DAMAGE

  - THE PLANT WILL BE DESIGNED FOR EASY CONNECTIONS TO THE TEMPORARY EQUIPMENT

- o SOME LOW PROBABILITY ACCIDENTS NOT ANALYZED IN CHAPTER 15 MAY REQUIRE OPERATOR ACTION IN LESS THAN 3 DAYS

  - ACCIDENTS DURING SHUTDOWN / REFUELING

  - COMMON MODE &/OR MULTIPLE FAILURES ASSUMED IN PRA SEQUENCES

AP600 PASSIVE SAFETY FEATURES

- o PASSIVE DECAY HEAT REMOVAL
  - NATURAL CIRCULATION HX CONNECTED TO RCS

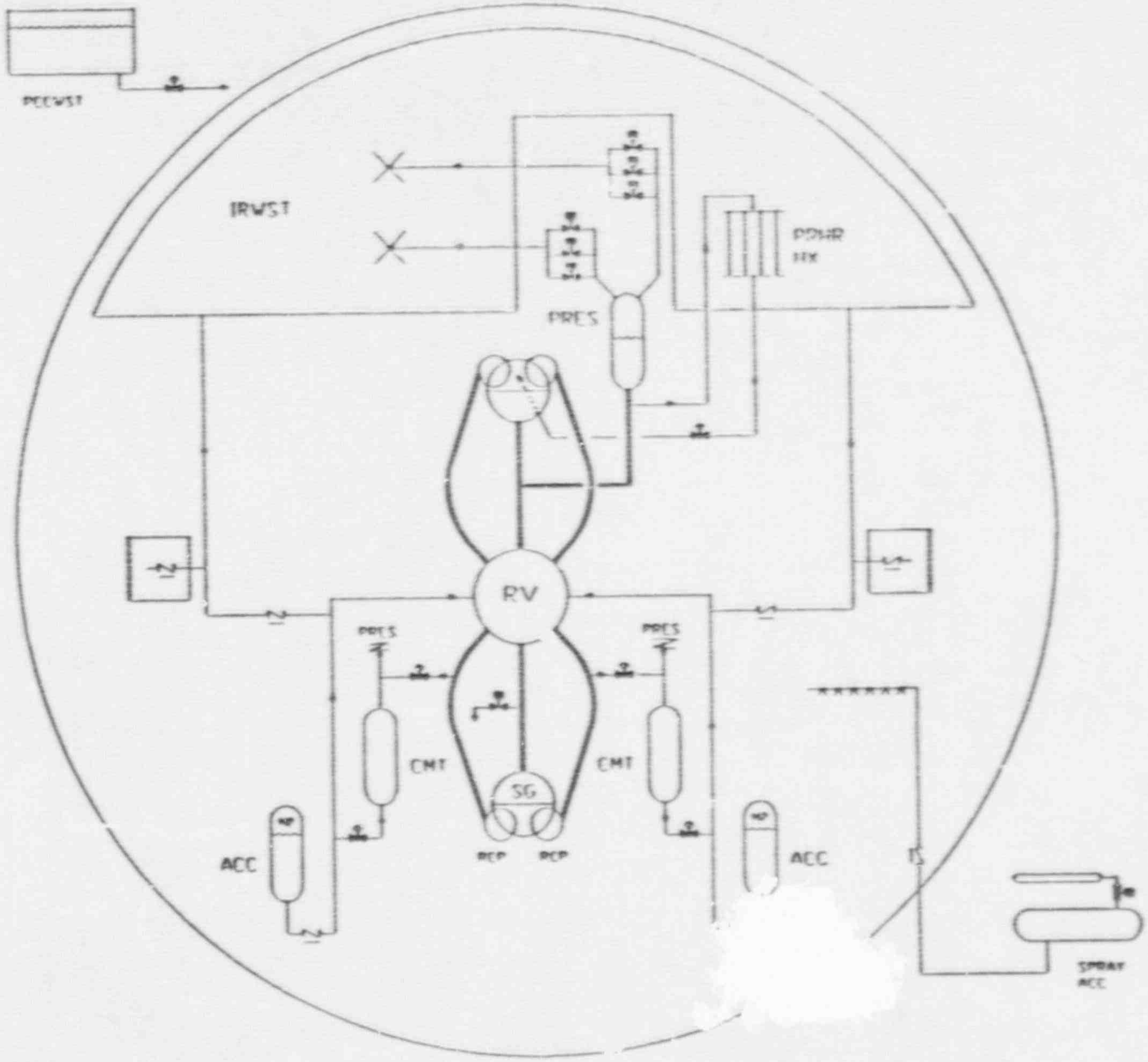
- o PASSIVE SAFETY INJECTION
  - N2 PRESSURIZED ACCUMULATORS
  - GRAVITY DRAIN CORE MAKEUP TANKS, RCS PRES
  - GRAVITY DRAIN IN-CONTAINMENT REFUELING WATER STORAGE TANK, CONTAINMENT PRES
  - GRAVITY DRAIN RECIRCULATION FROM CONTAINMENT
  - AUTOMATIC RCS DEPRESSURIZATION

- o PASSIVE CONTAINMENT COOLING
  - STEEL CONTAINEMNT SHELL TRANSFERS HEAT TO NATURAL CIRCULATION AIR COOLING AND GRAVITY DRAIN WATER EVAPORATION COOLING.

- o PASSIVE CONTAINMENT SPRAY
  - N2 PRESSURIZER ACCUMULATORS

- o EMERGENCY HVAC
  - COMPRESSED AIR FOR HABITABILITY OF ECR
  - CONCRETE WALLS FOR HEAT SINK

AP600 – PASSIVE SAFETY SYSTEMS



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AP600 - CORE DECAY HEAT REMOVAL

o NORMAL SHUTDOWNS

- ABOVE 350 F / 400 PSIG; STARTUP FEEDWATER TO SG FROM DEAERATING HEATER (2 PUMPS, BLACKOUT DIESEL)
- BELOW 350 F / 400 PSIG; SPENT FUEL COOLING SYSTEM (2 PUMPS & HX, CCW/SW COOLING, BLACKOUT DIESEL)

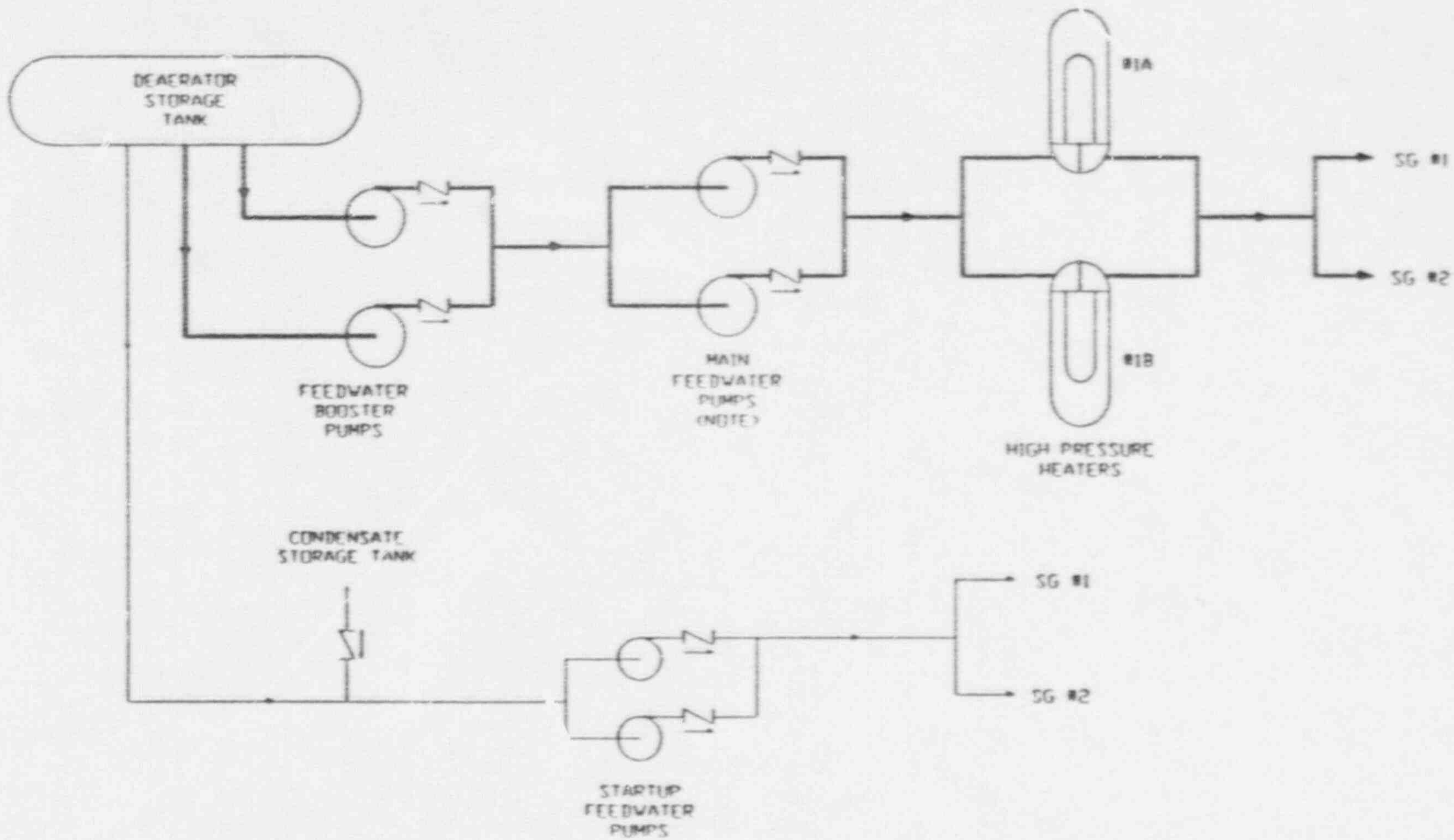
o TRANSIENTS (SAFETY)

- ANY RCS CONDITION; PASSIVE RHR HX LOCATED IN IRWST (1 HX, NATURAL CIRCULATION)
- BACKUP COOLING BY FEED & BLEED WITH CMT/ACCUM/ IRWST INJECTION & RCS DEPRESSURIZATION

o LOCA (SAFETY)

- INJECTION FROM CMT/ACCUM/IRWST ALONG WITH RCS DEPRESSURIZATION
- LONG TERM HEAT REMOVAL BY STEAMING TO CONTAINEMNT WITH PASSIVE CONTAINMENT COOLING PROVIDING HEAT SINK.

AP600 - MAIN & STARTUP FEEDWATER SYSTEMS



NOTE - VARIABLE SPEED MOTOR DRIVEN PUMPS

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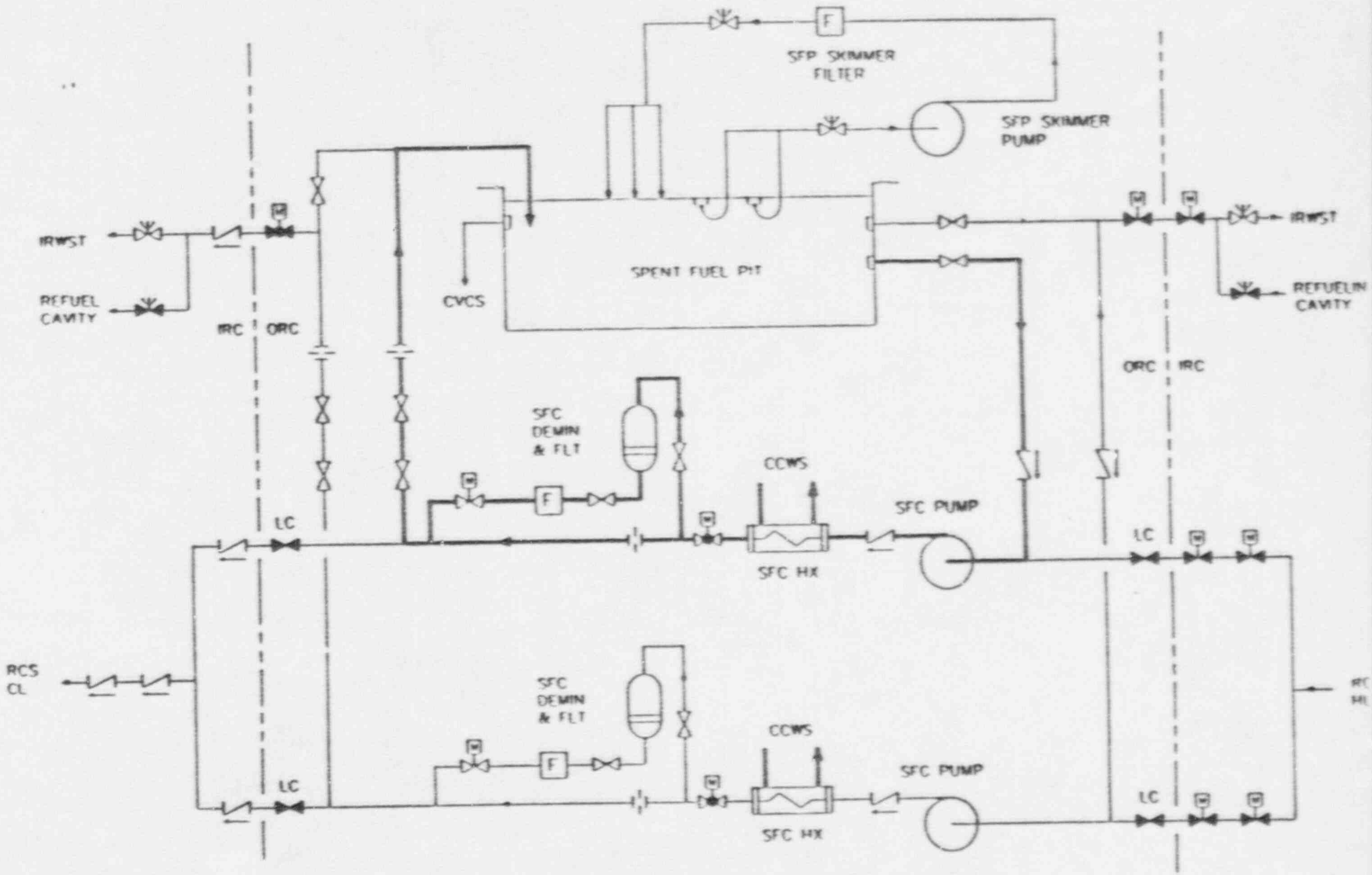
AP600 SPENT FUEL COOLING SYSTEM



WESTINGHOUSE - 4/88

AP600 SPENT FUEL PIT COOLING

- NORMAL COOLING BY NON-SAFETY COOLING SYSTEM
  - TWO PUMPS / HX
  - ANS SAFETY CLASS 2 / 3, BUT NOT ACTIVE
  - POWER FROM OFFSITE OR BLACKOUT DIESEL

- EMERGENCY COOLING BY HEAT CAPACITY OF PIT
  - LONG TIME TO HEATUP / BOIL DOWN

	BOILING /	TOP OF FUEL
MID CYCLE	8 DAYS	64 DAYS
UNLOAD 1/3 CORE	1 DAY	9 DAYS
UNLOAD FULL CORE	1/2 DAY	4 DAYS

  - EASY TO ADD WATER TO PIT OR RESTORE NORMAL COOLING IN TIME AVAILABLE

TL508/22/1989

AP600 MID-LOOP OPERATION

- IMPROVED RHR CONNECTION TO THE RCS HOT LEG DEVELOPED AND TESTED (1/4 SCALE)
    - SHORT SECTION OF LARGER PIPE (20") CONNECTED TO BOTTOM OF HOT LEG
    - ALLOWS PUMP OPERATION WITH LOWER WATER LEVELS BEFORE ANY AIR IS ENTRAINED AND WITH HOT LEG ESSENTIALLY EMPTY WITHOUT AIR BINDING

- RAISED SG ALLOWS HIGHER NORMAL MID LOOP LEVEL (80%)

- NARROW RANGE HOT LEG LEVEL INSTRUMENT WITH MCR READOUT AND ALARM

- ALL NORMAL RCS DRAIN OPERATIONS CONTROLLED AND MONITORED FROM MCR

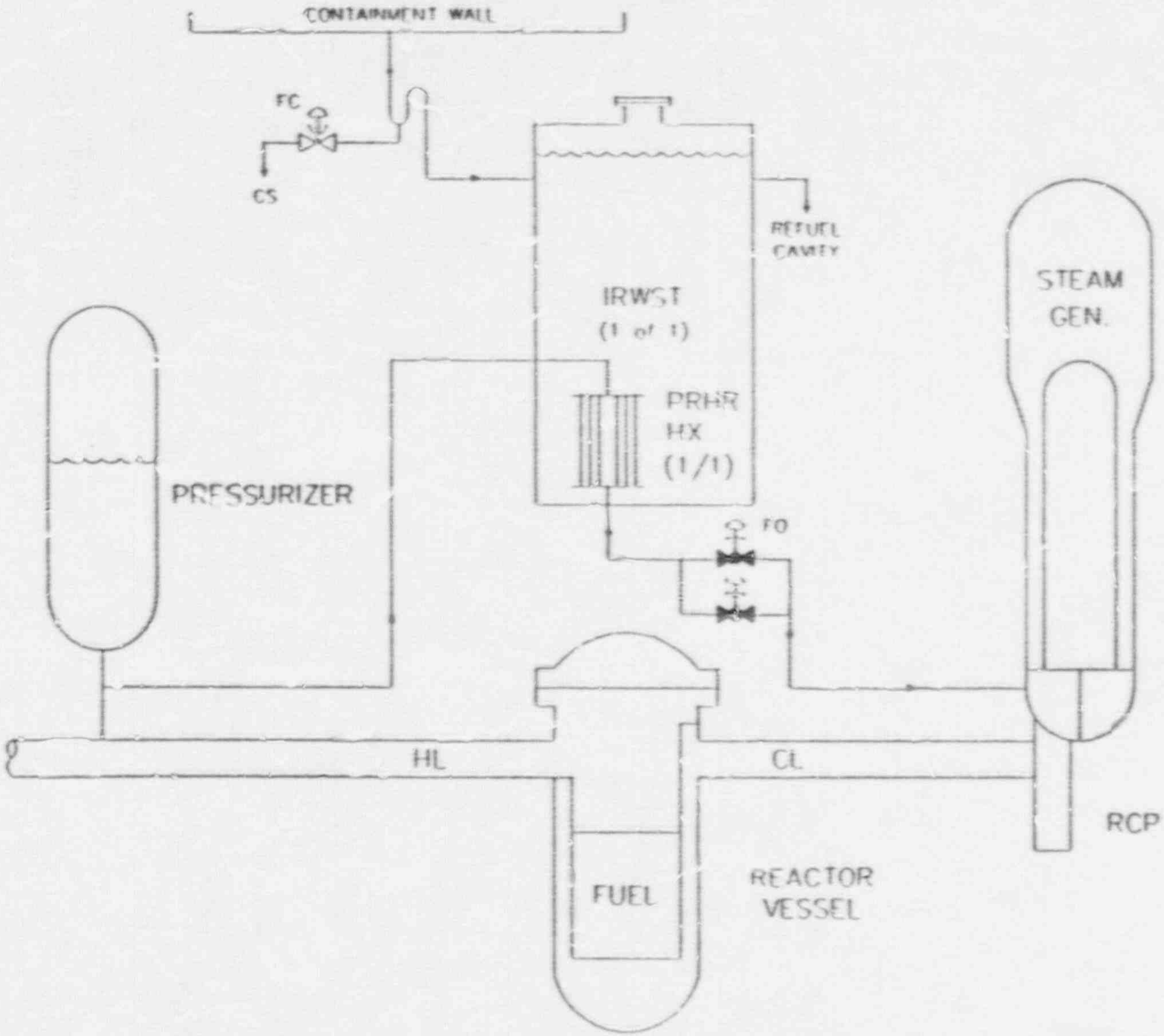
- SUCTION LINE DESIGNED TO PROVIDE PUMP WITH ADEQUATE NPSH AT FULL FLOW WITH SATURATED WATER

- SUCTION LINE ROUTED SO THAT IT IS SELF VENTING

- RUGGED PUMP DESIGN CAN TOLERATE SOME AIR INGESTION AND CONTINUE OPERATING; CAN BECOME AIR BOUND WITHOUT DAMAGE

TLS05/02/1989

AP600 - PASSIVE RHR HX



WESTINGHOUSE - 11/88

AP600 - RCS INVENTORY CONTROL

o NORMAL OPERATION

- CVCS MAKEUP PUMPS (2 PUMPS, NNS, BLACKOUT DIESEL)

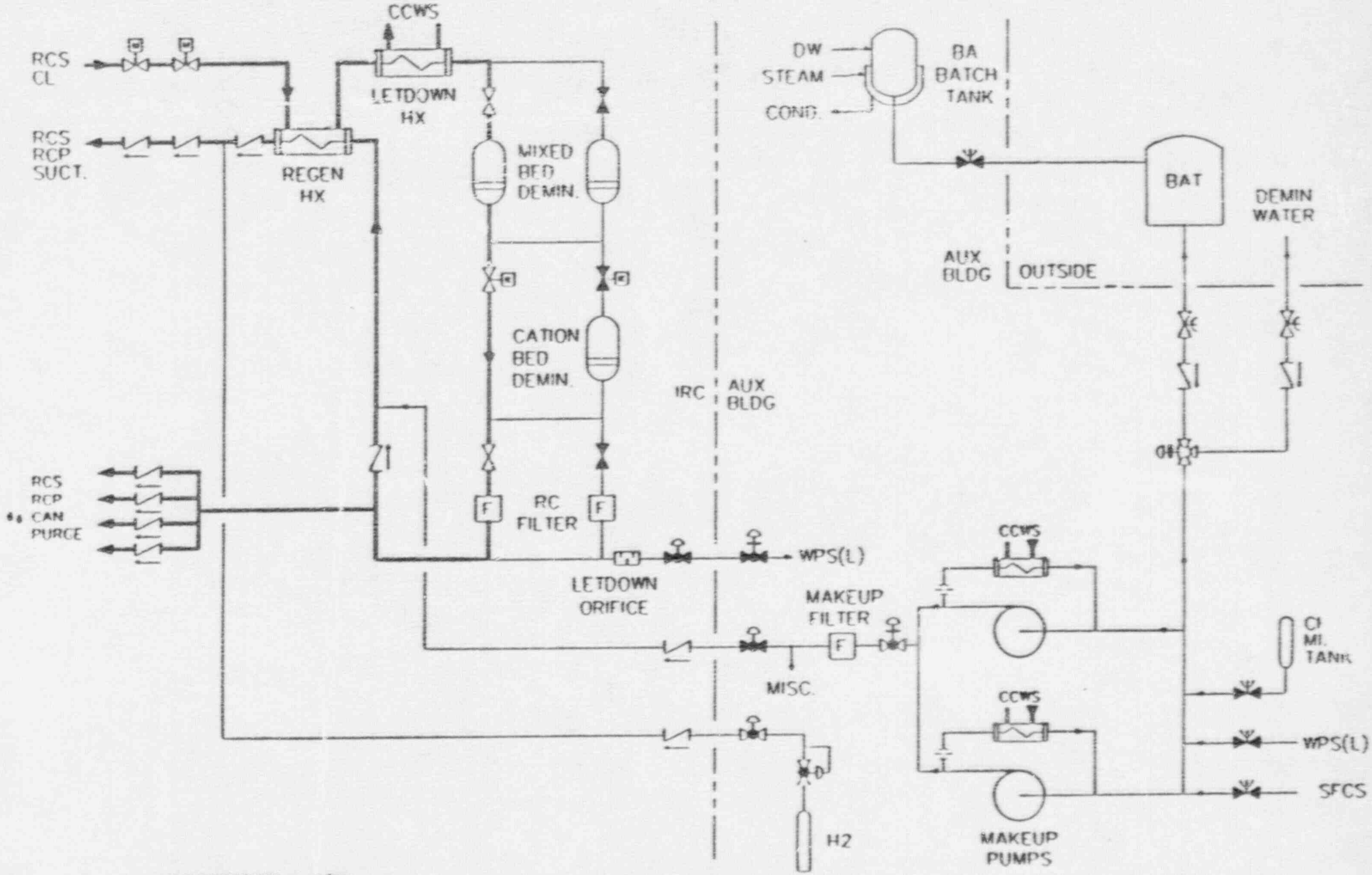
o TRANSIENTS AND SMALL LEAKS (SAFETY)

- CORE MAKEUP TANKS (2 TANKS, RCS PRES, GRAVITY)

LOCA (SAFETY)

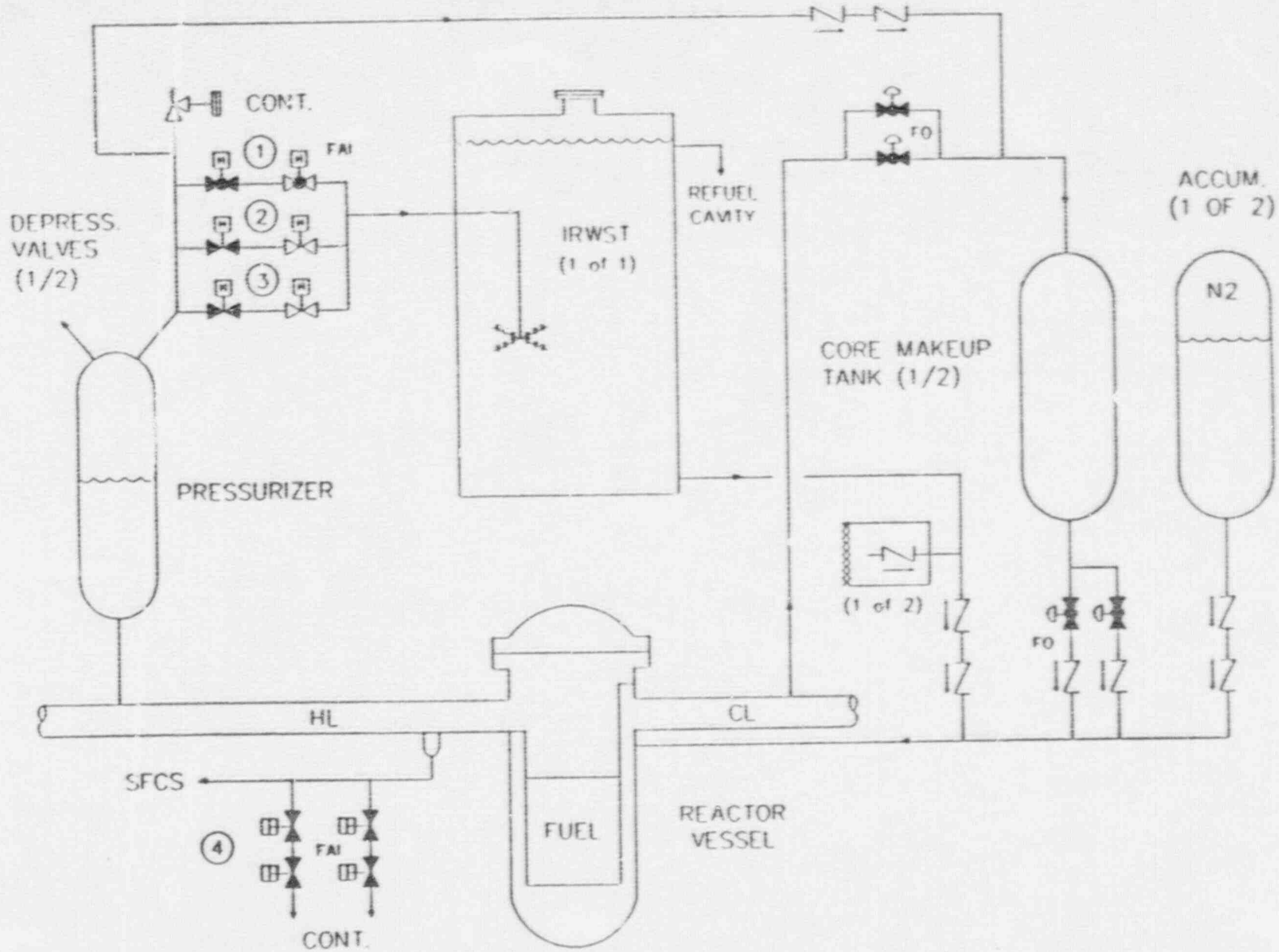
- CORE MAKEUP TANKS (2 TANKS, RCS PRES, GRAVITY)
- ACCUMULATORS (2 TANKS, 700 PSIG, N2)
- AUTOMATIC RCS DEPRESSURIZATION ALLOWS LONG TERM INJECTION FROM:
  - INCONTAINMENT REFUELING WATER STORAGE TANK (1 TANK, CONTAINMENT ATM)
  - RECIRCULATION FROM CONTAINMENT BY GRAVITY

AP600 - CHEMICAL AND VOLUME CONTROL SYSTEM



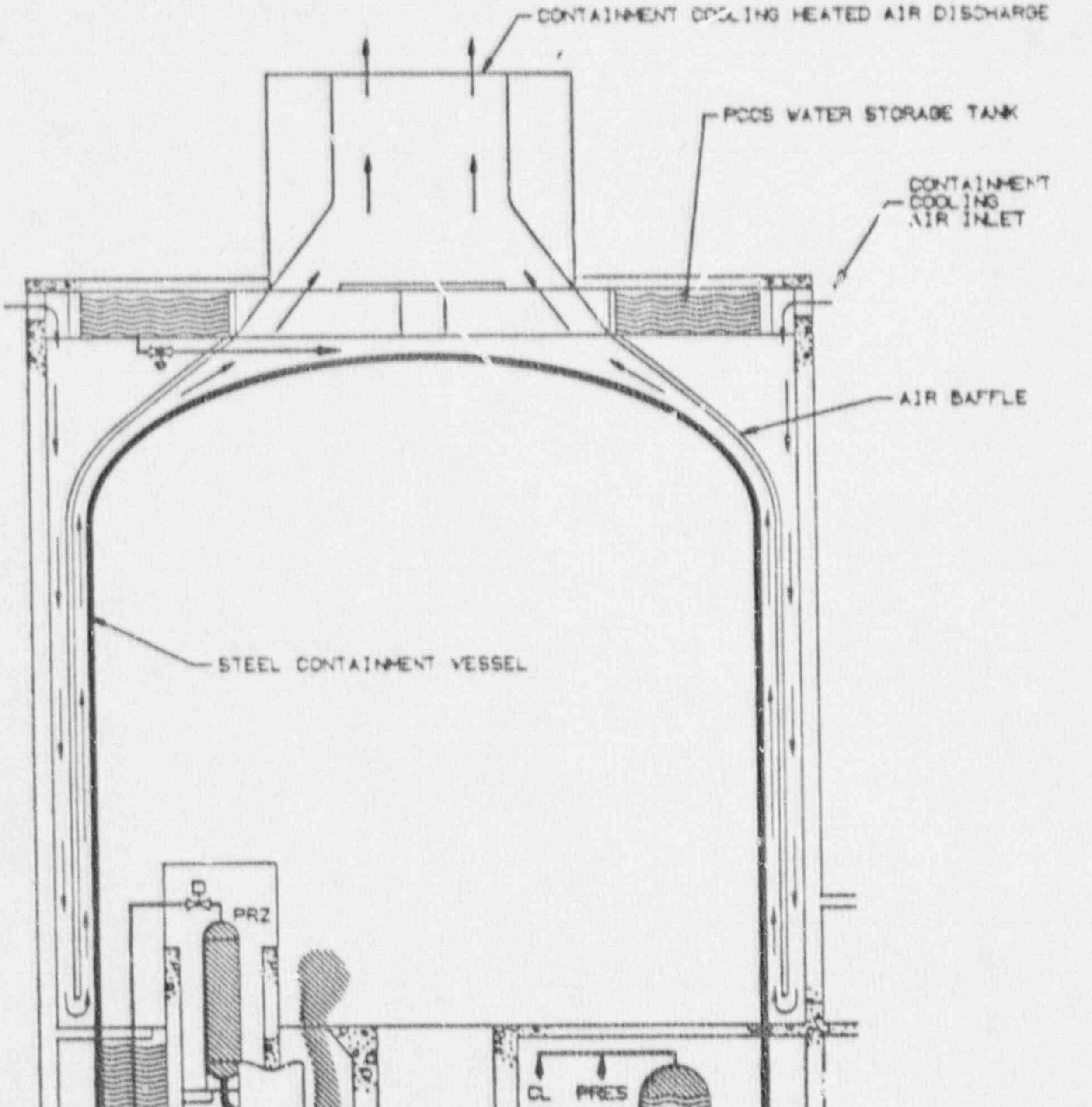
WESTINGHOUSE - 9/88

AP600 - PASSIVE SAFETY INJECTION SYSTEM

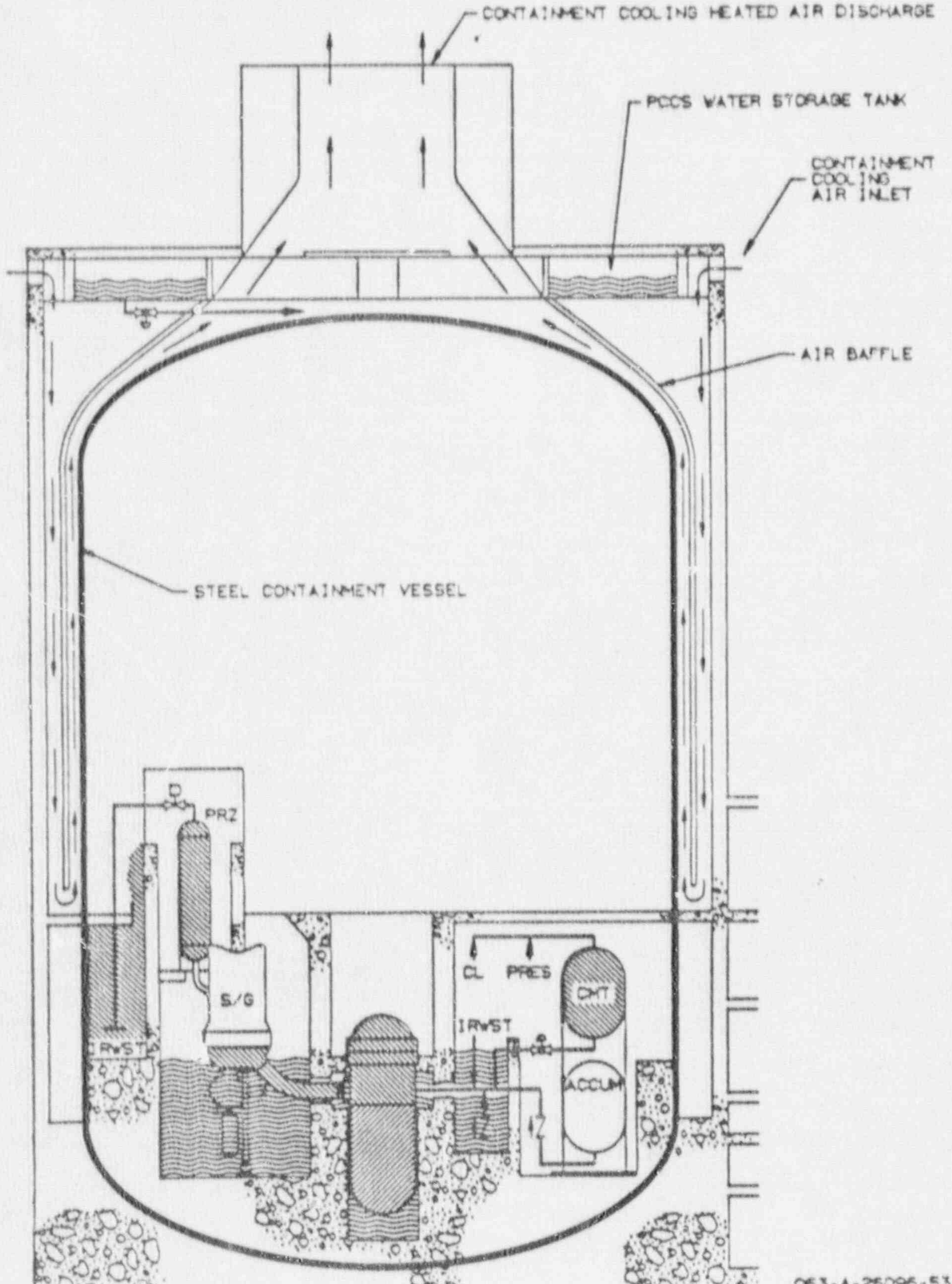


WESTINGHOUSE - 6/89

AP600 PASSIVE SAFETY INJECTION
LOCA INITIATION



AP600 PASSIVE SAFETY INJECTION
LOCA LONG TERM



AP600 - AUTOMATIC DEPRESSURIZATION DESIGN APPROACH

- o PROVIDE RELIABILITY
  - REDUNDANCY / DIVERSITY / SIMPLICITY
  - PROVEN VALVE DESIGNS; GATE OR GLOBE VALVE BODY WITH DC MOTOR OR AIR PISTON OPERATOR

- o MINIMIZE ACTUATIONS
  - GOAL OF LESS THAN ONE ACTUATION IN 100 YEARS
  - INADVERTENT USE REDUCED; 2/4 LOGIC, INTERLOCKS
  - REAL CHALLENGES REDUCED; NO RCP SEALS OR PRESSURIZER PORV, RELIABLE CVCS, GENEROUS CMT VOLUME

- o MINIMIZE CONSEQUENCES
  - GOAL TO RESTART IN 2 WEEKS AFTER EXPECTED ADS; IN LESS THAN 1 MONTH AFTER MAX FLOOD
  - DESIGN MAJOR RCS EQUIPMENT FOR TRANSIENT (FUEL, VESSEL, RCP, SG, ...) AS ASME SERVICE LEVEL 2 EVENT WITH 5 OCCURRENCES
  - PLANT FEATURES TO LIMIT FLOODING, CONTAINMENT HEATUP & PRESSURIZATION, & TO FACILITATE CLEANUP
  - MOST EQUIPMENT LOCATED ABOVE MAX FLOOD LEVEL OR DESIGNED FOR FLOODING

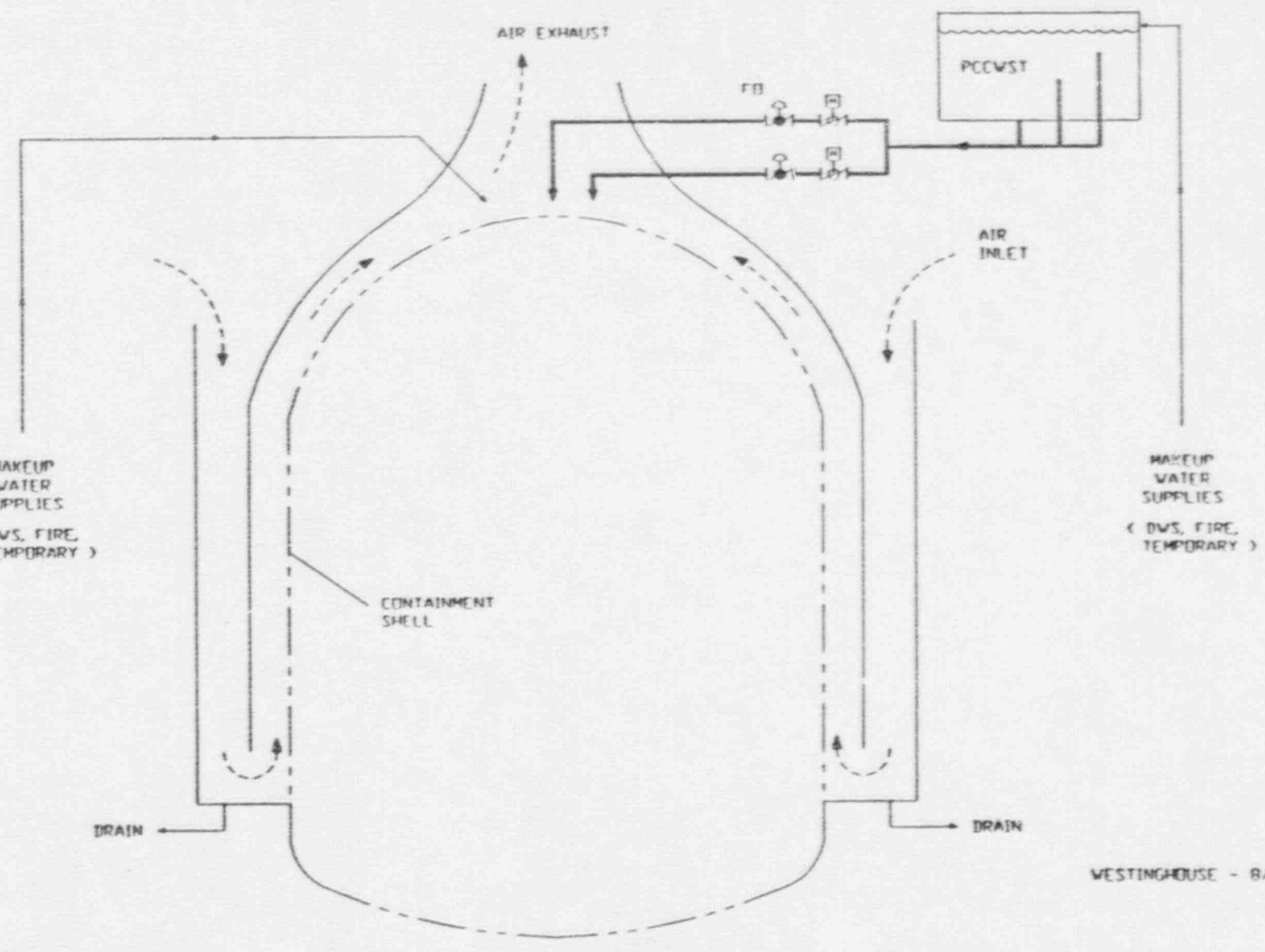
AP600 - CONTAINMENT HEAT REMOVAL

- o NORMAL OPERATION
  - FAN COOLERS COOLED BY CHILLED WATER (2 COILS, 4 FANS, NNS, BLACKOUT DIESEL)

- o TRANSIENTS (SAFETY)
  - IRWST ABSORBS DECAY HEAT FROM PRHR FOR ~2 HR BEFORE BOILING TO CONTAINEMNT
  - STEAM CONDENSES ON INSIDE OF CONTAINEMNT SHELL, MOST CONDENSATE DRAINS BACK TO IRWST
  - CONTAINEMNT SHELL COOLED BY NATURAL CIRCULATION AIR & GRAVITY DRAIN OF WATER FROM ELEVATED TANK

- o LOCA (SAFETY)
  - LARGE CONTAINEMNT VOLUME ABSORBS INITIAL MASS / ENERGY RELEASE
  - STEAM FROM LOCA & RCS DEPRES ENTERS CONTAINMENT, CONDENSES ON INSIDE OF CONTAINMENT SHELL, AND DRAINS BACK TO IRWST/SUMP
  - CONTAINMENT SHELL COLLED BY NATURAL CIRCULATION AIR & GRAVITY DRAIN OF WATER FROM ELEVATED TANK

AP600 – PASSIVE CONTAINMENT COOLING SYSTEM



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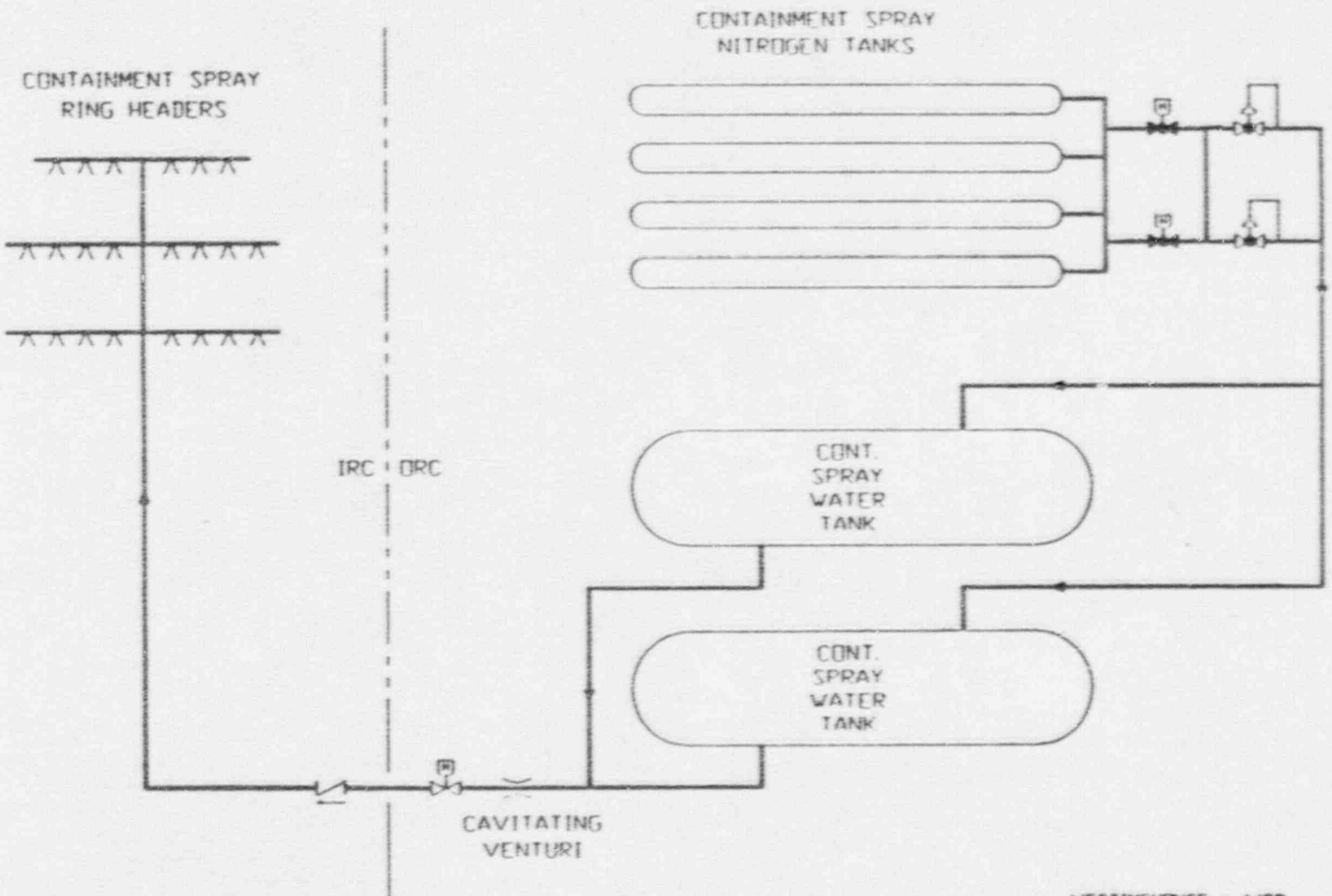
AP600 - PASSIVE CONTAINMENT SPRAY

- o ACCUMULATOR TYPE SYSTEM
  - COMPRESSED NITROGEN STORED IN SEPARATE GAS BOTTLES.
  - GAS ISOLATED FROM WATER BY REDUNDANT VALVES, DC POWERED MOTOR OPERATED.
  - PROVIDES 30 MIN SPRAY AT DESIGN FLOW, ANOTHER 10 MIN AT LOWER FLOWS.

- o ACTUATION TIMING
  - ACTUATED ON HIGH CONTAINMENT RADIATION (2/4) ALLOWS FOR DELAYED RELEASE.
  - SETPOINT WELL ABOVE LEVEL EQUIVALENT TO RELEASE OF GAP ACTIVITY.
  - NO CREDIT TAKEN FOR SPRAY IN CONTAINMENT PRESSURE CALCULATION

AP600 PASSIVE SAFETY INJECTION

(CONTAINMENT SPRAY FUNCTION)



WESTINGHOUSE - 1/89

AP600 - H2 CONTROL

- o DESIGN BASIS ACCIDENT
  - SLOW, LIMITED RELEASE OF H2.
  - TWO H2 RECOMBINERS WITH NON-SAFETY POWER PROVIDE RECOVERY CAPABILITY.
  - VENT VIA MINI PURGE PROVIDES BACKUP.

- o SEVERE ACCIDENT
  - RAPID, LARGE RELEASE OF H2.
  - 85% ZIRC-WATER REACTION WILL ONLY CAUSE A 13% FINAL H2 CONC IN THE CONTAINMENT.
  - H2 IGNITERS POWERED BY 1E BATTERIES PROTECT AGAINST LOCAL CONC AND LARGER ZIRC-WATER REACTIONS.
  - WILL REACESS IGNITERS PENDING FINAL EPRI REQUIREMENTS

AP600 - CONTROL ROOMS

o MAIN CONTROL ROOM

- PROVIDES MONITORING & CONTROL FOR NORMAL OPERATION, SAFE SHUTDOWN AND POST ACCIDENT

- 1E & SEISMIC EXCEPT FOR HVAC AND ELECTRICAL POWER

- POWER PROVIDED BY OFFSITE & ONSITE AC AND FOR 1/2 HOUR BY 1E BATTERIES; ALLOWS FOR RECOVERY OF AC POWER OR ORDERLY TRANSITION TO ECR

- HVAC HAS FILTERS FOR POST ACCIDENT HABILITY. AUTOMATIC 1E ISOLATION PROVIDED. NON-1E AC POWER AND COOLING WATER.

o TECHNICAL SUPPORT CENTER

- PROVIDES MONITORING FOR POST ACCIDENT

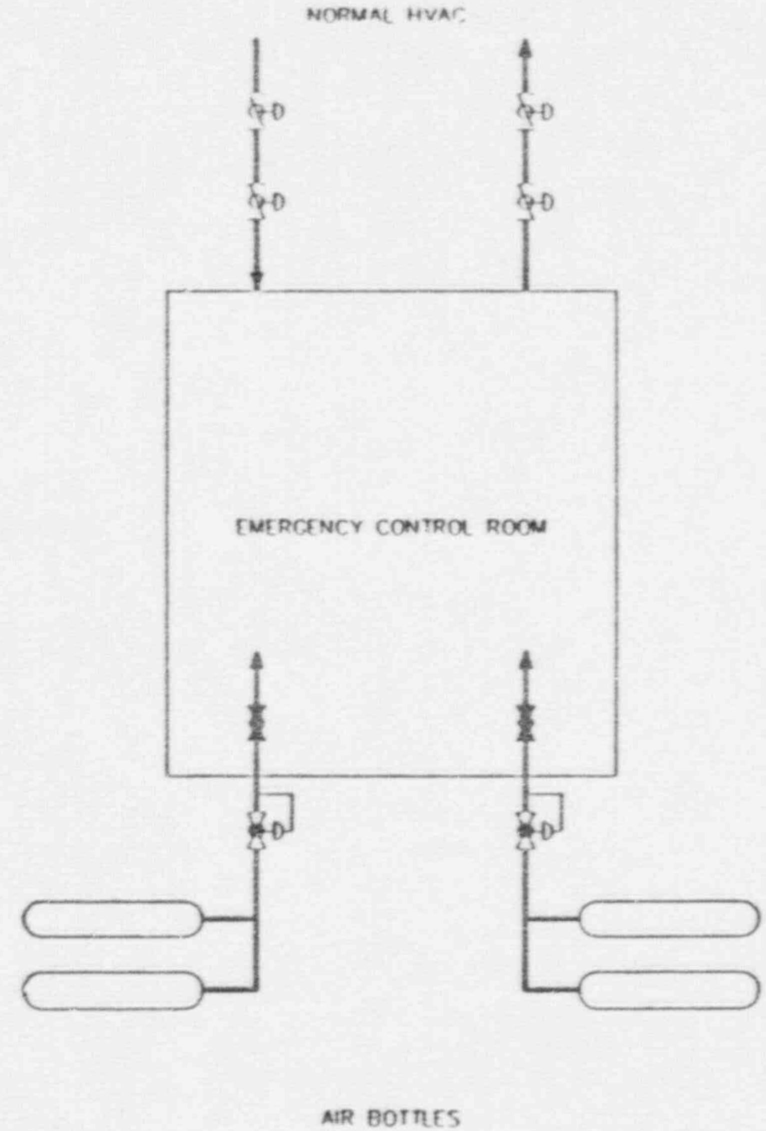
- NON-1E / NON-SEISMIC; HVAC HAS FILTERS TO REMOVE RADIATION, POWERED BY ONSITE AC POWER

o EMERGENCY CONTROL ROOM

- PROVIDES MONITORING & CONTROL FOR SAFE SHUTDOWN AND POST ACCIDENT

- COMPLETEY 1E / SEISMIC; 3 DAYS OPERATION WITH COMPRESSED AIR TO EXCLUDE RADIATION, CONCRETE WALLS FOR HEAT SINK, & BATTERIES FOR POWER

AP600 – EMERGENCY HVAC



AP600 - ELECTRICAL SYSTEMS

- o AC POWER SYSTEM
  - NON 1E
  - REDUNDANT BUSES TO POWER REDUNDANT NON SAFETY EQUIPMENT
  - TWO SEPARATE TIES TO OFFSITE POWER

- o ONSITE AC POWER SYSTEM
  - NON 1E
  - ONE DIESEL
  - AUTOMATIC START / LOAD EQUIPMENT ON ONE BUS. OTHER BUS CAN BE MANUALLY CONNECTED.
  - SUFFICIENT EQUIPMENT IS LOADED TO KEEP PLANT IN NORMAL HOT STANDBY AND TO ALLOW FOR SLOW COOLDOWN

- o DC POWER SYSTEM
  - SEPARATE 1E AND NON 1E BATTERIES
  - FOUR 1E BATTERIES SIZED FOR 72 HR WITH 4 WAY ELECTRICAL SEPARATION AND 2 WAY PHYSICAL SEPARATION
  - DESIGNED IN CONNECTIONS FOR TEMPORARY BATTERY CHARGER IF NEEDED AFTER 72 HR

AP600 - DESIGN SIMPLIFICATION

PLANT FEATURES	STD 2 LOOP	AP600
PUMPS - SAFETY	25	0
- NNS	188	139
HVAC FANS	52	27
HVAC FILTER UNITS	16	7
VALVES - NSSS (>2")	512	215
- BOP (>2")	2041	1530
PIPE - NSSS (>2")	44,300 FT	11,042 FT
- BOP (>2")	97,000 FT	67,000 FT
CONT PIPE PEN- TOTAL	93	48
OPEN	38	13
EVAPORATORS	2	0
DIESEL GENERATORS	2 (SC)	1 (NNS)
BLDG. VOL.- SEISMIC	9.4 MIL FT3	4.6 MIL FT3
- NON SEISMIC	6.2 MIL FT3	6.1 MIL FT3

Interest grows in the modular HTGR

By James Varley

Despite some recent setbacks, notably at Fort St Vrain and THTR-300, the high temperature reactor steadfastly refuses to die. The modular version, which is where worldwide effort is now concentrated, has, for example, been enjoying increasing support in the USA and proposals for public/private risk sharing in a lead project are currently being drawn up.

The investment to date in high temperature gas-cooled reactor (HTGR) technology has been put at around $5 billion. But as yet there is very little to show for it in the shape of operational power plants. With the final shut down of Public Service of Colorado's ill-fated Fort St Vrain unit in August after a rather brief and unsatisfactory career and the German THTR-300 due to be retired almost before it had really started up, there will be no HTGRs left in operation — the venerable 13MWe AVR experimental high temperature reactor at Jülich having closed down at the end of 1988.

Performance at both Fort St Vrain and THTR-300 has been well below what it should have been. Yet, surprisingly, morale in the HTGR community is by no means as low as might be expected and, worldwide, there seems to be a good deal going on, by way of design, development, planning, international agreements and feasibility studies — as evidenced by the 11th International HTGR Conference, held in Dimitrovgrad, USSR, 19 to 20 June.

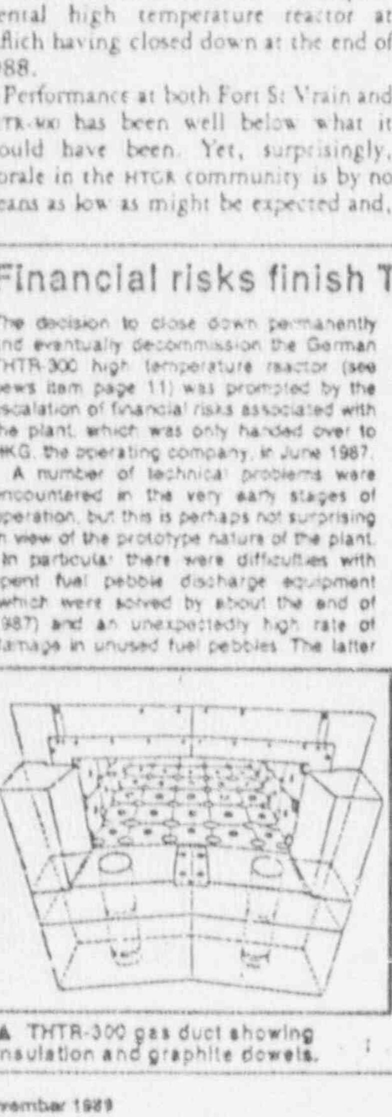
Proponents of the HTGR point to increasing interest prompted by environmental concerns coupled with a demand for a new generation of reactors with passive self limiting features. But perhaps the best recent news for HTGR people is that, all being well, the Japanese are actually going to start building one — albeit only 30 MWt. Following government approval of the budget, the Japanese aim to start construction, at the Oarai research establishment, in spring next year. The plan is to commemorate the event with a conference on HTGR technology, scheduled for 19-20 March.

Another important step forward was the creation in May 1989 of HTR GmbH, merging the HTGR activities of ABB with those of Siemens/KWU/Interatom and the subsequent signing of a memorandum of understanding with the USSR paving the way for a joint FR German—Soviet collaboration on a 200 MWt modular HTGR (the VGM) to be constructed at Dimitrovgrad (see *NEI*, December 1988). This would be a test unit initially operating at 750°C core outlet temperature for electricity cogeneration (80MWe) with only the steam generator and circulator in the loop. In the second phase it is planned to add an intermediate heat exchanger and raise the temperature to 950°C to demonstrate the production of process heat for the

Financial risks finish THTR-300

The decision to close down permanently and eventually decommission the German THTR-300 high temperature reactor (see news item page 11) was prompted by the escalation of financial risks associated with the plant, which was only handed over to HKG, the operating company, in June 1987.

A number of technical problems were encountered in the very early stages of operation, but this is perhaps not surprising in view of the prototype nature of the plant. In particular there were difficulties with spent fuel pebble discharge equipment (which were solved by about the end of 1987) and an unexpectedly high rate of damage in unused fuel pebbles. The latter damage turned out to be due to shutdown tests done during commissioning and once the plant was operating normally, a steadily declining damage rate was recorded.

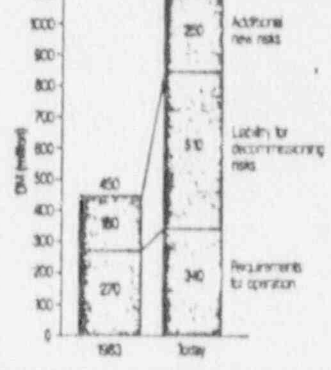


▲ THTR-300 gas duct showing insulation and graphite dowels.

September 1988, however, saw the start of an extended shut down when gas duct inspection revealed that the heads had broken off 35 of the 2500 insulation fixing bolts (subsequently found to be due to differential thermal expansion and irradiation embrittlement) and several graphite dowels had been displaced. Nevertheless the damage is not thought to have safety implications or be a bar to further operation.

Further difficulties arose in late 1988 when a reevaluation of the financial risks of continued operation was carried out. In 1983 a risk sharing contract had been drawn up in which the Federal and North Rhine Westphalia governments agreed to take on a liability of DM 450 million to cover the economic risks of operation and decommissioning — reflecting the plant's prototype status. As a result of the 1988 reevalation HKG estimated that this was no longer sufficient and called for governmental guarantees to be raised to DM1.1 billion. This risk escalation arises from uncertainties about fuel supply (due to cessation of fuel fabrication by Nukem following the TN bribes scandal and operating irregularities at Nukem facilities), inability to assure spent fuel storage capabilities (as required in the operating licence), the possibility of additional and as yet unknown requirements being imposed as a condition for obtaining a long term operating licence (the current licence being for 1000 full power days and due to expire in 1992), and a massive increase in the estimate for decommissioning (from DM 180 million to around DM 500 million).

The increased government guarantees were not forthcoming, resulting in HKG's announcement of its intention to shut down and decommission the plant.



▲ Escalation of financial risk guarantees needed by HKG to continue to operate THTR-300.

chemical industry and coal gasification.

HTR GmbH has also been busy in China, where agreement was reached earlier this year relating to the design and building of a small (10-20 MWt) modular HTGR, for test purposes, at the Institute of Nuclear Energy Technology, Tsinghua University, with construction scheduled to start at the end of 1991.

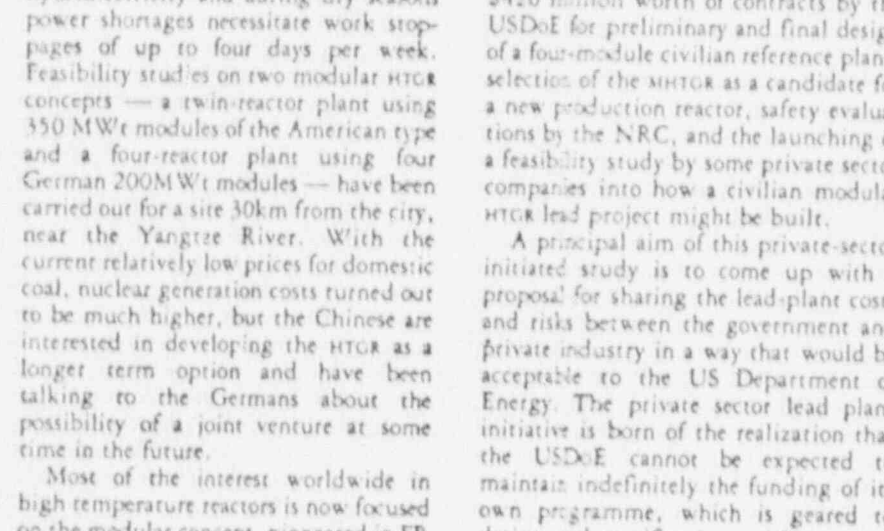
The Chinese have been looking at using the HTGR for enhanced oil recovery and cogeneration. A feasibility study into building a demonstration modular HTGR has also been done for the city of Zhongqing in south west China, which is located in a rapidly developing area. The city is 40 per cent dependent on hydroelectricity and during dry seasons power shortages necessitate work stoppages of up to four days per week. Feasibility studies on two modular HTGR concepts — a twin-reactor plant using 350 MWt modules of the American type and a four-reactor plant using four German 200MWt modules — have been carried out for a site 30km from the city, near the Yangtze River. With the current relatively low prices for domestic coal, nuclear generation costs turned out to be much higher, but the Chinese are interested in developing the HTGR as a longer term option and have been talking to the Germans about the possibility of a joint venture at some time in the future.

Most of the interest worldwide in high temperature reactors is now focused on the modular concept, pioneered in FR Germany, and in particular on the "side-by-side" variant — developed by KWU/Interatom — in which the core and steam generator are housed in separate vessels connected by a short run of pipework. The modular approach is characterized by small reactor sizes, a high degree of factory fabrication, and the use of steel vessels instead of the concrete vessels incorporated in the earlier "monolithic" designs. A number of modules may be installed in a single building to achieve the required power level for a given application.

US LEAD PROJECT

Since the mid-1980s, high temperature reactor development in the United States has been centred on the modular HTGR (MHTGR) and in the past year or so there have been some important developments, including the awarding of $420 million worth of contracts by the USDoE for preliminary and final design of a four-module civilian reference plant, selection of the MHTGR as a candidate for a new production reactor, safety evaluations by the NRC, and the launching of a feasibility study by some private sector companies into how a civilian modular HTGR lead project might be built.

A principal aim of this private-sector initiated study is to come up with a proposal for sharing the lead-plant costs and risks between the government and private industry in a way that would be acceptable to the US Department of Energy. The private sector lead plant initiative is born of the realization that the USDoE cannot be expected to maintain indefinitely the funding of its own programme, which is geared to design and certification, without some prospect of linking into a private sector project geared towards actually building a first plant. It is also recognised that substantial USDoE involvement would be needed for this lead plant.



▲ Simplified flow diagram for the civil modular HTGR being developed in the USA. (Diagram: General Atomics)

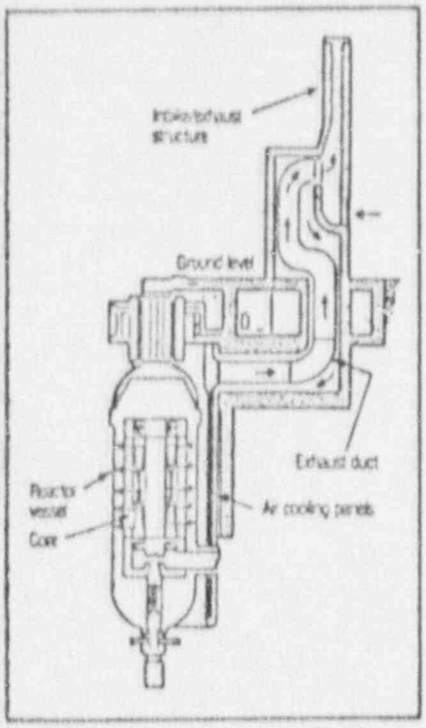
The study, which is being carried out by the utility Consumers Power, in conjunction with Gas-Cooled Reactor Associates (an organization representing American utilities with HTGR interests) and Philadelphia Electric (owner operator of Peach Bottom 1, the first HTGR in the USA), got underway in February 1989 and is being done in cooperation with General Atomics, Bechtel and Siemens/Interatom. It is scheduled to be complete by the end of this year and, as well as financial risk sharing, will also consider design requirements, licensing strategy, costing and scheduling. The study envisages that three main private-sector players would be involved in the lead project, a four-module plant:

- An operations company, led by Consumers Power, to operate, maintain and decommission the plant.
- An investor-owned generating company, which would own the plant and be co-licensee with the operator. Shareholders could include utilities, power plant vendors and, possibly, independent power producers.
- A power plant vendor company, initially consisting of General Atomics, Bechtel and Siemens/HTR GmbH.

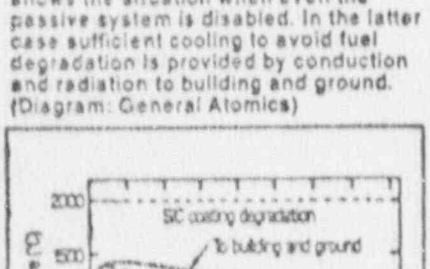
USDoE REFERENCE DESIGN

The plant that will eventually be built should the lead project idea ever come to fruition is currently about one year into its preliminary design phase. This design work, for a reference plant consisting of four 350 MWt modules, is being funded by the USDoE through contracts placed with General Atomics, Combustion Engineering and Bechtel National for the nuclear island, and with Stone and Webster for balance of plant, including a subcontract to Combustion Engineering for plant control design support. The contracts are for preliminary and final design, with Final Design Approval scheduled for 1996. In addition there is a continuing USDoE funded programme on supporting technology (covering materials, fuel and fission product experiments), for which the primary contractor is Oak Ridge National Laboratory.

But everything hinges on a continuing high level of Congressional support. The reference plant design contracts already referred to (amounting to $240 million for General Atomics and some $60 million each for Bechtel, Combustion Engineering and Stone and Webster) are conditional on availability of funds, with an appropriation being required each year from Congress. To carry the MHTGR through to final design and NRC certification, the MHTGR



▲ Passive reactor cavity cooling for the US civilian modular HTGR, which would be installed in a below-ground silo. The totally passive cavity cooling system provides a sink for residual heat if all forced cooling (both main circulator and shutdown circulator) is lost. But even if this passive air heat exchanger is lost, fuel does not reach its degradation temperature. The graph below shows best estimate core temperature in a case where all forced cooling is lost and there is no pressure in the system. The lower curve shows estimated core temperatures with the passive heat exchanger working, while the top curve shows the situation when even the passive system is disabled. In the latter case sufficient cooling to avoid fuel degradation is provided by conduction and radiation to building and ground. (Diagram: General Atomics)



contractors will be looking to substantial increases in Congressional appropriations over the next few years.

One encouraging sign is that in FY 1990 the USDoE's civil MHTGR budget had its first real increase for five years, rising to $25 million — $5 million above that for the previous year. A further considerable boost has come from the selection of MHTGR technology for the new production reactor to be built at Idaho. Through this channel the MHTGR has effectively attracted a further $120 million from Congress for FY 1990 in the form of funding for the production reactor project. A large element of the support for the MHTGR production reactor is based on perceived expected benefits to the civilian programme — through common development programmes, shared construction and operation experience and mutual infrastructure development.

IMPASSE ON CONTAINMENT

As it turns out, however, the USDoE's selection of the MHTGR for production use, as well as creating public relations problems by linking the technology into the weapons programme, has also saddled the civil MHTGR with an unforeseen regulatory difficulty.

The problem stems from the fact that the design of MHTGR that USDoE selected for its production reactor includes a containment structure (which was provided in response to a DoE requirement for a containment on all proposed production reactor concepts). The use of a containment goes against the fundamental tenets of the safety approach usually adopted in high temperature gas-cooled reactor technology, which is founded on the retention of radionuclides within the ceramic-coated fuel particles under all accident conditions.

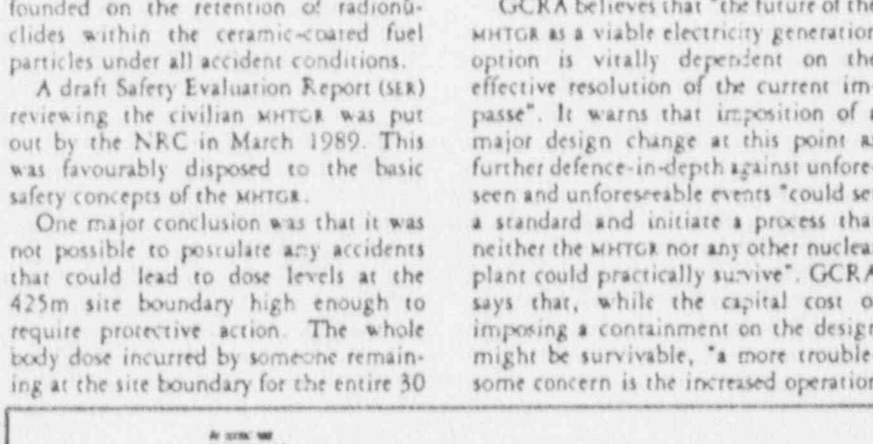
A draft Safety Evaluation Report (SER) reviewing the civilian MHTGR was put out by the NRC in March 1989. This was favourably disposed to the basic safety concepts of the MHTGR.

One major conclusion was that it was not possible to postulate any accidents that could lead to dose levels at the 425m site boundary high enough to require protective action. The whole body dose incurred by someone remaining at the site boundary for the entire 30 day duration of the worst case accident was estimated to be roughly equivalent to natural background.

Another important conclusion was that there do indeed appear to be no possible mechanisms under any accident conditions that could lead to fuel overheating and failure. This means that in the case of the MHTGR, the accident source term is limited to the percentage of defective fuel that would be expected during normal operation — ie it would be small and can be determined fairly precisely. In the jargon the MHTGR is said to have a "mechanistic source term". With an LWR, on the other hand, it is very difficult to say what fraction of the fuel inventory might leave it under severe accident conditions and a non-mechanistic source term approach has to be used.

Nevertheless, despite these strong endorsements, the NRC is not prepared to issue the all-important final SER (which would constitute a firm regulatory foundation for the MHTGR giving the programme participants confidence on the eventual licensability of their technology) until it has received from the USDoE an adequate explanation of the basis for "apparently different containment positions on two similar MHTGR designs" — ie the military and civilian versions.

GCRA believes that "the future of the MHTGR as a viable electricity generation option is vitally dependent on the effective resolution of the current impasse". It warns that imposition of a major design change at this point as further defence-in-depth against unforeseen and unforeseeable events "could set a standard and initiate a process that neither the MHTGR nor any other nuclear plant could practically survive". GCRA says that, while the capital cost of imposing a containment on the design might be survivable, "a more troublesome concern is the increased operation



▲ The military version of the US modular HTGR, with its containment concept, which has caused so much trouble for the civil version. (Diagram: General Atomics) and maintenance cost and risk arising from associated operational regulatory requirements".

USDoE is currently in the process of preparing to present to the NRC its case for having two containment concepts. One input into this has been a containment trade-off study carried out by General Atomics, Bechtel and Stone and Webster. The basic USDoE argument is likely to be that more uncertainty surrounds the operating conditions and technology to be deployed in the production reactor compared with the civil MHTGR, and its mission may change in the future, so it is considered prudent to include a containment from the outset to envelope all possibilities. There are no such uncertainties in the civil case and so a containment is not needed. It remains to be seen how the NRC responds to this line of reasoning.

WORLDWIDE STUDIES

Various studies relating to the safety and feasibility of the modular HTGR are also underway in FR Germany, the country that pioneered the concept, with a pronouncement from the Reactor Safety Committee scheduled for the autumn of 1989. But with capacity reserve margins high, load growth projections low, and a recent spate of well publicized and expensive problems, including those arising from the THTR-300 project itself, reducing the credibility of the nuclear industry to an all time low, the prospects for modular HTGR or any sort of nuclear power development range from bleak to non-existent — at least in the short to medium term.

Studies are also being carried out in a number of other countries on the modular HTGR, particularly as a source of process heat. In Indonesia, for example, a feasibility study has been done on using the modular HTGR for heavy oil recovery (and found it not to be economic at current oil prices), while in Czechoslovakia, where there are growing concerns about pollution from burning brown coal, the focus is on steam—methane reforming and coal gasification. Other CMEA countries have also expressed interest, including Bulgaria, Romania and Poland.

As already mentioned, the modular concept is now very much the dominant technology. But studies are still proceeding on two monolithic concrete-vessel designs, the HTR-500 under development in Germany (see *NEI*, September 1988) and the Soviet VG-400. The 1060MWt VG-400 has its intermediate heat exchangers and steam generator/main circulator assemblies housed in eight cavities within the concrete vessel wall.

Both the VG-400 and the VGM are part of an extensive Soviet effort in HTGR design and development directed towards eventual applications in process heat supply and cogeneration, which, according to information presented at the Dimitrovgrad conference, involves about 1000 professionals (some 130 Soviet delegates, representing 35 institutions were present at the meeting and the subsequent International Atomic Energy Agency Technical Committee Meeting on Gas-cooled Reactor Technology, Safety and Siting, held immediately after the conference itself).

At the opposite end of the HTGR size range from the monolithic plants is the 10-20MWt gas cooled district heating reactor, the GHR-10, which is being developed by ABB-Atom in collaboration with the Paul Scherrer Institute, Switzerland, and whose conceptual design has recently been completed.

In the high temperature reactor field there does indeed seem to be no shortage of conceptual designs — the challenge now is to find ways of moving down the long and difficult road towards commercial realization in the form of reliable, economic and environmentally benign power plants. □

A visit to the Kurchatov Institute

By M. T. Simnad

Work on the HTGR is given high priority at the USSR's Kurchatov Institute, with over 300 staff assigned to the programme. The combination of high temperature process heat and electricity generation, steam for municipal heat, high efficiency, and unique passive safety features is the motivation for the importance attached to the programme, explain Dr N. N. Ponomarev Stepnoi, First Deputy Director of the Institute, and his senior colleagues.

The Soviets have selected the German pebble bed core design rather than the prism design favoured by the US and Japan because of their ability to design and to irradiate the pebble bed fuel in the USSR, and joint Soviet—German efforts are underway with a view to building the first modular HTGR at the Lenin Institute, Dimitrovgrad. In contrast with the Germans, the USA has restricted the transfer of HTGR technology and thereby has limited the exchange of essential core design data. There is however much to be gained for both the US and the Soviet Union in an exchange of information on HTGR techology. A reappraisal of the US policy on this subject would be a timely and constructive move.

Research and development on the HTGR has been in progress for about ten years at the Kurchatov Institute. Facilities there include two pebble bed type HTGR critical facilities. These units are fully operational and provide the reactor physics data for the 200MWt and the 1060MWt pebble bed HTGRs that are under development in the USSR. The critical facilities are loaded with large numbers of 6 cm diameter graphite-matrix fuel balls (TRISO-coated 8 to 10 per cent enriched UO_2 particles are dispersed in an inner 3 cm diameter graphite matrix).

The Materials Test Reactor at the Kurchatov contains a helium gas loop that can be operated up to 900°C peak temperature at pressures of up to 100 atmospheres. It has been used for very extensive irradiation testing of the TRISO-coated fuel particles, the fuel balls, and graphite bodies. The uranium dioxide fuel particles are produced by a sol gel process at another Institute in Podolsk. The TRISO-coating process has been mastered, including more advanced coatings such as five-layer coatings that have a silicon-alloyed pyrolytic carbon layer. Fission-product releases as low as 10^{-4} to 10^{-7} R/B (release/birth) at 1200°C have been achieved, which compares with the best that has been attained anywhere else. This augers well for maintaining an extremely clean primary coolant circuit in the modular HTGRs. Some very impressive work is also in progress at the Kurchatov on development of pyrocarbon-bonded graphite composites for the pebble fuel matrix material and for the core graphite components to improve their radiation stability.

Computer tomography is used to examine the fuel balls in the pre- and post-irradiation investigations of fuels. Uranium migration is reported to be lower in coated fuel particles with uranium dioxide kernels compared with oxycarbide or alloyed oxide kernels. Burnups of up to 35 per cent FIMA have been achieved in the in-pile helium loop tests, in which the graphite corrosion rates have also been determined. The behaviour of the fuel under thermal cycling and in a loss of coolant circulation condition has been examined. Burnup determinations are also made by measurements of spontaneous neutron emissions from curium-242. The distribution of fuel and fission products in the coated particles is determined by laser layer-by-layer skinning followed by track emulsion (for fuel and fission products) and gamma spectrometry (for fission products). These tests have been carried out on coated particles irradiated to burnups of up to 17 per cent FIMA at temperatures of 1000 to 1860°C. Diffusion coefficients of caesium in the kernels and the coatings have been estimated from these concentration profiles. Graphite irradiation tests have also been carried out.

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FOCUS ON THE USA

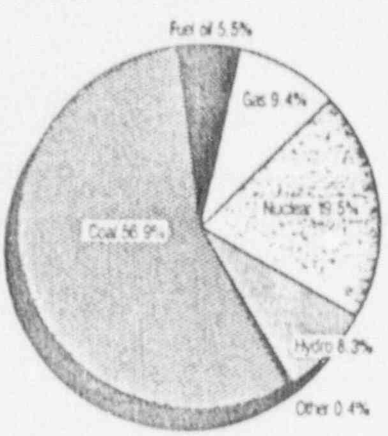
Where US nuclear energy stands in the world – a digest of statistics

The United States has far more nuclear stations in operation and has produced far more nuclear generated electricity than any other country.

However, the last US order for a reactor not subsequently cancelled, Callaway 1, was placed in 1973, and only four reactors (totalling 4753MWe) with definite projected operating dates are still under construction. These are Bellefonte 1 and 2 (1263MWe PWRs; 89 per cent and 58 per cent complete respectively, with planned commercial operation January 1994 and January 1996), Comanche Peak 1 (1150MWe PWR; 98 per cent complete, planned commercial operation December 1989), and Watts Bar 1 (1218MWe PWR; close to 100 per cent complete, with planned commercial operation October 1991). There are five reactors (totalling 5842MWe) in varying stages of construction that have unspecified completion dates (Comanche Peak 2, Perry 2 and Watts Bar 2) and two (WPPSS 1 and 3) that have been delayed indefinitely. But with nuclear programmes in most other Western countries stalled or severely cut back, the United States will still in 2000 have a clear lead in capacity over its nearest rivals, France and Japan.

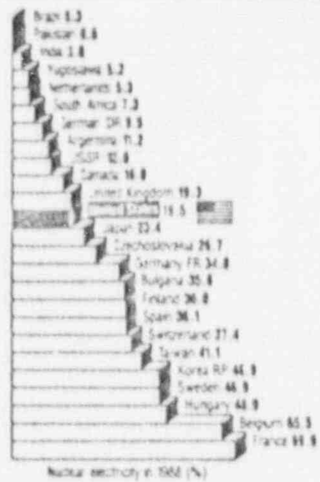
This dominance in size of the US nuclear programme has not been reflected in the average performance. The average has been brought down by a few stations with very poor figures. However, performance is getting steadily better. Some 13 US stations had load factors in 1988 of over 86 per cent, and nine of these had load factors of over 90 per cent.

FUEL MIX



▲ Sources of US electricity in 1988 (from Edison Electric Institute data).

CAPACITY AND GENERATION

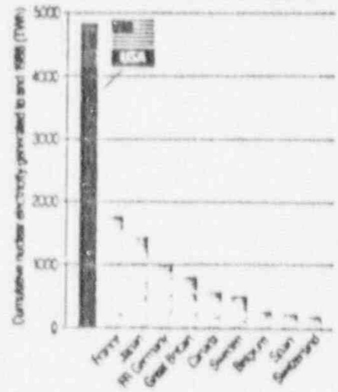


▲ Nuclear contribution to 1988 electricity generation.

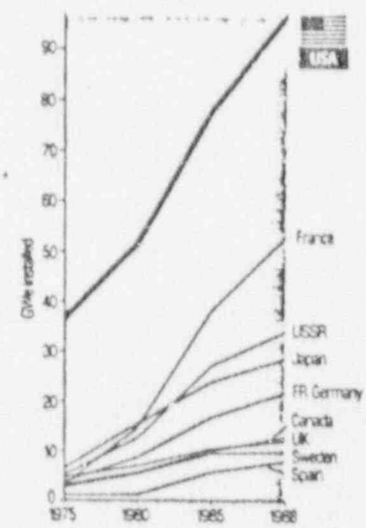
The nuclear share in some individual states of the USA (as given in Edison Electric Institute figures) is much higher than the average US level, for example Vermont 81.5%, South Carolina 62.5%, Connecticut 61.2%, New Jersey 59.2%, Illinois 56.1%, Maine 52.6%, Virginia 46.6%, Mississippi 38.2%, Arizona 37.3%, North Carolina 37.2%. (*Source: Edison Electric Institute*)

The world's top ten reactors in terms of lifetime electricity production to end 1988

Reactor name	Country	TWh generated
Biblis A	D	102.2 (14.5y in op)
Unterweser	D	94.13 (10.3y)
Biblis B	D	89.59 (12.9y)
Zion 1	USA	84.72 (15.5y)
Zion 2	USA	84.21 (15.1y)
Tihange 1	B	84.20 (13.1y)
Cook 1	USA	83.92 (13.1y)
Conn Yankee	USA	82.97 (20.5y)
Oconee 1	USA	82.90 (15.8y)
Stade	D	82.10 (17.0y)

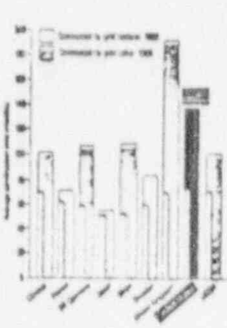


▲ Cumulative nuclear generation to end 1988.

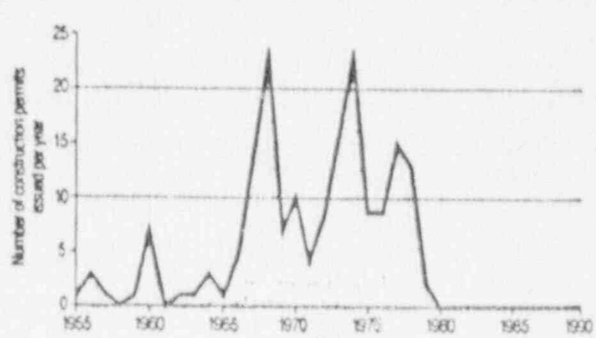


▲ Nuclear installed capacity growth (net GWe).

CONSTRUCTION

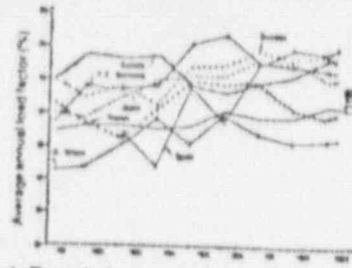


▲ Nuclear plant construction times (source: IAEA).

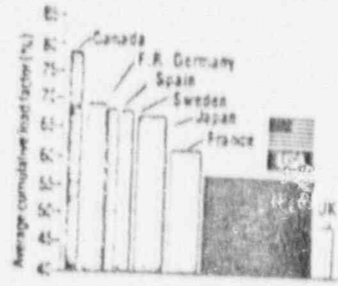


▲ Rise and fall in US construction activity as measured by permits/year (source: CEA).

PERFORMANCE



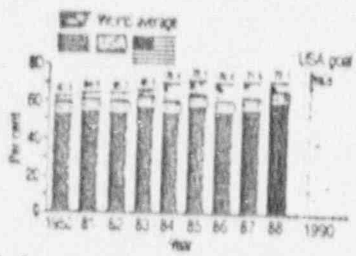
▲ Trends in average annual load factor.



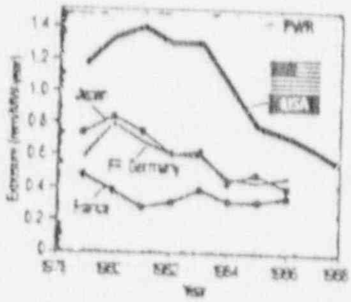
▲ Cumulative average load factors (width of bars correspond to cumulative electricity generated).

Top ten US nuclear plants in terms of lifetime load factor

Unit	Lifetime load factor	Ranking USA	Ranking World	First power
Palo Verde 3	88.2	1	3	12 87
St Lucie 2	82.6	2	25	6 83
Farley 2	80.4	3	34	5 81
Prairie Isl 2	79.3	4	38	12 74
Point Beach 2	79.1	5	41	8 72
Kewaunee	77.1	6	55	3 74
Beaver Valley 2	76.5	7	59	8 87
Callaway 1	75.8	8	63	10 84
Prairie Isl 1	75.7	9	65	12 73
Calvert Cliffs 2	75.1	10	72	12 76



▲ Average availability (source: IAEA and Inpo).

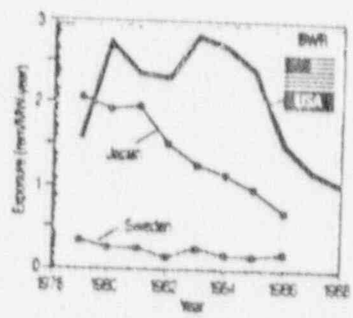


Top ten US plants in terms of 1988 load factor

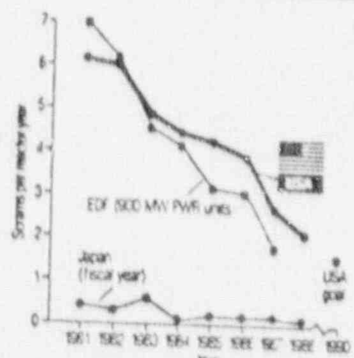
	Load factor	US ranking	World ranking
St Lucie	99.7	1	2
Farley 2	98.3	2	4
Millstone 1	95.5	3	8
North Anna 2	95.5	4	9
Monticello	93.5	5	11
Oconee 1	92.6	6	14
Palo Verde 3	92.6	7	15
San Onofre 2	91.1	8	17
Vermont Yankee	90.3	9	21
Callaway 1	89.1	10	25

US utilities operating three or more nuclear plants

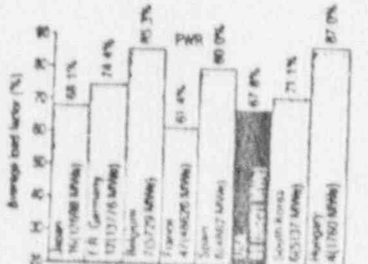
Operator	No of units	Average annual load factor 1988	Average lifetime load factor to end 1988	Total MWe	Reactor years
Com Ed	11	62.8	54.9	11271	117.6
Duke	7	75.6	61.5	7638	64.2
TVA	5	7.1	37.9	5660	56.8
VP	4	64.7	60.7	3696	51.2
CPL	4	59.2	53.7	3397	46.1
FPL	4	73.0	67.8	3240	49.7
NE Util	3	81.9	68.7	2601	33.9
Phil Elec	3	26.1	51.4	3444	32.7
Ariz PS	3	71.7	66.0	4006	7.2
S Cal Ed	3	62.6	56.4	2818	32.5
NSP	3	87.0	74.3	1702	46.8
PSEG	3	68.9	57.2	3510	22.1

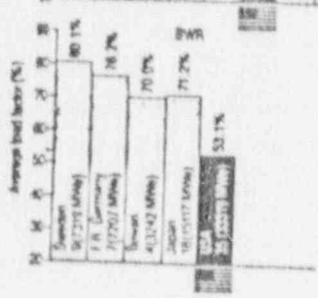


▲ Trends in occupational radiation exposure.



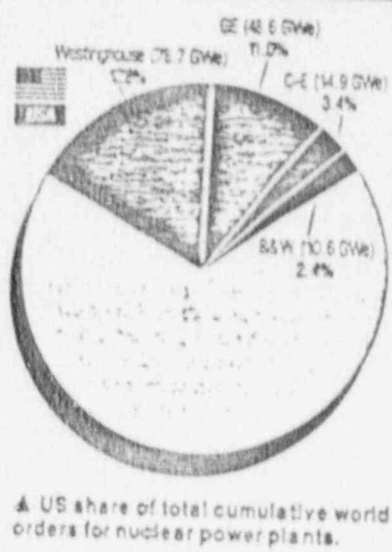
▲ Trends in reactor scrams (per reactor year).



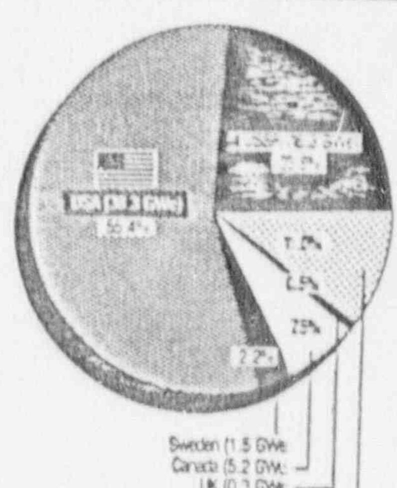


▲ PWR and BWR load factors in 1988 (each bar shows numbers of reactors and installed MWe).

ORDERS AND EXPORTS



▲ US share of total cumulative world orders for nuclear power plants.



▲ US share of world reactor exports (cumulative).

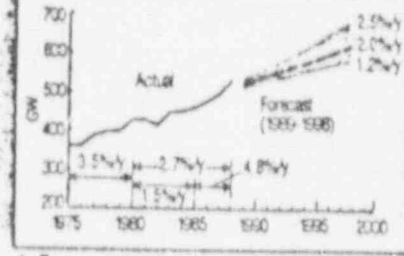
Will an American utility ever order another nuclear power station?

Visitors to the United States from other countries with nuclear industries have over the past ten years been repeatedly asking whether a US utility will ever order another nuclear power station. If so when? With continuing public opposition, regulatory problems and construction delays have appeared intractable and assurances from industry spokesman that the nuclear option would not be lost, that the public would eventually come to see the advantages of nuclear power, no more than wishful thinking.

Is the perspective any clearer today? The US Nuclear Regulatory Commission (NRC) has recently introduced a new single combined licence concept for future nuclear plants, aimed at bringing stability and predictability to the licensing process. The backlog of construction has been worked through and in many parts of the country the previous large margins of spare capacity have been eroded. The performance of nuclear stations has been improved. Despite heavy spending by utilities on efficiency programmes, electricity demand is rising much faster than new capacity to meet it is being installed, and there is a return to dependence on imported oil for electricity generation.

▲ Projected average annual electricity demand growth rates for regions of the USA, 1989-98 (source NERC).

Furthermore, in an important new initiative, the US Department of Energy (DoE) is holding a series of public hearings nationwide aimed at developing a National Energy Strategy that would reconcile energy and environmental goals. The aim is to put a draft energy plan to Congress next April and to adopt a final version by December 1990. The plan is expected to include recommendations for legislation, regulations and policy. The US Council for Energy Awareness (USCEA) reports that 80 per cent of Americans believe that nuclear energy should play an important role in the DoE's strategy.



▲ Forecast growth rates (%/y) for US summer peak demand (source NERC).

If this indicates a turning of the tide, there is as yet no sign of a utility preparing to place a new nuclear order or that one of the many cancelled and delayed projects is to be reinstated. Utilities and investors still consider there are too many uncertainties surrounding the large financial commitment involved in a nuclear plant.

Is the situation about to change? We present a series of articles looking at various aspects of this question.

Reducing the uncertainties

By James J. O'Connor

We do not currently anticipate any new power plant construction in our company's ten-year planning horizon. But this year has seen a development at the federal government level that is a step in the right direction towards restoring nuclear power as a viable alternative for generating electricity. I refer to the NRC's standardization rule (Part 52) modifying the two-step licensing requirement for new nuclear generating stations.

Under the old system, after a construction permit was issued and the plant was built, an operating permit, also requiring public participation, would then be required. The United States was the only country in the world with utilities that would build power plants under such conditions. As might have been expected, as operation of newly completed power plants languished year after year in public licensing hearings, hundreds of millions of dollars were lost by utilities and their investors. Although we feel the recent NRC rules changes should have gone further to assure the resolution of all siting and safety issues prior to the commencement of plant construction, at least the problem has been recognized.

Federal regulatory licensing, however, is not the only problem. Reducing the current considerable uncertainty surrounding construction of a nuclear plant in the United States is an essential key to reviving the industry. Before any utility chairman would again ask his board to commit billions of dollars, he must be able to demonstrate that the venture will be brought to fruition.

Up until the last ten years, the financial and investment community recognized that in the United States a traditional compact existed between utilities and state regulatory agencies. A utility could build a power plant and rely on receiving a fair rate of return if they built it at a reasonable cost. Those days are clearly over. In its place is an adversarial relationship that many utilities have interpreted as "You shall not build capacity at any cost."

This has led to the prospect of shortages of electrical capacity in some areas of the country. The supply of electricity in the Northeast and Middle Atlantic states is very tight. Even here in the Midwest, Wisconsin really has no capacity over and above its peak demand. For the short term, utilities will undoubtedly try to manage with purchases from third-party suppliers, with aggressive demand-side management, and with increases in the use of oil and gas.

The major sacrifices that could soon be required of industry, commerce and residential customers if new plants are not soon built should cause state regulatory agencies to take notice. The problem should become even more acute as the older generating equipment reaches the end of its useful life. More long-term solutions will have to be found. New baseload generating facilities will be required.

Another element is the heightened awareness of the environmental problems associated with burning fossil fuels. A typical 2000MWe coal-burning central station, for example, produces over 500 000t of flyash and sludge a year that must be disposed of. This is in comparison to about 50t a year of spent nuclear fuel, capable of being repro-

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Regulatory reform: utility and investor opinion analysed

Earlier this year two reports' prepared for the US DoE were published which contain the reactions of chief executives from nuclear utilities and representatives of the financial community to the NRC's licensing reform proposals.

Utility executives stressed that while the licensing reforms were crucial they did not constitute the sole criterion upon which any future determination to construct nuclear facilities would be made. The rate regulatory environment was also a major disincentive.

The main conclusions from the utility survey were first that the nuclear option is not "dead", but fundamental measures must be taken to improve its condition. Second, licensing reform and standardized design will not by themselves completely revive interest in nuclear power. Third, there is a strong degree of inter-relationship among a number of the reform options. Fourth, there is some scepticism that the necessary licensing reform can be achieved. Fifth, while all of the respondents felt that licensing reform is essential, a large number of utility executives found issues other than licensing reform also vital to the revival of nuclear power.

One major licensing issue that serves as a barrier to nuclear plants is the possibility of a post-construction public hearing. Seventy per cent of the respondents said they would not order a plant, or it is unlikely that a plant would be ordered, if the possibility of such a hearing exists, even if the scope of issues which could be raised was very limited.

Although a number of utilities indicated that NRC's recent actions and President Reagan's executive order concerning off-site emergency planning all tend to help remove some degree of uncertainty, it is also clear that Congressional action is either most desirable or essential in the minds of many. In addition to the permanance of law, Congressional action would give licensing reforms, several respondents indicated the need for an expression of society's commitment and felt that such an expression can only come from the Congress.

Half of the executives surveyed mentioned economic regulation or the perfect hindsight employed during prudency reviews as a major barrier to the construction not just of nuclear plants but also of any capital intensive technology. Public education and building public support for nuclear pwer were viewed as critical by half of the respondents.

At present, the report says, "powerful disincentives to further investment in nuclear power exist, while there are few, if any, incentives. If the nuclear option is to be revived, it is apparent that, at a minimum, significant changes must be made in how it is regulated, both by the federal government and by the various states."

INVESTORS VIEW

The report analysing the licensing reforms "from the perspective of the financial markets and potential investors" says that although the NRC's actions resolve the majority of the deficiencies in the licensing process the problem is that they are not legislation. The NRC rulemaking, it concludes, "is not sufficient to create a climate conducive to an investment in a new nuclear power plant absent Congressional endorsement of the policies it contains".

Without legislation there is a danger that a future Commission could issue a rulemaking that woud emscerate the fundamental provisions and protections in the current rulemaking. Congress would not necessarily need to provide any additional incentives or guarantees, but "federal legislation and administration endorsement of the NRC's action would send the correct signal to the market place".

Concern is also expressed that without legislation to ensure there is only one public hearing, NRC's interpretation of the Atomic Energy Act could face legal challenge.

The report recognizes that the NRC's backfit rule "has set a reasonable precedent for determining the viability, necessity and effectiveness of potential retroactive design changes. In addition, the nuclear industry's developments in pursuit of standardized plants and the establishment of higher standards of operation and training will enhance investor confidence".

It is emphasized that licensing reform is only one aspect of importance to the financial community in determining whether to back a nuclear power project. "Whether investors would judge a nuclear power project an acceptable financial risk would also depend on a number of factors – some relating to general economic conditions, some to the electric utility industry generally, some to nuclear power specifically. Investors would, for example, consider the state of the economy, the political climate for nuclear energy, and the potential returns available from other investment opportunities".

The report suggests that the nuclear utilities could learn from other industries forced to cope with unusual risk, which have developed a number of methods for making their ventures acceptable to investors. These include segregation of risk-causing activities in a separate subsidiary; agreements whereby those responsible for construction accept specific risks or, through a turnkey contract, undertake to complete the entire construction process at a specified price; and leasing or purchase of service as an alternative to owning high-risk assets.

'*Analysis of nuclear licensing reform proposals from the perspective of utility chief executive officers*, Southern States Energy Board, March 1989 and *Nuclear licensing reform and the financial markets*, Touche Ross & Co and Science Concepts Inc, May 1989.

A. James O'Connor: "We have been pleased with our investment in nuclear power"

cessed, for a comparably sized nuclear plant.

There are also new concerns about atmospheric warming and the resulting global weather consequences from the 10 000 000t of carbon dioxide a year that this typical 2000MWe coal-burning plant would produce. Furthermore, if the station burned coal produced in our state of Illinois, about 330 000t of sulphur dioxide every year would be emitted into the air. The US government is close to enacting laws to limit dramatically the sulphur dioxide emissions from coal fired stations.

Here in northern Illinois, we have built 13 nuclear units and retired one since we began construction of Dresden 1, the first privately financed nuclear unit, in the late 1950s. We have been pleased with our investment in nuclear power. Today, this is one area of the United States where the demand for electricity can be met. In October of last year, our Braidwood unit 2 went into commercial operation, bringing to completion our construction programme, which currently represents 11.9 per cent of US nuclear capacity. For the first seven months of 1989, over 86 per cent of our generation came from nuclear power.

A US utility cannot ignore for long a fuel source of which the United States owns over 30 per cent of the world's reserves, is quarter the cost of coal, and contributes absolutely nothing to climatic change.

Financial risk still too great

By William S. Lee

The economy in the parts of North Carolina and South Carolina served by my company continues to grow, and we are formulating a 15-year plan to supply the electricity necessary to support that growth. We anticipate the need for additional base-load generation in the early 2000s, but additional nuclear capacity is not currently a part of that plan.

That decision is based on external circumstance rather than on our experience with nuclear. Our seven nuclear units have been excellent performers, and we consider them to be the clear economic and environmental choice for our customers.

With responsible standardization and licensing reform, nuclear units will again be part of our future generation plan. The current climate does not provide the confidence we and our investors need to assume the financial risk of a project as capital-intensive as a nuclear power plant.

On the other hand, a broader perspective on the role of this valuable resource may be emerging. A reassessment of nuclear's role has begun in this country, driven in part by environmental questions raised about fossil fuels as well as by a growing awareness of the economic importance of electricity.

In addition, with the formation in 1979 of the Institute of Nuclear Power Operations (INPO), the US nuclear industry made a commitment to achieving excellence in nuclear operation, maintenance and training; and we have made significant progress towards that end. Our success, both individually and collectively, requires that commitment by each of us, irrespective of national borders. The World Association of Nuclear Operators (WANO), formed in 1989 and modelled after INPO, is a giant step towards assuring that excellence worldwide.

Additionally, we have to address directly and forcefully the political and regulatory issues that stand in the way of renewed nuclear investment.

The absence of a political "solution" to the permanent disposal of high-level waste is no small obstacle to future nuclear development. The US government must act decisively to provide assurance – to the public and to utilities – that high-level waste will be disposed of safely and economically and in a timely manner. That decisive action is no less important for the disposal of low-level radioactive waste materials.

The regulatory process on both federal and state levels offers its own stumbling blocks. We are left without the needed assurance that a nuclear plant can be completed on schedule (and, therefore within budget); that even when completed and licensed, the plant will be allowed to operate; or that when operating, the plant's cost will be fully included in the price of electricity. Under those circumstances, investment capital would not be available on reasonable terms, effectively precluding new nuclear orders.

A reformed licensing process, with certification of standardized designs, prelicensing of sites, and issuance of a combined construction permit and operating licence, is fundamental to the consideration of new orders. The efforts to streamline the federal licensing process, notably with the revised NRC rule Part 52, are positive steps, but are insufficient to ensure a predictable, responsible, and timely licensing process.

A Bill Lee addressing the WANO Inaugural meeting in Moscow, May 1989 (photo: CEGB).

Return of the nuclear option in the United States should be accompanied by a commitment to close the fuel cycle. For nuclear to make the worldwide contribution that it must in the next century, we must recycle fuel. This will conserve uranium as well as reduce the volume of high-level waste.

When new orders are considered, it appears that the advanced light water reactor (ALWR) design is the only viable near-term option for us. Based on proven technology, the ALWR will give us improved operating and maintenance characteristics, enhancing nuclear's environmental advantages. Coupled with regulatory reform, the ALWR will give us stable and predictable economics. After a demonstration, we also believe the modular high temperature gas reactor (MHTGR) will offer significant promise.

A solution to the waste disposal problem; a streamlined and predictable licensing process; closing the fuel cycle: a simpler and better machine; a commitment to excellence – Quite a challenge!

But, a challenge very much worth the taking. Those of us who deal with the political, economic and environmental challenges of energy use and availability recognize how much more difficult the path becomes without nuclear. We have the resources and the leadership to move ahead. Nuclear belongs on our list; let's put it there.

William S. Lee is chairman and president, Duke Power Company 422 South Church St, Charlotte, NC 28242, USA.

Generating companies: their time has come – again

By Andrew C. Kadak

Utilities in the United States are changing. Forces driving this change are largely economic and political. By facing these economic and political realities, utilities may still be in a position to fulfil their mandate of supplying electricity on demand.

To do this utilities must be competitive. Utilities are judged on the price and reliability of their product – which is still electricity. Trends indicate that utilities are slowly evolving out of the generation business into the contracting for power, distribution and sales business because of the advantages of such arrangements for the ratepayer and the stockholder. This adaptation to a more competitive environment suggests that utilities must closely examine their core busniness.

If the core business is the reliable supply of competitively priced electricity, independent of how generated, utilities ought to be in a position to buy or generate the cheapest form of electricity. If, in order to fulfill this objective, utilities need to leave generation to companies whose only business is generation, not sales or distribution, that is the direction in which they should proceed. By not being distracted by other pressures, these generating companies can focus on maximizing generation at minimum cost.

This type of arrangement is particularly appropriate for nuclear power generation. The design, construction and, most important, the operation of nuclear plants is quite different from coal, oil or hydro power operation. The standards are different and demand a radically different culture. Safe and economic operation of nuclear power plants requires total dedication by staff and all levels of management from the line supervisor to the chief executive officer. The focus ▷

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of the organization should not be blurred by transmission and distribution problems, customer service departments, billings to residential consumers, handling customer complaints, restoration of power after storms, etc. The entire focus of the organization should be the production of electricity reliably, safely and economically.

The Yankee Atomic Electric Company was organized in 1954 by ten utilities in New England to design, build and operate the Yankee Rowe plant. These utilities had no intention of running the plant, but signed a contract to purchase the electricity generated. They purposely kept the Yankee organization strictly focused on nuclear power plant design, construction and operation.

This same principle was used by others to build Connecticut Yankee (Haddam Neck), Vermont Yankee and Maine Yankee. To this day, Yankee's only business is operating the Yankee Rowe plant and supporting the operation of other nuclear plants in New England. The operating record of the Yankee plants demonstrates the soundness of the original decision made 35 years ago.

Thus, although Yankee is considered a utility, it really is a specialized generating company whose only business is nuclear operation and support.

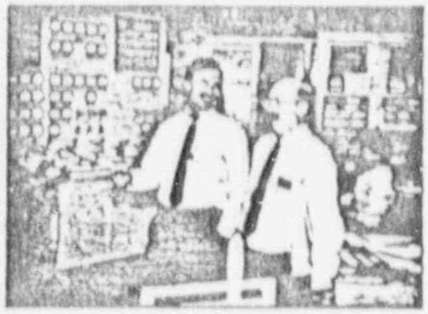
There are several reasons why the Yankee organization worked so well. First, with the focus strictly on nuclear, a dedicated, talented nuclear organization can be established along with a culture that demands attention to detail and safety. Both are crucial to reliable and economic operation. Alvin Weinberg, an early nuclear pioneer, used the words "nuclear priesthood" to describe the kind of skill and dedication required to manage nuclear technology. At Yankee, we do not consider ourselves "ordained", but our commitment is certainly total.

Second, because of this commitment, Yankee has invested in developing in-house skills, thereby avoiding reliance on outside contractors for engineering, licensing and operational support. We developed our own expertise with sophisticated accident analysis tools, probabilistic safety analyses, environmental science and laboratory capabilities, just to name a few.

To this day, Yankee is the only "utility" that has NRC licensed loss of coolant analysis methods. When something needs to be done or NRC issues new directives, we need only look to our internal staff to resolve or address the issue. This intimate knowledge of the plant yields reliable, safe and economic performance.

Finally, top management focus is essential to strong nuclear operations, a point that cannot be overemphasized. For instance, as president of Yankee, my priorities are clear and narrowly focused on nuclear electricity production. I do not have to worry about future electricity production. I do not have to worry about future electricity demand projections, implementation of conservation and load management programmes or where the next unit of capacity is going to come from.

By achieving a proven performance record, investor and utility confidence will return to the nuclear power option. There is no doubt



▲ Andrew Kadak (left), with plant superintendent, Normand St. Laurent in the Yankee Rowe control room.

that many problems of the nulcear industry are self-inflicted. These problems have to do with companies embarking on the nuclear power road without the experience or expertise to understand the technology and its demands.

Specialized generating companies have a proven track record of sustaining high levels of performance at nuclear power plants. Thus, since investors always seek reliable performance, it follows that they will take another look at the nuclear option when competent generating companies are responsible for construction and operation. Public and political support for nuclear power also depends on the perception of the competence of the companies operating the plants and the performance of the plants.

We know that a majority of the public believes that nuclear power must play a vital role in the future supply of electricity. This belief is strengthened by the environmental advantages of nuclear power which are becoming more apparent each day with the realization that nuclear power does not cause global warming, acid rain or air pollution. The public will not, however, accept poor performance despite all the positive advantages nuclear offers. Nor will the public accept high priced electricity. Single focus, high expertise nuclear generating companies offer the best potential for keeping costs down and performance up.

These arguments have been recognized in the US nuclear industry as traditional utilities, faced with increasing competition and regulatory pressures, reassess their positions. Many have already taken the steps to consolidate their nuclear organizations and separate them from their traditional utility business. Some have even gone so far as to spin off generating companies whose only business is nuclear. The pressures of the marketplace are making it happen.

Generating companies can also provide an inherent technical base for the next generation of nuclear plants and orders. If a utility or an independent power producer wants to build a nuclear plant, they could contract with the generating company to participate in the design, construction and operation of the new plant.

In my opinion, a likely scenario is that the next nuclear order in the United States will not be a so-called advanced reactor, but rather an improved, replicated plant with which there is already considerable experience. What generating companies offer for the future is the experience and knowledge to judge what will really work as designs appear on the drawing boards. Input from such companies will be sought long before these plants enter the demonstration stage. No longer will utilities be the insufficiently-informed buyers of the latest, greatest, technology. Basically, generating companies should be the buyers or builders, and then the operators of the next generation of nuclear plants.

In summary, generating companies offer the utility industry a chance to put the operation of their nuclear plants into a single focus organization that has demonstrated its performance to the investor, regulator, ratepayer and public. It is an idea that was born in 1954 with the Yankee Atomic Electric Company soon after President Eisenhower announced his Atoms for Peace initiative. It is an idea whose time has come again.

Inherent safety: the key to revitalization

By David P. Hoffman

Revitalization of the nuclear option may be the single most important contribution to America's future national energy and environmental well-being. However, no single utility can undertake another new plant project with the existing uncertainties in both the regulatory and public arenas.

Abandonments of partially completed nuclear plants and recent shutdowns of the Shoreham and Rancho Seco plants have largely been the result of public concern about the safety and economics of current nuclear technology. These concerns are why no new nuclear plants have been ordered by US utilities for more than ten years. This ongoing lack of public confidence in nuclear power has forced the US utility industry to abandon the nuclear option in trying to meet today's growing electricity demand. Unless public confidence in nuclear technology is restored, it will be another ten years, and maybe more, before another nuclear plant is ordered in this country. Equally important, rising public concern about the environment – and especially global warming – is an important reason to restore public confidence

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▲ David Hoffman: "national commitment" is needed to develop passively safe plants

in nuclear technology and revive the nuclear option.

A regulatory climate that reduces the burden or uncertainty during both the construction and operational phases of a plant life is a requirement. One-step licensing is an important ingredient to reduce the last-minute risk to investors. Operational regulations could be improved through standardized plant designs. Also, some form of government indemnification will be necessary to start the nuclear building process again – at least for advanced designs.

The only apparent way to revive commercial nuclear power is by developing an inherently (passively) safe nuclear plant. One that is insensitive to operator actions and where it is physically impossible to melt the fuel under any operating conditions. There is no disagreement among leading nuclear experts concerning the feasibility of such an undertaking. In fact, there are several companies in the US and Europe that are pursuing efforts along these lines, although at a modest rate. What is needed is a *national commitment* to accelerate and assure the success of such an effort. A new "safe nuclear energy" initiative with government support of initial projects could provide utilities with the confidence they need to again invest in an energy source which most feel is vital to the country's future.

Building the first new-generation plant probably does not fit into any single utility plan at this point, and most likely will not in the foreseeable future. Therefore, the initial plants would require a coalition of companies, possibly operating as an independent power producer, with sufficient government support for resultant profitability. Once these new plants prove themselves, the private utility sector will return to the nuclear option as the best choice to combat growing environmental concerns.

The real challenge: garnering public confidence

By Henry Lee

To some, the long time that has elapsed since any nuclear station was ordered in the United States represents a serious economic and energy problem, demanding the concerted attention of government and industry. At best, advocates of this position see the United States losing its competitive position as manufacturers of nuclear equipment. At worst, they see a spectre of inadequate and insecure electricity supplies – the price of which is held hostage by ever more stringent environmental regulations and an oil market controlled by the whims of an unpredictable cartel.

To others, though, this situation is the result of prudent decisions by a society concerned about the safety of nuclear plants in the wake of Chernobyl, worried about nuclear waste disposal, and in recent years appalled by both the capital and operating cost of these facilities.

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▲ Henry Lee: the nuclear industry must demonstrate "both technological competence – which it has – and strategic and political creativity – which it has not"

Nuclear advocates are quick to point out that the problems with nuclear power are exaggerated by a news media prone to sensationalism, by politicians who are short-sighted, and by a public which has been lulled into complacency by a decade of low priced energy. They argue that government should be more pro-active and not over responsive to the "emotions" of the public.

This view ignores political and economic reality. Does one really expect that the news media will suddenly see the light and pledge to refrain from sensationalism; that political officials facing elections every two to four years will espouse energy solutions which are both unpopular and perhaps unnecessary? Does one expect the public to decide that now is the time to immerse itself in the esoterics of nuclear engineering so that they too can conclude that this technology is in their best interest? The answer is clearly "No".

Does this mean that nuclear power is dead? The answer is also "No". The fact remains that nuclear fission is an incredibly efficient and clean method of producing energy and over the long term, society has no choice but to tap this resource. The real question is when will this take place and under what conditions.

I would suggest four factors which will contribute to the answer:

- Higher oil prices – as long as oil, as well as gas and coal, supplies are cheap and plentiful, the public will be reluctant to embrace a technology with which they are uneasy and which has overshot original cost projections by a factor of four.
- Increased environmental concerns – global warming and even local air pollution problems are apt to put increasing pressure on the cost of fossil fuels.
- The development of smaller and safer nuclear technologies – the former to placate the utility executive reluctant to risk large blocks of capital and the latter to assuage the public's concern over a repeat of Three Mile Island.
- A resolution of the nuclear waste disposal problem – while this is primarily a political, not a technical problem, the fact remains that we are about to enter our fifth decade without a solution.

Attempts by government to force-feed the nuclear option to reluctant localities and a sceptical public will only catalyze more opposition. History is replete with examples of paternalistic governments ignoring market forces and public will. In almost every instance, those efforts resulted in failure. On the other hand, market forces will change over time. Oil prices will not be low forever. The public is not unreasonable and will change its position if circumstances warrant.

The challenge to the nuclear community is to work to garner public confidence, and this can only be done if the industry demonstrates both technological competence – which it has – and strategic and political creativity – which it has not. How the industry fares in meeting this challenge will determine when and under what conditions utilities will again consider the nuclear option.

NOVEMBER 1989

nuclear engineering INTERNATIONAL

SBWR: SIMPLE IS BEAUTIFUL

- Wallchart of GE's small passive plant for the 1990s
- But will a US utility ever order another nuclear reactor?
- HTGR update
- Emergency preparedness
- Simulators, training, human factors
- Upgrading, repair, refurbishment

nuclear engineering
INTERNATIONAL

NOVEMBER 1989
Volume 34 No 424

Next month's issue

Developments in reprocessing technology: recycling the products of reprocessing. Core management. Instrumentation and control.

January: Steam generators: reliability, inspection, repair, replacement. Corrosion, materials degradation and assuring structural integrity. Seals, sealing systems and fasteners. Datafile: Finland.

February: Environmental monitoring. Radiation protection. Refurbishment, upgrading, plant life extension, spare parts procurement and outage services. Advances in filtration. Gas and liquid sampling.

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Framatome, B&W take aim at US services market

UNITED STATES: In a move that prompted intense speculation for weeks in advance, Framatome officially announced at a press conference in Washington, DC on 14 September that it has invested $50 million to become a partner in Babcock & Wilcox's nuclear services business. The two manufacturing giants will be equal partners in a new joint venture, B&W Nuclear Service Company.

The new firm will aim to compete for a share of the $2.5 billion/year US nuclear services market.

The two companies expect the new alliance to provide advantages neither enjoys on its own. For Framatome, which previously joined forces with B&W in 1987 to create B&W Fuel, the venture represents another step into the American market. Framatome gains a partner that can help open doors in the American services market.

For B&W, which has designed fewer US reactors than its competitors, the venture provides an infusion of capital and access to Framatome's state-of-the-art technologies and international experience. B&W believes these will strengthen its ability to serve a broader base of plants than simply those it designed.

The new venture is to start up formally in the fourth quarter of 1989. However, B&W Nuclear Service Co cannot market any technology developed jointly by Framatome and Westinghouse until a prior agreement, the Technology Cooperation Agreement, expires in 1992.

Jean Claude Leny, Framatome's chairman, said the move reflected his company's belief in the American nuclear market. "We wouldn't have invested $50 million if we had no confidence in the future of the nuclear industry in the United States," he said.

NEW REACTOR

In addition, Leny announced that Framatome and B&W will join forces, along with the KWU Group of Siemens AG, in a tripartite venture to evaluate development of a new reactor designed specifically for the North American market. B&W would, in effect, join Nuclear Power International, the partnership formed earlier this year by Framatome and Siemens to market reactors worldwide (*NEI*, June 1989).

Asked if the new design would be a small modular reactor, like Westinghouse's AP600 or General Electric's SBWR, Leny said details would not be available for several months. However, he drew a distinction between the global and American markets. In most parts of the world, interest remains in large pressurized water reactors of the order of 900-1400MWe, Leny said, but now, smaller designs with special safety characteristics would be developed if that is what is required to meet the needs of the American market.

The new alliances will enable B&W and Framatome to offer a full spectrum of products and services – reactors, fuel and services – to American utilities. The three operations will be separate joint ventures under an umbrella company, B&W Nuclear Technologies, headed by Charles Pryor, vice president and general manager. Currently, Pryor is head of B&W's nuclear services division.

B&W had been the only one of the four major American nuclear manufacturers that had not announced plans to consider a new reactor design for the 1990s. In addition to Westinghouse and GE, Combustion Engineering is cooperating with Rolls-Royce and the UK Atomic Energy Authority in developing the SIR reactor. Pryor says B&W's new tripartite venture with Nuclear Power International would need to "move quickly" to catch up with the others.

For the immediate future, however, B&W and Framatome will concentrate on the services market. On its own, B&W has provided services in the past to most American reactors and derives more than 35 per cent of its revenues from work on nuclear plants designed by other manufacturers, Pryor said. Recently B&W won contracts from Baltimore Gas & Electric to do repairs at the Calvert Cliffs plant; from South Carolina Electric & Gas for steam generator tube sleeving; and from Commonwealth Edison to service the Byron units. These plants were designed by others, Pryor noted. □

▲ (L or R) Jean Claude Leny, chairman and chief executive officer of Framatome; Charles W. Pryor, now vice-president and general manager of B&W Nuclear Service Co; and Robert E. Howson, chairman and chief executive officer of McDermott Intl.

Consumers power to replace SGs at Palisades

UNITED STATES: Consumers Power has announced it will replace, rather than repair, the two steam generators at its Palisades nuclear plant (780 MWe, PWR) at a cost of $100 million.

Consumers has scheduled the replacement for autumn 1990, when Palisades is to undergo its next refuelling. The work is expected to take about five months. To remove the existing steam generators and replace them with new ones, Bechtel Power, the prime contractor, will cut a hole in Palisades' prestressed concrete containment. It will mark the first time this will be tried in the US, although the KWU Group of

Siemens recently used this method while replacing three steam generators at the Ringhals plant in Sweden (*NEI*, September 1989).

Palisades, which has been operating since 1971, was the first PWR built in the US with a Combustion Engineering nuclear steam supply system. The replacement of the steam generators is not expected to have a significant impact on the company's proposal to sell Palisades to an independent power generating company formed by Consumers Power and Bechtel. Applications for approval of the sale are pending before the Federal Energy Regulatory Commission and the Nuclear Regulatory Commission. □

O&M costs: nuclear up, coal down

UNITED STATES: In 1988 the cost of producing electricity decreased at coal-fired plants and increased at nuclear plants, according to a recent report from US research company the Utility Data Institute. The report, which gives details of operation and maintenance (O&M) costs for power plants in the United States, showed that for the second year the average cost of power from coal plants was cheaper than that from nuclear plants.

The switch in power costs first happened in 1987, the report says, and is due to changes in both fuel and non-fuel costs.

Fuel cost. The cost of fuel at coal plants declined 0.055 cents/kWh in 1988, while the average cost of nuclear fuel increased by 0.016 cents/kWh. "Fuel represents 80 per cent of O&M costs at fossil fuel plants and less than 40 per cent at nuclear plants," UDI president Liz Hannon explained, "so any decrease in the cost of coal has a significant increase on total variable production costs."

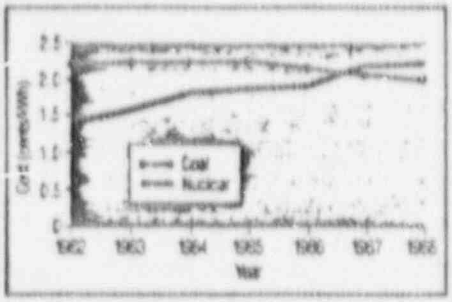
Non-fuel costs. Non-fuel O&M expenses at coal plants decreased, on average, by 0.01cents/kWh in 1988 while at nuclear plants they increased by an average of 0.017cents/kWh.

The UDI report, *1988 Production costs: operating steam-electric plants*, covers variable O&M costs for 423 coal fired plants, 73 nuclear plants, 204 gas-fired plants and 81 oil-fired plants. It does not include fixed expenses, such as taxes, interest and insurance, associated with power plant operation.

NUCLEAR O&M SPOTLIGHTED

Nuclear O&M costs were considered in more detail in another report, the *1988 Nuclear performance compendium* from US analyst Temple, Barker and Sloane.

The report says that nuclear plants in



▲ Average variable O&M expenses, including fuel, for US power plants (UDI).

the United States generated 15 per cent more electricity in 1988 than in the previous year, but operation and maintenance costs rose by 22 per cent.

The analysis shows that O&M costs for 101 operating US nuclear units rose, on average, from $59 million to $72 million. The analysis notes, however, that the effect of higher O&M costs was tempered by the higher capacity factors achieved by US nuclear units in 1988. □

Judge OKs bribery suit against Westinghouse

UNITED STATES: A federal judge tentatively ruled on 13 September that the Philippines government can move forward with its bribery suit against Westinghouse Electric and Burns & Roe Enterprises.

The suit charges that the two firms bribed a crony of former Philippine President Ferdinand Marcos in order to win a contract to build the PNPP 1 nuclear plant (639 MWe, PWR) on Bataan peninsula. PNPP 1 was completed in 1986 but has never operated. The government of current Philippine President Corazon Acquino is seeking to recover its investment in the plant, plus damages.

In May, Judge Dickinson R. Debevoise of the US District Court in Newark, New Jersey, ruled that the bribery suit could continue, but that 14 other claims against the companies for negligence and fraud should be judged by the International Chamber of Commerce. That dispute is still being resolved, as well as Westinghouse's claim that it is owed $30 million in fees for work it performed on the plant.

In his later ruling, Debevoise rejected Westinghouse's initial motion to dismiss the bribery case. Westinghouse lawyers argued that the allegation had previously been investigated by the US Department of Justice, which found no evidence to support the charge. The judge, however, said he would consider another such motion for dismissal after more evidence has been presented. □

CANDU 3 licence sought

The Atomic Energy of Canada intends to submit its CANDU 3 design for licensing in the United States, and has written to this effect to the chairman of the US Nuclear Regulatory Commission. In the letter, sent through AECL's US subsidiary AECL Technologies, AECL president Don Lawson emphasises "the seriousness of AECL's intent to be an active participant in the US market", but says that there are no plans to seek funding from any US government agencies.

SMUD rejects Quadrex offer

The directors of the Sacramento Municipal Utility District have rejected an offer by Quadrex Corp to purchase and operate the Rancho Seco nuclear plant, which was retired following a local referendum in June (*NEI*, July 1989, p11). The five-member SMUD board rejected the offer by a 3-2 margin. The board concluded that the $250 million Quadrex expected to raise for the venture would not be enough capital to ensure SMUD would not be subject to further costs if Quadrex failed.

Fastener test bill goes to senate

The US House of Representatives in September passed and sent to the Senate legislation that would require fasteners to be inspected, tested and certified in lots before being offered for sale. The Fastener Quality Act passed the House by a voice vote. Its sponsor, Representative John Dingell (Democrat, Michigan), chairman of the House Energy Committee, said it would "restore quality, accountability and traceability to fasteners intended for critical uses."

Firms indicted by Grand Jury

A federal grand jury has indicted Aircom Fasteners Inc of Arlington, Texas, and Yamaguchi-Seisakusho Co Ltd of Japan, on charges that the companies manufactured or sold substandard nuts, bolts and screws used in the Comanche Peak nuclear plant. Executives of both firms were also indicted by the grand jury, which alleged that Yamaguchi-Seisakusho manufactured the parts using low-grade boron, then stamped them to indicate they were made of higher-grade steel alloy. Aircom Fasteners allegedly sold them to Texas Utilities, the owner of Comanche Peak. The two firms face fines ranging from $3 to $6.2 million. Individual executives named in the indictments face maximum prison terms of 78 years and a total of $3 million in fines.

GE settles Mark I suit

General Electric has settled out of court a suit brought against it in 1987 by the Nebraska Public Power District alleging that the Cooper nuclear plant's Mark I containment failed to meet Nuclear Regulatory Commission requirements when the station opened in 1975. NPPD had alleged that GE knew, or should have known, of potential problems with the design of the Mark I containment. NPPD has since spent approximately $30 million to modify the containment to comply with NRC regulations. The companies settled before a scheduled October trial date.

IAEA holds 33rd conference . . .
Representatives from 98 member states met at the International Atomic Energy Agency's General Conference in Vienna on 25-29 September.

. . . third term for Blix . . .
Hans Blix was elected director general of the IAEA for the third time at the General Conference. His term of office will last until November 1993. In his address at the opening of the conference Blix noted that world demand for energy is likely to double over the next 15-20 years and that it would matter whether the increase was in nuclear or fossil. He observed that the Paris summit of industrialized nations in July had recognized that nuclear power plays an important role in limiting the output of greenhouse gases.

. . . Israel censured . . .
A resolution, submitted by 14 Middle Eastern countries, stated that the General Conference was "gravely concerned about Israel's growing nuclear capacity" and called on Israel to submit all its nuclear installations to Agency safeguards. The resolution, principally opposed by members from Western Europe and the United States, also requested the director general to consult with Middle Eastern states about applying Agency safeguards to all nuclear installations in the area.

. . . desalination plan floated . . .
Plans for the IAEA to assess the technical and economic potential for using nuclear heat reactors in the desalination of sea water were adopted in a resolution on 29 September.

. . . Soviet proposal . . .
Responding to a Soviet proposal for an international research centre to be located in the Chernobyl control zone (*NEI*, October 1989), Hans Blix said that the idea seemed to fall naturally within the IAEA's mandate to foster research and the exchange of information. However, there had only been time for a preliminary exchange of views at the General Conference, in an informal meeting, and it was too early to say how the idea would evolve.

. . . no decision on South Africa . . .
Consideration of suspension from the IAEA for South Africa, because of its refusal to sign the Nuclear Non-Proliferation Treaty, will be deferred for a year, it was decided at the conference. The decision reflected hopes that the South African government would take steps towards signing the treaty before the NPT review in August 1990.

. . . Blix praises lack of tension
An easing of national tensions in the world had been reflected at the Vienna conference, said Hans Blix in his closing speech, with less time being spent on political controversies and more on technical and scientific matters. The director general praised the positive climate of the five-day conference, saying that there was "an evident and growing desire for international co-operation".

Study predicts 1990s shortfall

UNITED STATES: Electricity supply deficiencies are likely to develop in the USA and Canada in the next ten years, according to a new report, and action should be taken now to enable utilities to meet future demand.

The report, *Electricity Supply and Demand for 1989-1998*, is produced by the North American Electricity Reliability Council. Studying electricity usage in 1988 and predicting demand in the 1990s, the report found several areas where shortfalls were likely.

In *ECAR* (East Central Area) acid rain legislation may mean the loss of over 9000MWe of capacity.

MAAC (Mid-Atlantic Area Council) has experienced record peak loads during the past six summers. In summer 1988 voltage reductions and load management were frequently used, and in future extra capacity will be needed.

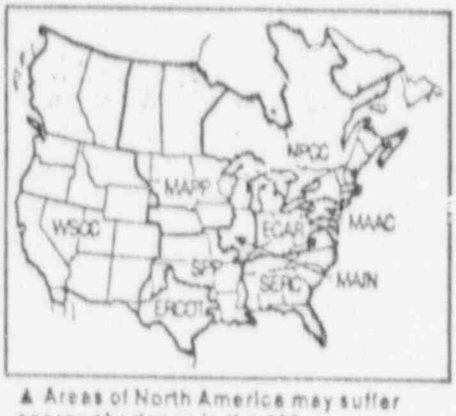
In *MAIN* (Mid-Atlantic Interconnected Network) generating capacity will be inadequate to ensure reliability in the late 1990s. Extra load management capability or short lead-time capacity will be needed.

In *NPCC* (Northeast Power Coordinating Council) capacity beyond that already planned will be required in New England as early as 1993. Unavailability of the Shoreham and Seabrook plants where capacity shortages may occur in the early 1990s is "of great concern". Electricity supply is put at risk by dependency on uncommitted resources for generating capacity.

In *SERC* (Southeastern Electric Reliability Council) load management will be needed to maintain an adequate supply. If peak demand exceeds that forecasted a deficiency could quickly develop.

In other regions reliability problems are mainly due to transmission line deficiencies.

The report says that because large generating units take eight years or more to build, utilities are relying on many different short lead-time options to forestall or prevent generating capacity deficiencies. These include: load management; conservation and other demand management programmes; enhanced maintenance of older generating units; purchases from non-utility generators and other utilities, and short lead-time capacity additions such as combustion turbines. □



▲ Areas of North America may suffer energy shortages in the '90s.

Looking forward to the new generation

VIENNA: Development of a "new generation" of nuclear reactors is probably assured, said James Schlesinger, former chairman of the US Atomic Energy Commission recently, but there is no guarantee that the new reactors will ever be employed.

Schlesinger was speaking as chairman of an IAEA discussion meeting on "The new generation of nuclear power", run in parallel with the IAEA General Conference on 26-27 September. Opening the meeting, he looked forward to the future and spoke about the development work in progress on new types of reactors.

Evolutionary designs could be deployed as early as the 1990s, he said, and there was some enthusiasm for those designs because they could make use of previous experience. Innovative designs, however, required much more work in development and would not be available as commercial products until the next century. Schlesinger praised the work that had been done but warned against overoptimism – vendors are far more willing to supply, he said, than utilities are to buy, pointing out that in the US no order has been placed and filled since 1973.

Summarising discussions at the end of the meeting Schlesinger listed some of the problems that must be solved before nuclear power can be revived. The key to future development, Schlesinger argued, was standardization. There was no point in paying for R&D on a wide variety of different designs, he said, when the barriers were not only technical, but political. They existed in licensing, public acceptance and costs. There were also important issues in waste management and plant operation that needed to be solved and investors would not underwrite such risks.

Between the supporters and opponents of nuclear power was the general public, Schlesinger noted. The public was prepared to be persuaded, and that now should be the task of the nuclear industry. □

SBWR

Simplicity: the key to improved safety, performance and economics

By R. J. McCandless and J. R. Redding

In GE's SBWR design, which has recently won $50 million worth of backing from US DoE, every feature, every system, every piece of equipment must justify its existence – or it must go. Each must perform a needed function in the simplest way because simplification is the key to high performance and competitive economics. The SBWR has the potential to become a safe, economical and environmentally sound energy source for the 1990s, GE believes.

In the United States, there are new forces at work reshaping energy policy. There is increasing concern for environmental problems. The public is alarmed by oil spills, acid rain, ozone depletion and global warming. Responding to these concerns, the US Congress is considering comprehensive legislation that recognizes the interrelationship of energy and environmental issues. At the same time, the need for additional electrical generating capacity is becoming clear.

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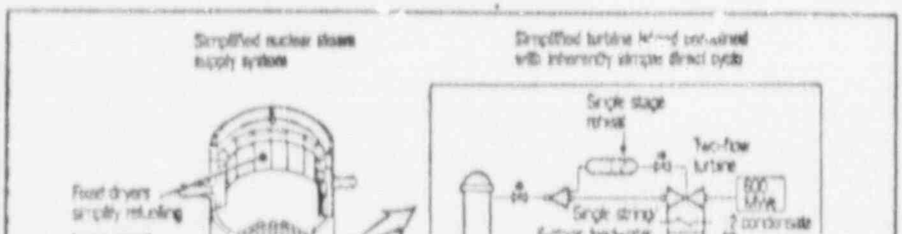
As a result, there has been renewed interest among government and industry policy makers in the re-emergence of nuclear power, and particularly in potential future nuclear units combining the characteristics of smaller size, greater simplification and more passive safety features. GE's newest BWR plant – the Simplified Boiling Water Reactor (SBWR) – aims to respond to this interest by performing its function in the simplest possible way.



▲ The SBWR's simplified nuclear steam supply system, simplified turbine island and passive safety features. By reducing the number of components, GE hopes to improve safety, reliability and availability.

STRONG ENDORSEMENT

The SBWR has earned the praise and endorsement of government, the nuclear industry and utilities. In September 1989, the US Department of Energy (DoE) selected the SBWR as a candidate for licensing in the USA. A $50 million contract is being negotiated for a programme to design a standardized, investor-ready SBWR plant and to obtain NRC certification by 1995.

Another $50 million plus will be added to the SBWR endeavour by a team of US and overseas firms, as well as the Electric Power Research Institute (EPRI), the non-profit research and development arm of the US electric power industry.

The GE-led team will execute detailed engineering and submit an investor-ready design for a complete 600MWe plant, consisting of a nuclear island, a turbine island and balance of plant, which will be submitted to the Nuclear Regulatory Commission for final design approval and design certification.

This programme is a major DoE initiative to make passive, simple LWRs available to utilities in the mid-90s.

The SBWR is endorsed by nuclear industry organizations worldwide in the strongest possible way by their direct participation in the SBWR design and certification programme. In the US, the GE-led SBWR team includes Bechtel Power, Burns and Roe, Foster-Wheeler Energy Applications, Southern Company Services, the Massachusetts Institute of Technology (MIT), the University of California at Berkeley, and various utilities. Internationally, it includes

Ansaldo (Italy), Hitachi (Japan), Toshiba (Japan), ENEL (Italy), KEMA (Holland), and Nucon Engineering and Contracting (Holland).

The SBWR design has also been endorsed by the Electric Power Research Institute and the Advanced Light Water Reactor Utility Steering Committee, which represents many US utilities. These organizations reviewed and evaluated several competing passive LWR concepts and concluded that the SBWR has the potential for meeting utilities' needs for future capacity additions. On the basis of this assessment, the SBWR was selected to receive US utilities' financial support.

GE began studies of the SBWR in 1982. The results attracted the support of EPRI in 1985. GE, teamed with Bechtel and MIT, conducted a conceptual design study which established that the SBWR design objectives were feasible. In 1986, GE performed a study for the Japan Atomic Power Company which established the preliminary feasibility of the SBWR concept.

Also in 1986, the DoE provided essential support when it selected GE (again teamed with Bechtel and MIT) to develop and test key features of the SBWR. This programme, scheduled for completion in 1990, has led to major advances in the SBWR's development.

In 1988 and 1989, many worldwide participants joined the SBWR development programme. GE is currently conducting joint development studies with KEMA, Ansaldo, ENEL, Nucon, Hitachi and Toshiba. These studies have improved and finalized the SBWR design definition. Recently, GE joined forces with the Japan Atomic Power Company and the other Japanese BWR utilities to develop simplified BWR technology, including full-scale testing of passive containment cooling systems. Finally, GE is performing selected SBWR studies for EPRI.

DISTINGUISHING FEATURES

Key SBWR features reflect design objectives which are consistent with today's requirements:

- Smaller rating.
- Simpler operating plant.
- Simpler safety systems, such as using passive features, always keeping the core covered with water, and avoiding the need for operator action.
- Use of existing technology.
- Competitive power generation costs.

Features of the 600MWe SBWR which help to meet these objectives and which distinguish it from similar light water reactors are:

SBWR features compared

Feature	Current BWR	Advanced BWR	Simplified BWR
Fuel	*GE standard*	*GE standard*	*GE standard*
Control rods	Hydraulic	*Electro-hydraulic*	*Electro-hydraulic*
I & C	Analog	*Digital*	*Digital*
Cabling	Hardwired	*Multiplexed*	*Multiplexed*
Turbine	43in last stage buckets	*52in last stage buckets*	*52in last stage buckets*
RX circulation	External pumps	Internal pumps	*Natural circulation*
Safety systems	Active	Simplified active	*Passive:* Gravity-driven core cooling Depressurization valve Isolation condenser Passive containment cooling

- The use of simple, proven direct cycle, natural circulation core flow and low core power density. This nuclear steam supply design, which has no steam generators and no forced recirculation system, coupled with a simplified two-flow, low-pressure turbine arrangement, comes close to having the irreducible minimum equipment needed to generate power.
- The use of a large reactor pressure vessel surrounded by a passive pressure suppression containment system which includes large pools of water that can be injected by gravity into the vessel and keep the core completely covered with water.
- The use of a passive containment cooling system which removes residual heat from the containment without operator action for several days. First introduced in the SBWR, this system, and the gravity driven core cooling system, are now standard features of simplified light water reactor plant designs.
- Significant plant-wide reductions in the need for safety grade systems and other equipment. In particular, the complete elimination of the safety-related pumping systems and diesel generators is a major plant simplification, as is the elimination of the associated complex of support auxiliaries and power distribution equipment.
- Application of state-of-the-art technology developed initially for the Advanced Boiling Water Reactor, or ABWR (*NEI*, June 1986), eg advanced control and monitoring techniques and control rod drive technology. Currently, two ABWR projects are under way in Japan and the ABWR is nearing certification in the United States as a standard plant. The table, above, shows how the SBWR builds upon ABWR technology.

SBWR design simplifications

Design feature	Reduction
Recirculation system	
Pumps	100%
Piping	100%
Supporting equipment (cooling, power, controls, restraints, snubbers etc)	100%
Non-essential process equipment	
Pumps	60%
Heat exchangers	30%
Tanks	50%
Valves	30%
Other equipment	40%
Essential safety grade equipment	
Diesel generators	100%
ECCS pumps, heat exchangers etc	100%
Cooling water system	100%
Standby gas treatment system	100%
Safety/relief valves and discharge piping	80%
Seismic category 1 buildings	
Number of buildings	83%

PURSUING THE LIMITS OF SIMPLICITY

The SBWR power production systems use the simplicity inherent in a direct cycle nuclear plant. Steam from the SBWR reactor is admitted directly to the main turbine. This direct cycle BWR configuration is in itself a major plant simplification because it cuts out the complicated, expensive and potentially unreliable intermediate heat exchange equipment needed to support an indirect cycle.

The evolution of the BWR is notable for the increasing simplification of the coolant recirculation system. Early BWRs used external recirculation pumps and piping loops. Later versions reduced the number of external loops by adopting internal jet pumps. The ABWR carried on this simplification process by introducing internal recirculation pumps and cutting out jet pumps and all external forced recirculation system hardware. The SBWR carries this simplification to its logical conclusion by eliminating these systems altogether.

Instead, the SBWR uses natural circulation to recirculate coolant within the vessel. Because boiling occurs normally in a BWR core, density differences between steam/water mixture inside the core and the water in regions outside the core can (with no support from pumping systems) result in a considerable amount of gravity-induced core flow. Pumps, piping, valves, instruments, power supplies and control systems are thus eliminated, plant arrangements are simpler, and maintenance is less troublesome.

GE's advanced-turbine technology is used in the SBWR. The turbine system is a single-flow high pressure section coupled with a two-flow low pressure section with 52in buckets. The two-flow approach eliminates a low-pressure rotor and shell, simplifies the condser and cuts out piping, valves and other equipment. This turbine is state-of-the art technology and has been selected for the lead ABWR units in Japan.

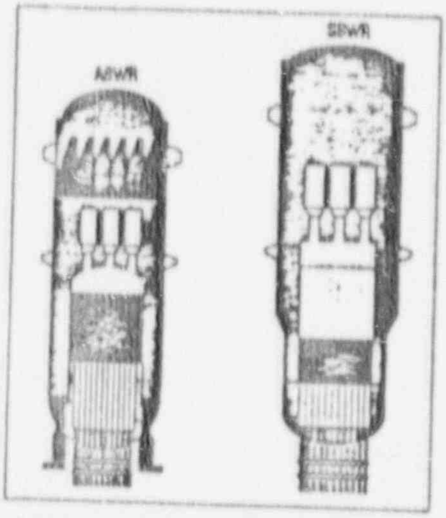
Other turbine system design simplifications include the use of adjustable speed motor driven feedwater pumps, elimination of the separate steam seal system, and application of a single, reliable 100 per cent feedwater heater string. These modifications account for a 45 per cent reduction in the volume of the turbine building when compared to BWRs of similar capacity.

DISTINCTIVE PRESSURE VESSEL

The SBWR reactor pressure vessel (RPV) is 24m high and has a diameter of 7m, except at the bottom where it is 6m. This change in diameter gives the SBWR RPV its distinctive look.

The RPV height is a key factor in estabiishing the required natural circulation core flow. Core flow is enhanced by incorporating a "chimney" in the space between the top of the core and the steam separator assembly. The diameter is wider at the top to increase the water inventory above the core, thereby removing the need for additional equipment (eg accumulators) or capacity in the emergency core cooling systems. The diameter is smaller at the bottom to reduce the volume of water needed to be replaced to provide core cooling.

The large reserve of water above the core translates directly into a much longer time being available before core uncovery can occur as a result of feed flow interruption or a loss-of-coolant accident. This means that there is ample time for the automatic systems or the plant operators to re-establish reactor inventory control using any of several non-safety-related systems. Timely initiation of these systems will preclude initiation of the emergency safety equipment. This easily controlled response to loss of normal feedwater is a significant operational benefit. In addition, the larger RPV volume leads to a substantial reduction in the SBWR pressurization rate that would occur after a rapid isolation of the reactor from the normal heat sink. This characteristic permits considerable simplification of the pressure relief equipment.



A Comparison of ABWR and SBWR reactor pressure vessels. In the SBWR, the steam dryers are installed in an annular ring round the top of the pressure vessel.

Steam separation occurs inside the RPV by passing the steam/water mixture sequentially through an array of steam separators and steam dryers. The SBWR uses standard BWR steam separator assemblies attached to a removable cover on the top of the chimney assembly. The larger SBWR vessel permits the steam dryers to be permanently installed in an annular ring around the top of the RPV. The dimensions of this annular dryer bank are such that removal of the separators and refuelling operations can be performed without the need to take out the dryer assembly. This simplifies refuelling operations.

LOW POWER DENSITY

The SBWR core is designed to have a power density of only 42kW/litre. This is less than the power density of most BWRs (about 50kW/l) and about half that of other LWRs. Low power density results in better economics through lower fuel cycle costs and 24-month operating cycles, which increase the output of the plant. Low power density also translates into more thermal and hydrodynamic stability margins than current operating plants. Vessel embrittlement, which has never been a problem for the BWR because of its low power density compared to other LWRs, is not a concern for the SBWR.

The SBWR core, like those of all BWRs, is designed to be able to use any standard BWR fuel design.

SIMPLE, PASSIVE SAFETY

Major improvements have been made in the SBWR for handling operational transients and accidents. Improvements have been achieved by building inherent margins into the design, enhancing the capability of normally operating systems and introducing passive safety systems. By building inherent margins into the design (eg by adding more water inventory and lowering pressurization rates), system risks such as safety/relief valves opening after reactor isolation are not a problem.

The capability of normally operating systems to handle transients and accidents has been enhanced, by, for example, using adjustable speed, motor-driven feedwater pumps and high-capacity control rod drive pumps with back-up power. The use of systems routinely employed in normal operation to control transients and accidents improves the operators' ability to respond correctly to these non-routine events. Finally, passive safety-grade systems such as gravity-driven emergency core cooling systems and simple condensing heat exchangers to remove residual core heat have been included in the design to provide added confidence in the plant's ability to handle transients and accidents. The design also retains several non-safety grade motor-driven systems as a backup to the passive systems.

By eliminating many of the pumps, motors, motor-operated valves and other active safety-grade equipment required for transient and accident control, the SBWR does not require a complex of on-site, diesel-driven, Class IE emergency power sources and their supporting distribution and control equipment. This, and other simplifications, have resulted in a substantial reduction in the building space required to house safety-related equipment. For the SBWR, the need for seismic Category I buildings to house safety-related equipment is limited to the reactor building. This results in a consolidation of safety-related areas, which are normally more widely dispersed in existing plants.

In the event of a loss-of-coolant accident (LOCA), the SBWR core will not experience uncovery or fuel heat-up due to loss of reactor coolant inventory. this is achieved by:

- Eliminating all large nozzles from the lower region of the RPV.
- Providing a large inventory of water in the RPV region above the core.
- Depressurizing the reactor in the

event of an accident to near ambient conditions.

● Flooding the reactor with low pressure gravity-driven flow from the elevated pools located in the containment.

Following completion of the blowdown/flooding sequence, there is sufficient water in the GDCS and suppression pools to flood the entire containment to a height of at least one meter above the top of the active core. Consequently, the core will remain adequately cooled for an indefinite period following a LOCA.

The DoE-sponsored integrated system testing of the SBWR emergency core cooling configuration was successfully completed in late 1988. These GDCS tests have convincingly confirmed the effectiveness of the simple, passive, gravity-driven core cooling system.

The gravity flooding approach to emergency core cooling is a major plant innovation and key element of the SBWR simple and passive approach to safety.

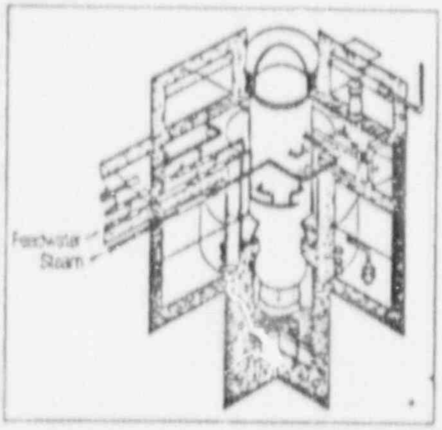
ISOLATION CONDENSERS (IC)

Boiling in the core is a normal, engineered feature of the BWR. It occurs both during normal power operations and briefly as a result of post-scram decay heat production during plant transients and accidents. Because condensation of high-temperature steam is an extremely efficient heat transfer mechanism, direct removal of BWR core decay heat through condensing heat exchangers has always been viewed as an attractive option. Earlier BWRs have successfully used condensers for pressure control during isolation events, and ICs are considered proven technology. Decay heat steam is piped to IC tube bundles submerged in pools of water located above the core and outside the containment. This steam condenses on the inside of the tubes and heats up the surrounding water. Decay heat is ultimately released to the atmosphere as boil-off from the IC pools.

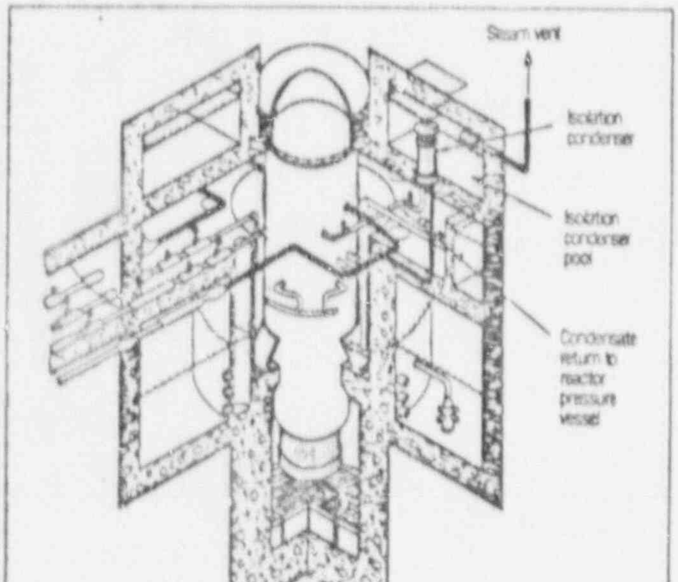
The innovation made in the SBWR is to extend the use of IC technology to the removal of long-term, post-accident decay heat from the containment. This passive containment cooling involves using the IC equipment to condense steam that has been released to the containment by an accident. This steam is channelled by natural circulation to the IC tube-side heat transfer surfaces where it rapidly condenses. The condensate returns by gravity flow to the reactor and non-condensibles are passively purged to the suppression pool. Heat transfer from the tubes to the surrounding IC pool water is accomplished by natural convection and no forced circulation equipment is required. Steam produced in the IC pools is vented into the atmosphere. There is sufficient water inventory above the top of the tube bundles to handle three days of decay heat removal. After three days, simple operator action (the addition of low pressure water to the pools) can continue the passive heat removal indefinitely.

Design simplification efforts have led to the SBWR having only four conventional pneumatic/spring-actuated safety relief valves (SRV) to provide overpressure protection, whereas existing plants can have as many as 20. This reduction is possible because of the low rate of pressure increase for the SBWR reactor system following upset events.

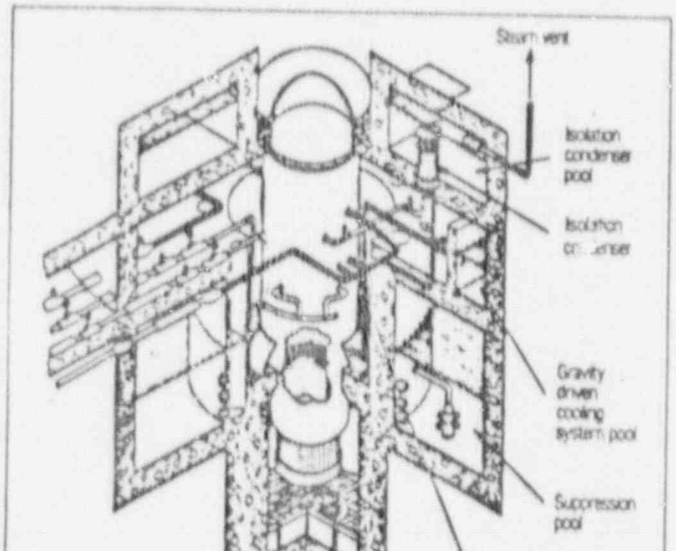
Furthermore, there is ample time during which IC operation can be initiated. This will terminate a reactor pressure increase below the SRV lifting setpoints. Fewer SRV actuations mean



▲ SBWR in normal operation.



▲ SBWR isolation response. The reactor is isolated and the isolation condenser removes decay heat to the atmosphere, so there is no containment heatup.



▲ SBWR loss of coolant accident response. The suppression pool absorbs blowdown energy, the reactor pressure vessel is depressurized by the depressurization valves and then flooded by the gravity driven cooling system. The isolation condenser removes decay heat. No containment flooding is required for most breaks.

less SRV maintenance and less of its associated occupational exposure.

Operation of the GDCS requires reactor depressurization. This depressurization is accomplished with six depressurization valves (DPV) located on the steam pipes in the upper drywell. Squib valves were selected for the DPV function because they are simple, reliable, eliminate all leakage concerns and have low maintenance requirements. A prototype DPV is being designed and tested under DoE sponsorship, and this programme is scheduled for completion in 1990. Progress to date indicates that application of squib type valve technology to the SBWR will be very successful.

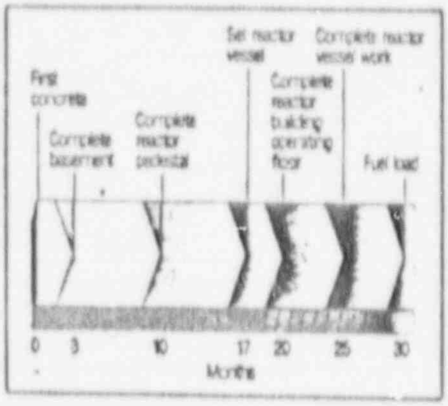
CONSTRUCTION

The reactor building has been arranged to take advantage of the reduced quantity of essential safety-grade equipment. This equipment has been grouped within areas of the containment and the control room. The non-safety cooling systems have been segregated and positioned at the ends of the reactor building. This arrangement leads to numerous benefits during plant construction that reduce cost and improve the installation schedule.

During plant operation and maintenance, non-safety equipment is more accessible and safety equipment security easier to control. All plant safety-related equipment is located in the reactor building. Previous designs had safety equipment distributed in several buildings and intermixed with non-safety equipment. For example, a typical BWR has safety equipment located in as many as six buildings, which requires that each meet seismic Category 1 criteria. Consolidating all safety related items into a single building has major construction cost/schedule and operational benefits – including simpler building interfaces and better access control during construction and operation.

Careful attention has been given to ease of construction with this building arrangement. The building features full 360° accessibility on all floors with wide perimeter aisles for ease of worker movement and simple transport of equipment. Generally, the major cooling equipment has been placed on the lowest floors of the building to allow early installation during construction.

Modularization techniques will be used to reduce costs and shorten construction schedules. These techniques will be applied to such items as reinforcing bar assemblies, structural steel assemblies steel liners for the containment and associated water pools, and selected equipment assemblies. This latter includes equipment such as isolation condensers, drywell piping, HVAC units and water treatment equipment.



▲ Construction schedule milestones.

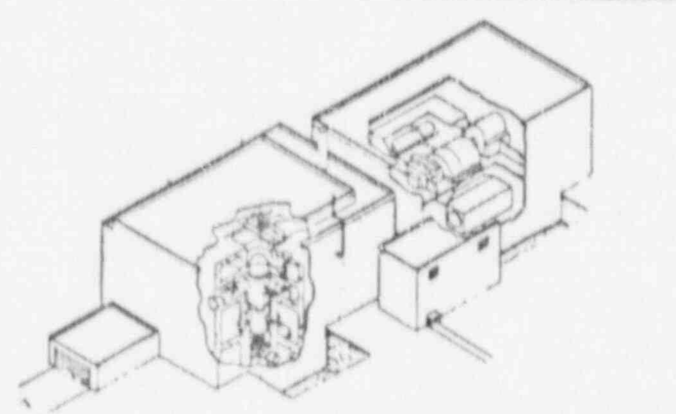
OFFSITE CONSEQUENCES

Analyses of the radiological dose consequences for SBWR accidents show off-site doses after an accident will be less than 1 rem. This low dose rate is achieved without use of a standby gas treatment system (SGTS). This is a major plant simplification because an SGTS needs safety-grade equipment requiring extensive testing, maintenance and surveillance.

The low offsite radiological consequences predicted for the SBWR provide a technical basis for reassessing the need for plant evacuation zones and extensive evacuation planning. □

SBWR performance summary

Plant safety and protection	
Core damage frequency	$<1 \times 10^{-6}$/y
Severe accident offsite dose more than 1/2 mile from plant	<1 rem
Plant performance	
Availability	90%
Planned outage time	<25 days/y
Reference fuel cycle length	24 months
Unplanned automatic scrams	<1/y
Plant design life	60 years
Low level radwaste shipped	<2500ft³/y
Plant personnel exposure	<85 man-rem/y
Constructability	
Capital cost	<$1 350/kWe
Modularization	Extensive
Construction schedule from pouring of first structural concrete to start of fuel loading	30 months
Licensability	
Safety	Passive
Transients and accidents	Slow acting; 20% margins
Technology	Proven, tested
Certification	Standard plant



Simplified construction, operation and maintenance

Power generation support systems
- Reactor cleanup and shutdown systems combined
- Pool clean-up systems simplified
- Reduced nuclear boiler instrumentation
- Simplified core power monitoring

Operation
- No recirculation pumps
- Electrically controlled CRDs
- Enhanced safety margins
- Full scope plant automation
- Improved man-machine interface

Construction
- Extensive structural modularization
- Selective equipment modularization
- Significantly reduced safety envelope
- Multiplexing of plant control systems

Maintenance
- Separate clean and controlled areas
- Eliminated active safety systems
- Reduced surveillance requirements and technical specifications
- Standardized digital instrumentation and control systems with self-testing and auto-calibration

AP600 'Lead Plant' Strategy To Help Meet U.S. Power Needs in the 1990s

After a 15-year hiatus in orders for nuclear power plants, the United States is beginning to realize the necessity of reviving the nuclear option. Over the last few years, the Department of Energy and the electric power industry have put in place a national strategy for developing a new generation of advanced nuclear plants.

But given recent economic and environmental developments, it is becoming clear that the pace of this program is too leisurely. Utilities may find themselves having to order new capacity before the new reactor concepts are market-ready. This would eliminate the possibility of considering nuclear energy for the next round of power plant construction in this country and lead us into further dependence on energy sources that are increasingly unattractive for economic, political, or environmental reasons.

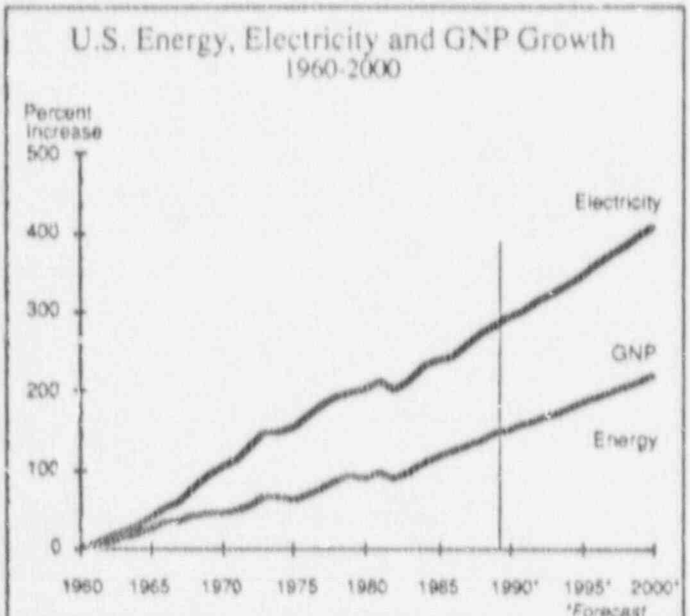
This outcome is avoidable. The United States could begin the next generation of advanced nuclear power plants, based on tested and proven technology, as early as 1995, by adopting Westinghouse's proposed "lead plant" strategy and the modern new reactor design that it is completing, the AP600.

Electric Power Demand and Supply Outlook

Growth in electricity demand continues to be strong, but orders for new power plants have not kept pace. As a result, electricity is already in short supply in some regions of the country. To avert a widespread shortfall in electric supply in the late 1990s and beyond, utilities will have to order new plants within the next few years. Yet there is growing concern about the environmental and economic security risks involved with an excessive dependence on our large-scale fossil-fuel alternatives: oil, natural gas, and coal. As a result, there is a growing realization that nuclear power must play a major role in our energy future.

The Nuclear Success Story

Nuclear electric power has achieved a remarkable record of success in over 30 years of operation. Worldwide, nuclear electricity generation has increased at an unprecedented rate, about five times faster than any other new source of energy. Currently, 417 nuclear power plants are in operation producing about 17 percent of the world's electricity. Nuclear electricity is being produced in 25 countries, and more than half of these nations derive at least 25 percent of their electricity from nuclear power.



Since 1960, electricity has grown faster than any other form of energy. Alone among forms of energy, electricity has kept pace with a growing GNP. These trends are expected to continue in the future.

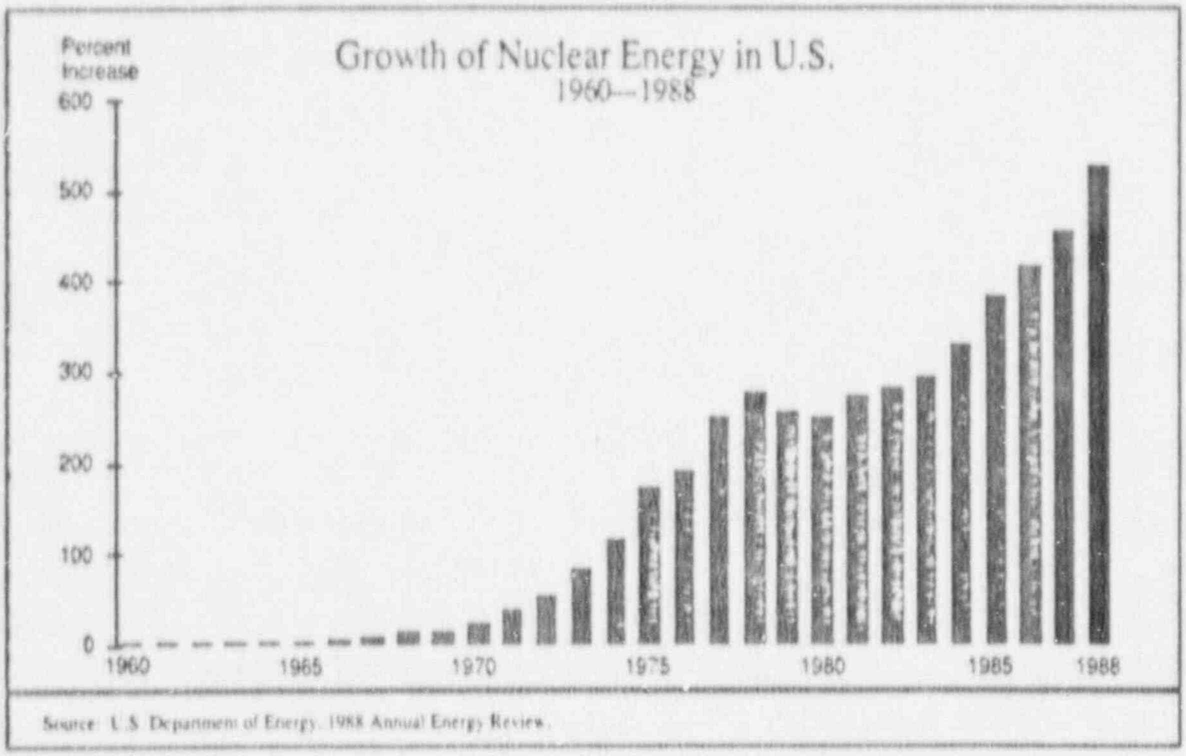
Source: U.S. Department of Energy, 1988 Annual Energy Review and 1989 Annual Energy Outlook.

In all of these countries, nuclear power has made an important contribution to energy security and an improved environment.

- **International Acceptance**
  Many countries now rely heavily on nuclear power. In 1988 France met 70 percent of its total electricity needs with nuclear generation. Other leading producers of nuclear power are Belgium (65 percent), Hungary (49 percent), Sweden (47 percent), South Korea (47 percent), Taiwan (41 percent), Switzerland (37 percent), Finland (36 percent), Spain (36 percent), Bulgaria (36 percent), West Germany (34 percent), Japan (28 percent) and Czechoslavakia (27 percent). As of mid-1989, 113 operating reactors in the U.S. were producing 20 percent of the nation's electricity.

- **Environmentally Clean**
  Nuclear plants do not have the environmental impacts that are inevitable with electric power plants fueled by oil, coal or natural gas. Past experience has demonstrated the additional level of environmental protection made possible by nuclear power, especially in air pollution. Belgium, for example, has reduced its sulfur dioxide emissions by 66 percent through increasing its use of nuclear power. Similarly in France, between 1980 and 1986, SO_2 and NO_x emissions in the electric power sector were reduced by 71 percent and 60 percent respectively.

- **No Greenhouse Gases**
  Nuclear power does not produce carbon dioxide, which is a major contributor to the greenhouse effect. To slow global warming, the world needs to reduce worldwide emission of CO_2 from burning fossil fuels, which is now estimated to be about 20+ billion tons per year. The 1988 Toronto Conference on the Changing Atmosphere proposed a 20 percent reduction in CO_2 emissions from present levels, or a reduction of about 4 billion tons per year, by the year 2005. By comparison, the International Atomic Energy Agency estimates that the use of nuclear power worldwide in 1987 avoided the emission of 1.6 billion tons of CO_2. Expanded use of nuclear power will continue to help the nation and the world avoid further CO_2 emissions.



- **Reduced Foreign Oil Use**
  The substitution of nuclear power for oil is estimated to have reduced the world market for oil by as much as $50 billion annually. This has resulted in lower oil and gas prices and has helped the world recover from the economic recession caused by OPEC's actions and to reduce inflation.

- **Reduced Power Costs**
  The use of nuclear power in the U.S. since 1973 has resulted in a cumulative reduction of $65 billion in electricity costs, mostly in reduced oil imports.

- **Renewed Commitment to Excellence**
  The U.S. utility industry is fully committed to excellence in all aspects of nuclear plant operations, as evidenced by:

  —Formation of the **Institute of Nuclear Power Operations** (INPO) to establish, monitor, and maintain high standards of operational performance

  —Formation of the **Nuclear Management and Resources Council** to focus on management issues in nuclear power plant operation, in consultation with the Nuclear Regulatory Commission

  —Continued funding of the **Electric Power Research Institute** (EPRI) programs to develop improved technology.

- **International Cooperation on Safety**
  An international consensus on nuclear power safety has also been achieved. With the recent formation of the **World Association of Nuclear Operators**, all countries with commercial nuclear power programs participate freely in an exchange of detailed information at the working level.

The next round of nuclear power plants in the United States will need to be different from those of the past because of the major changes that have been taking place in the electric utility industry.

- Because of Federal and state regulatory changes, electric utilities are finding themselves in a far more difficult environment than ever before and are increasingly facing competition from cogenerators and independent power producers.

- State regulators have imposed economic penalties on many utilities that built large new power plants since the late 1970s as high inflation and interest rates drove up construction costs and increased construction schedules.

As a result, utilities today are generally not interested in building the large size, long construction-time projects of the past. Future nuclear plants will need to be smaller, simpler, and more rapidly constructed.

Need for Advanced Technology

In addition to the changes in the electric utility environment, there are other fundamental reasons for applying advanced technology to future nuclear power plants.

The last round of nuclear projects ran into unexpected difficulties in regulation, operation and cost. Both technological advancements and institutional changes will be required to restore full confidence in nuclear power, after the problems that it encountered:

- **Licensing Complexities**
  The United States developed a complex and unpredictable system for licensing and regulating nuclear power plants. The licensing process was oriented to individual projects and to customized, "one-of-a-kind" plant designs. The process invited piecemeal and repetitious consideration of technical issues. The two-step licensing process and the hearings required for it allowed opponents to intervene and even to question the operation of completed plants constructed under Federal guidelines.

- **Three Mile Island Accident**
  The accident at TMI increased public apprehension about the risks of nuclear power. It also reduced the confidence of utility management and the financial community.

- **Changing Regulations**
  Costs rose significantly for many nuclear construction projects because of rapidly changing regulations. In the years following TMI, federal regulators imposed numerous new requirements on utilities building nuclear plants. This forced the redesign of plants already under construction, resulting in delays and associated cost increases.

- **Economic Instability**
  Economically, many nuclear projects suffered from inflation and interest rates that reached double-digit levels. Coupled with regulatory uncertainty and large increases in the time required to complete and license plants, these factors caused plant costs to increase dramatically, and they undermined nuclear power's basic economic advantages.

As a result of the changes in the electric utility environment and the past experience of nuclear power technology, any future design must meet four fundamental criteria:

- It must excel in safety, reliability and maintainability standards.

- It must be economically competitive with fossil-fired units.

- It must protect utility investments by providing predictable construction costs and schedules, assured licensability, and predictable operating and maintenance costs.

- It must be supported by the public and by government officials as safe and environmentally sound.

A joint government-industry effort in the United States is dedicated to restoring nuclear energy as a realistic option for utilities in the 1990s. EPRI and DOE are sponsoring a major program to develop a simplified advanced light water reactor (ALWR) for the next increment of nuclear power generating capacity. The greatest promise lies in providing a design that both builds on the extensive operating experience with current designs of light water reactors and incorporates technological improvements.

Because it is based on proven light-water reactor technology and incorporates the latest design and safety innovations, the AP600 offers the most effective, and least expensive path to bringing an advanced nuclear power plant into operation before the end of this century, and in time to meet America's rapidly increasing demand for electricity.

The AP600 Solution

The AP600 accomplishes this by incorporating state-of-the-art advances into the pressurized water reactor (PWR) technology base that has proven itself safe and efficient in more than 30 years of operation around the world.

The AP600 is the optimized progression from the highly successful Westinghouse two loop, 600 Megawatt (MW) PWR. Westinghouse's 600 MW size plants consistently achieve availability factors 10 percent above the national average.

The AP600 is designed to meet the safety, operational and financial criteria set by industry, government and the public for the next generation of nuclear plants.

- It is mid-sized to allow utilities to add capacity in smaller increments and more closely match capacity additions to demand growth.

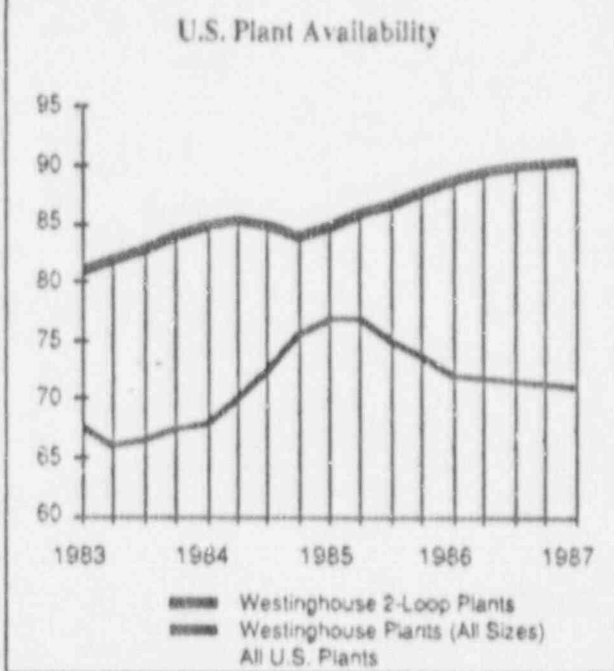
- It is equipped with passive safety systems that rely on natural forces like gravity, instead of operator actions, to control the reactor under accident conditions.

- It is less complex than the nuclear designs used in the past, to save on basic materials, simplify operation and maintenance, and improve availability.

- It features a low power density core with more operating margin, proven high-reliability components, and an 18- or 24-month cycle between refueling. These factors support a high plant availability of 90 percent and a capacity factor of 85 percent, resulting in lower electricity costs.

- It is modular to reduce construction time and costs. Construction on site was the accepted practice in the 1970s and 1980s for the large reactor sizes that were built in that era. The AP600 strategy calls for assembling components into modules in a shipyard-like factory and transporting them to the site for final construction. The controlled manufacturing environment of the shipyard will lead to better quality and more predictable schedules.

- In the AP600 program, the final design will be standardized, completed and certified by NRC before construction begins. Because it is based on the world's most widely used nuclear power technology, the AP600 will not require a demonstration plant for NRC licensing or investor utility confidence. Thus, the AP600 can be in operation sooner than other advanced reactors now being considered.

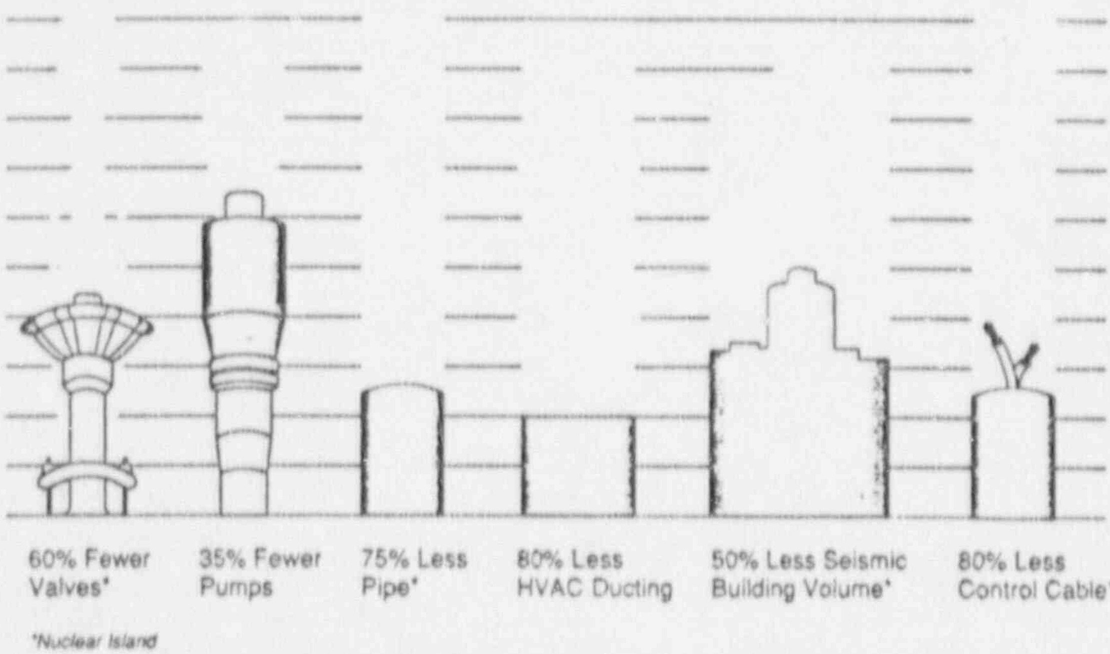


Westinghouse conventional 600 MWe two-loop plants are top performers, with availability 10 percent higher than the national average.

Source: Westinghouse Electric Corporation.

First Nuclear Era	Next Nuclear Generation
Two Step Licensing	One Step Licensing
Custom Design Engineered during Construction	Final Design Completed before Construction Begins
Engineered Safety Features	Passive Safety Features, Design Simplicity, Cost Reduction
Field Construction	Modular Construction
Cost-plus Contract	Turnkey Contract
Regulatory Disallowances	Regulatory Preapprovals

AP600 Reductions in System Components



Passive safety features allow the AP600 to be much simpler than units built during the first nuclear era. Compared to other units of the same size, the AP600 design needs no safety-grade diesel generators, and it provides tremendous reductions in valves, pumps, and other active safety-related equipment.

What Needs To Be Done?

The conceptual design for the AP600 has been completed, and the mechanism for gaining design certification is in place. But the strategy is not complete. If the nation realistically expects to have nuclear power available as an option in this century, we must plan more aggressively to address the institutional issues blocking nuclear power.

If we wait until design certification is complete, at the end of 1994, before taking the next steps, the next nuclear power plants will not be available to help us meet our growing power needs for years into the next century.

The Department of Energy, working with EPRI and the utilities, has awarded Westinghouse a five-year contract that will result in certification of the AP600 design by 1995. To complete the strategy for bringing one of these reactors into operation, beyond these design certification efforts, a number of other actions must be taken.

- **Congressional Action**
  The Congress must enact legislation allowing approval of standardized plant designs and a one-step licensing process to avoid last-minute delays. The one-step licensing process should include early and meaningful public involvement, rigorous pre-defined inspection, and testing to verify that the plant is constructed properly under the terms of the license. This approach would allow nuclear power to become a realistic generating option for both utilities and independent power producers (IPPs).

- **Regulatory Changes**
  Federal and state regulators must reduce the risks to utilities and investors by removing the serious uncertainties about costs and schedules associated with licensing. Even with one-step licensing, utilities and IPPs must have state utility commission or Federal Energy Regulatory Commission assurances that they will be able to operate plants that have been built properly

under all required approvals, and that their investment in the plants will be allowed to earn an appropriate level of return.

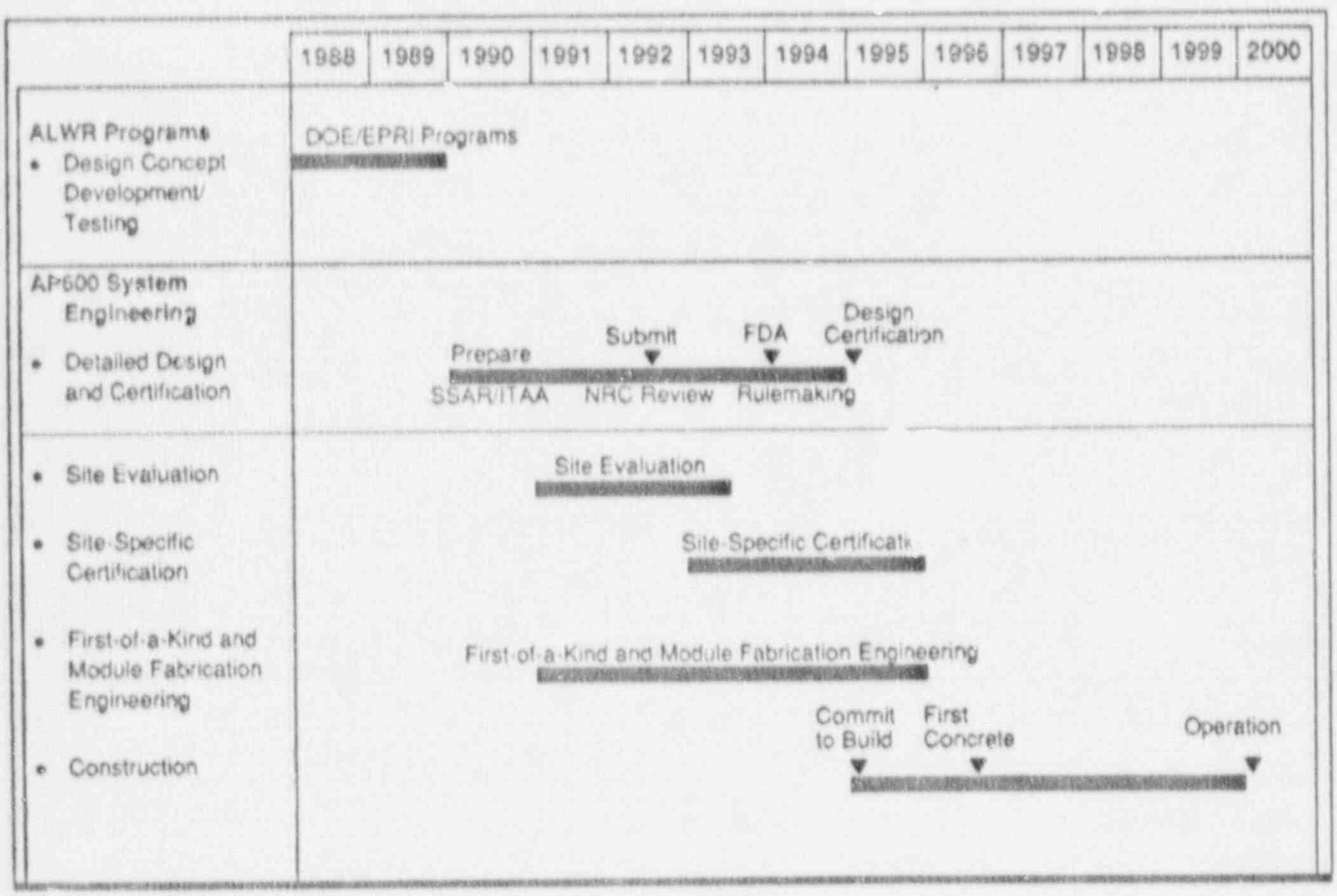
- **Placement of Institutional Concerns on a Parallel Track with Design**
  These institutional concerns should be addressed concurrently with the ongoing reactor design activity. If efforts are not begun to resolve them until an advanced design is completed, several additional years would be lost before the country had the advantage of additional nuclear electric power.

Westinghouse Proposes "Lead Plant" Strategy

The design certification program now underway on the AP600 is a good first step toward the ultimate goal, but given the growing need for new capacity, it is no longer sufficient by itself. We need to speed the other elements that are essential for reviving nuclear power as a realistic option.

Since completion of any of the advanced reactor concepts will involve substantial federal funding, the program should concentrate on the reactor concept most likely to

AP600 "Lead Plant" Implementation Plan



meet all the program's criteria in the least amount of time and for the least amount of money.

To this end, Westinghouse and a team of its industrial partners are proposing a "lead plant" program using the Westinghouse AP600 as the model plant for stepped-up development. Under this plan, several AP600 plants could be in operation by the end of 1999.

In the "lead plant" strategy, several key activities would be undertaken in parallel, rather than sequentially as is now the plan. These include:

- Site selection and certification.
- Completion of first-of-a-kind lead plant engineering.

By conducting these activities in parallel with reactor design and certification, and with efforts to achieve the institutional improvements that are necessary, the "lead plant" approach would expedite the operation of our first advanced nuclear power plants by three years or more.

To help make this plan a reality, Westinghouse is fully committed to innovative approaches and working with all involved parties and the Federal government.

Development of an advanced light water reactor on a timely basis is critical to the future energy security of our nation. The AP600—as the "lead plant"—offers the nation the fastest, surest, and most economical way of reaching this goal by the end of the 1990s.

from Tom Kenyon (P.M.)

PASSIVE AND SIMPLIFIED SYSTEM FEATURES FOR THE ADVANCED WESTINGHOUSE 600 MWe PWR

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Abstract

An important manifestation of new development activities in the area of nuclear plant design is the widespread interest in small plants incorporating passive, rather than active, safety features and simplified systems design. The small plant may be the best choice for applications on a relatively small grid or when load growth rate is expected to be low on a large grid. In response to this potential for small plant applications, the U. S. Department of Energy (DOE), the Electric Power Research Institute (EPRI) and Westinghouse are sponsoring conceptual design development of a pressurized water reactor (PWR)' plant design called the AP600, reflecting the advanced passive safety features and the chosen 600 MWe plant output.

Key Words: PWR, 600 MWe, Passive Systems, Safety, Simplified

1. Background

Present PWR designs have proven themselves to be sound, safe performers. It is widely thought, however, that a new generation of PWRs following in the wake of TMI and Chernobyl will have to incorporate step improvements in safety, reliability, operability and power generation costs. The AP600 conceptual design provides significant improvements in these areas while employing proven LWR component technology. This paper describes the basic reactor and primary coolant system features, the passive safety system features, and some non-safety system features of the AP600.

2. Basic Reactor Coolant System Features

The Reactor Coolant System of the AP600 employs hermetically sealed canned motor pumps mounted in the inverted position and a novel closed-coupled steam generator and pump configuration. These features improve plant reliability, simplify operation, and impro e plant safety. This compact loop and component arrangement is shown in Figure 1. The principle modification to the units is in the bottom channel heads which have been reconfigured to permit the direct attachment of two W 49,000 gpm canned motor pumps. The pump suction nozzles are welded to vertical channel head outlet nozzles effectively combining the SG and RCPs into a single structure. Thus, the need for a separate set of pump supports is eliminated.

3. Reactor Design

3.1 Core

The AP600 uses a low power density reactor core consisting of 145 fuel assemblies of the 17 x 17 OFA (Optimized Fuel Assembly) design with an active fuel length of 12 ft (3.65 m). The design is based on well developed low enriched fuel core technology. Soluble boron and burnable poisons are used for shut down and fuel burn-up reactivity control. Low worth grey rods (12 rod control clusters) are included for load follow and power regulation. The reference fuel cycle is for an 18 month period with a 3 region core. The 145 Fuel Assembly core is surrounded by a stainless steel and water neutron radial reflector which serves to reduce neutron leakage and thus reduce enrichment and fuel cycle cost.

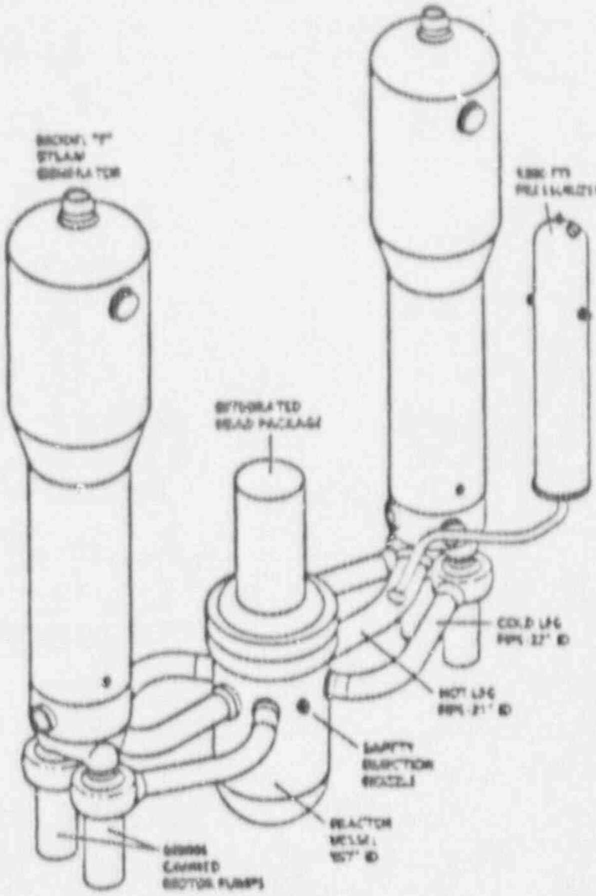


Figure 1. AP600 Reactor Coolant System

3.2 Reactor Vessel and Internals

The reactor vessel and internals of the AP600 PWR are of essentially conventional design so no manufacturing development is required. The reactor vessel inside diameter is 157 in (3.99 m) and its overall height is 435 in (11.05 m). Two 31 in ID hot leg nozzles are spaced 180°F apart and four 22 in ID cold leg nozzles are spaced 90° apart. Two 6.81 in ID direct vessel safety injection nozzles are also incorporated in the upper cylindrical region of the reactor vessel.

4. Reactor Coolant System

4.1 Steam Generator

The AP600 steam generator consists of a proven W Model 'F' tube bundle and secondary side (shell, tube supports, baffles, separators, dryers, and feedwater header), and a modified primary side channel head. The tube bundle is of the vertical "U" tube type and employs 5508 - .688 in OD tubes with a wall thickness of 0.040 in. The tubing material is thermally treated Inconel-690. The active heat transfer area is 55,000 ft^2 (~ 5110 m^2). Each steam generator is rated at 909 MWT and generates 3.95 x 10^6 lbs/hr (1.79 x 10^6kg/hr) of saturated steam at 922 psia (6.36 MPa).

The steam generator channel head has been modified as shown in Figure 2 to permit the direct attachment to the suction nozzles of two (2) canned motor reactor coolant pumps. The design provides ample access and space for tube inspection, plugging, and sleeving by either manual or robotic means while preserving good nozzle entry and exit flow characteristics. This configuration provides an attachment to the bottom mounted reactor coolant pump of sufficient strength and rigidity that the steam generator and pump can be considered as a single structure for support purposes. Attaching the pumps to the steam generator has the effect of increasing the operating weight of the unit by only 10 to 12%. A novel robotically handled multi-port nozzle dam (shown in Figure 2) permits steam generator inspection and maintenance operations while reactor refueling operations are underway.

4.2 Reactor Coolant Pump

The canned motor pump was selected for the AP600 because of its demonstrated record of high reliability and the simplified auxiliary fluid systems needed for its operation. Since this type of pump is hermetically sealed, it improves plant safety by eliminating the possibility of a shaft seal LOCA. The reference reactor coolant pump is a modified Model 8006 canned motor pump designed to develop 3300 shaft horse power at 1770 RPM. The 3 phase pump induction motor is designed for 4000 volts at the terminals and is supplied from the 4160 volt plant bus or an emergency flywheel motor/generator system. Basic motor casing dimensions and weight remain essentially unchanged from the Model 8006 pump which saw service at the Shippingport Nuclear Power Plant.

An aspect of the application of canned motor pumps is that they characteristically have very little rotating inertia to maintain coastdown flow to the core after a loss of power event.

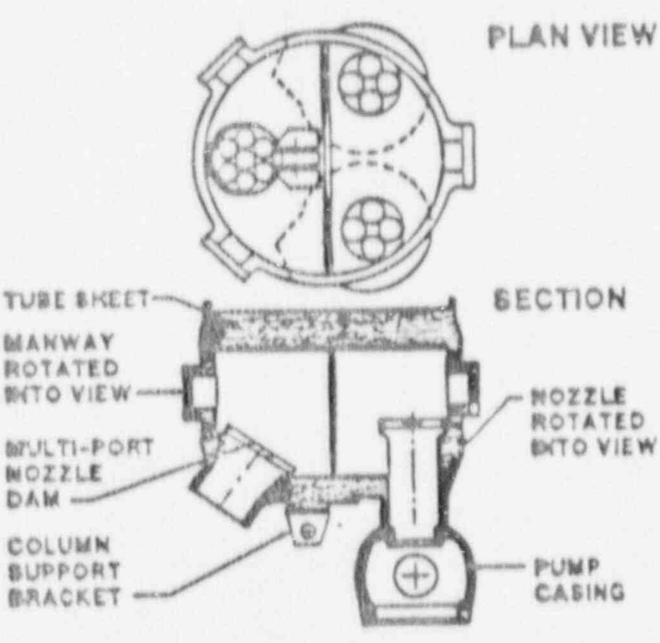


Figure 2. AP600 S/G Channel Head Region Arrangement

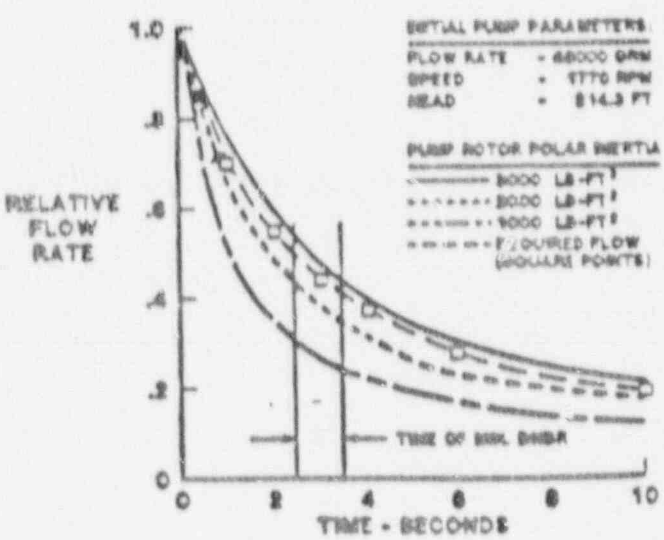


Figure 3. AP600 Loss of Power Transient-Flow Coastdown

Parametric thermal/hydraulic/nuclear analyses have been performed to determine the flow coast down required to satisfy DNBR requirements during a Complete Loss of Flow Accident (CLOFA). The value which must be satisfied is a function of the time to initiate rod trip, rod insertion time, the DNB correlations used and their limitations, the calculation methodology, and the DNBR margin specified.

The results of coast down calculations for various assumed values of pump rotating inertia are shown in Figure 3. It was assumed in this calculation that all four canned motor pumps lose power and coast down simultaneously. The use of currently accepted licensing trip time, and rod insertion time practices require the flow coast-down characteristic shown in Figure 3 by the square points. The reference Model M-8006 pump presently has an inertia of about 1000 lb ft^2 at 1770 RPM. An equivalent pump inertia value of about 3000 lb ft^2 will achieve the required flow coast down.

One alternative to meet the coastdown requirements is an Electrical Inertia System using four 4000 HP 1200 rpm 6 pole synchronous motor/generators with coupled flywheels (one set per pump) to assure that the needed core flow is maintained for the required period of less than 10 sec. Under normal operating conditions the Flywheel M/G set spins at synchronous speed with no power requirements except those due to windage and internal electrical losses. In the event that the pump bus voltage is lost, a line breaker between the bus and the pump opens on low voltage while a breaker connecting the pump to the Flywheel M/G set remains closed. The M/G set converts to a generator driven by its flywheel's rotating inertia, and supplies power to the pump motor for continued pump impeller rotation.

A second alternative for providing adequate flow coast down for the CLOFA event is to increase the rotating inertia of the canned motor pump by factor of about 3. A promising motor and pump design with a rotating inertia of 2970 lb-ft^2 is being pursued. In this configuration the pump thrust runner is greatly increased in size, a heavy uranium disc is incorporated (canned) within the runner, and the pivoted pad thrust and journal bearings needed by the pump are used to develop the tight clearances which reduce flywheel drag losses.

4.3 RCS Support System

The compact symetrical configurati[illegible] of the Reactor Coolant System combined with acceptance of the "Leak-be[illegible]e-Break" criteria makes possible an extremely simple, effective and inexpensive support system. The simplicity of the supports in turn permits excellent access to the steam generators and pumps for inspection and maintenance functions. Basically the rigid attachment of the pumps to the steam generator channel head obviates the need for separate sets of component supports. The use of two symetrical cold leg pipes in each loop provides lateral stiffness at the bottom of the steam generator eliminating the need for lower lateral supports for plants at moderate and low seismic ($\leq$ 0.3g SSE) sites. The adoption of the Leak-before-Break philosophy effectively eliminates pipe whip restraints and shields.

5. Passive Safety Injection System

The Passive Safety Injection System includes two functions - residual heat removal and reactor coolant inventory control.

5.1 Residual Heat Removal Function

A passive RHR heat exchanger is provided to remove core decay heat in case the normal and startup feedwater systems are not available. The passive RHR heat exchanger is located in a natural circulation loop on the Reactor Coolant System (RCS). Refer to Figure 4 for a sketch of the Passive RHR heat exchanger system arrangement. The heat exchanger is located in the containment inside the In-containment Refueling Water Storage Tank which serves as the heat sink. The bottom of the heat exchanger is located about 8 feet above the loops.

The passive RHR heat exchanger is actuated by opening either of the air operated valves, which fail open on loss of power or signal. If the reactor coolant pumps are operating, the flow through the passive RHR heat exchanger will be forced circulation from the higher pressure cold leg through the heat exchanger to the hot leg. In case the reactor coolant pumps are not available, the flow will be natural circulation from the hot leg to the top of the passive RHR heat exchanger to the cold leg. The air operated control valves give the operator a means of controlling the RCS temperature to a constant value or if desired to cool down the RCS.

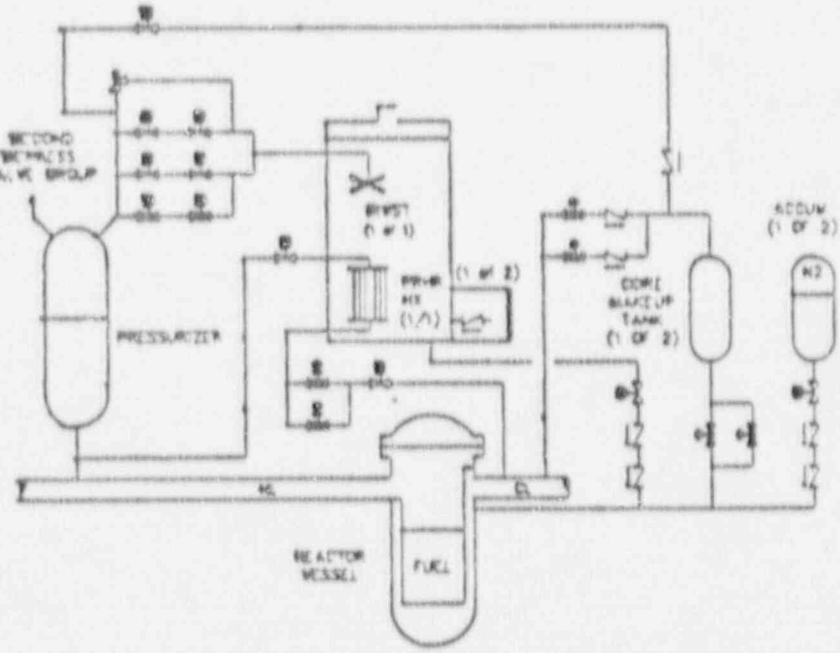


Figure 4. AP600 Passive Safety Injection System

The in-containment refueling water storage tank will absorb decay heat for several hours before the water becomes saturated. However it will take several days to boil off sufficient water from the in-containment refueling water storage tank before the heat removal capability degrades. This provides ample time to recover main or startup feedwater or to align the normal RHR cooling equipment which is part of the AP600 spent fuel cooling system.

The passive RHR heat exchanger is made up of headers to which tubes are welded. The tubes are oriented vertically and are about 20 feet long. There are four headers which are arranged in parallel, separated by several feet to promote good mixing of the steam generated on the surface of the tubes with the water in the in-containment refueling water storage tank. Preliminary transient analysis for the AP600 indicates that about 400 tubes which will remove about 2% decay heat is sufficient to meet the safety requirements.

The passive RHR heat exchanger which replaces the safety grade auxiliary feedwater system does not rely on pumps, AC power, or air/water cooling systems. The functioning of the passive RHR heat exchanger is also not affected by failure of the steam generator pressure boundry, such as steam or feed line breaks or steam generator tube ruptures.

5.2 Passive Safety Injection Function

Passive reactor coolant makeup is provided to accommodate small leaks when the normal makeup system is unavailable and to accommodate larger leaks resulting from Loss of Coolant Accidents (LOCA). Safety grade reactor coolant makeup and safety injection are provided by a set of water tanks; two core makeup tanks, two accumulators and a in-containment refueling water storage tank. Refer to Figure 4 for a sketch of the arrangement of these tanks.

The core makeup tanks are designed to provide makeup for small RCS leaks at any pressure and to provide safety injection for small LOCA. These tanks utilize gravity for their injection force; they are located above the reactor coolant loops and have a pressure balance line connected to the top of the tank to equalize pressures. Each of the core makeup tanks has a capacity of 2000 ft^3 and is maintained full of borated water. The tanks are designed for the same pressure as the RCS. The discharge from the core makeup tank is routed from the bottom of the tank to a separate safety injection nozzle on the reactor vessel; the injection water enters the cold leg downcomer region. This discharge line is normally isolated by two parallel air operated valves that fail open on loss of air pressure or control signal.

Two separate pressure balancing lines are provided for each core makeup tank; one line from the top of the pressurizer and another line from a reactor coolant cold leg pipe. The line from the pressurizer is a smaller line that provides reactor coolant makeup following transients or whenever normal makeup is not available. It is designed for a flowrate of about 15 lb/sec. This line is normally open and contains a check valve to prevent possible back flow or leakage from the cold legs which are at a higher pressure when the reactor coolant pumps are operating. In order to allow core makeup tank injection the reactor coolant pumps are tripped when the pressurizer level reaches a low-low level.

The line from the cold legs to the core makeup tanks is a larger line that provides reactor coolant makeup capability as required for LOCA. It is designed for a flowrate of about 250 lb/sec. This line is normally isolated by two parallel air operated valves that fail open on loss of air pressure or control signal. If the cold legs become voided as they do during a LOCA this line provides a greater flow of steam to the top of the core makeup tanks which allows for a greater flow of water to the reactor coolant system.

The accumulators are required for large LOCA's because of the need for very high makeup flows to refill the reactor vessel downcomer and lower plenum. The accumulator tanks contain borated water with an overpressure of nitrogen; each tank has a capacity of 2000 ft^3 with about 1700 ft^3 of borated water and 700 psig of nitrogen.

Because there is limited volumes of water in the core makeup tanks and in the accumulators additional sources of water are required in the longer term. The in-containment refueling water storage tank is relied on as the longer term source of makeup water. However, in order to get injection from the in-containment refueling water storage tank the RCS pressure must be reduced to about 10 psig above containment pressure. An automatic depressurization system is provided to accomplish this function. A series of valves connected to the pressurizer provide a phased depressurization capability. The discharge from these valves is sparged into the in-containment refueling water storage tank to minimize the consequences of a spuriour opening of one the depressurization valves. These valves are arranged in 3 stages, with the first stage being smaller. This staging reduces the peak flow rates and the resulting loads on the discharge piping, spargers, and the IRWST. Refer to Figure 4 for a sketch of the RCS depressurization valves.

After about 10 hours the in-containment refueling water storage tank will also empty; however, by that time the containment will be flooded up to above the reactor coolant loop level and the water in the containment will drain by gravity back into the RCS. A stable long term core cooling/makeup to the RCS is thus established. The passive containment cooling system supports this operation by removing heat from the containment; steam released from the RCS is condensed and the condensate drained back down so that it is available for recirculation back into the RCS.

This passive safety injection system eliminates the need for high and low head safety injection pumps as well as the need for the matrix of safety grade/redundant active support systems such as the diesel generators, cooling water systems, and HVAC systems. This allows a major reduction in the number of pipes and valves in the safety injection system and reduces to an even greater extent the number of pumps, valves, instruments, and pipe in the support systems that require special QA and are covered by technical specifications. Side benefits of this approach are reduced occupational radiation exposure, inservice tests & inspections, and maintenance.

6. Passive Containment Cooling - Activity Control

Passive containment cooling is provided in case the normal containment fan coolers are not available or an accident has occurred that requires containment heat removal at elevated pressures and temperatures. A passive containment spray system is provided to remove iodine and cesium during an accident in order to reduce offsite doses.

6.1 Passive Containment Cooling

The passive containment cooling system is a safety grade system which is designed to remove heat directly from the reactor containment vessel and transmit it to the environment such that the containment design pressure is not exceeded. The passive containment cooling system does not rely on any active components except for a few valves and air dampers required to initiate its operation. Figure 5 shows a cross section of the AP600 containment illustrating the passive containment cooling features.

The passive containment cooling system utilizes the steel containment vessel as a heat transfer surface. The surrounding concrete shield building is used along with a baffle to direct air from the top-located air inlets down to the bottom of the containment and back up along the containment vessel. In addition, a water storage tank is supported by the shield building at an elevation sufficient to allow gravity drain of the water on top of the steel containment vessel. The air and the evaporated water exhaust through an opening in the roof of the shield building.

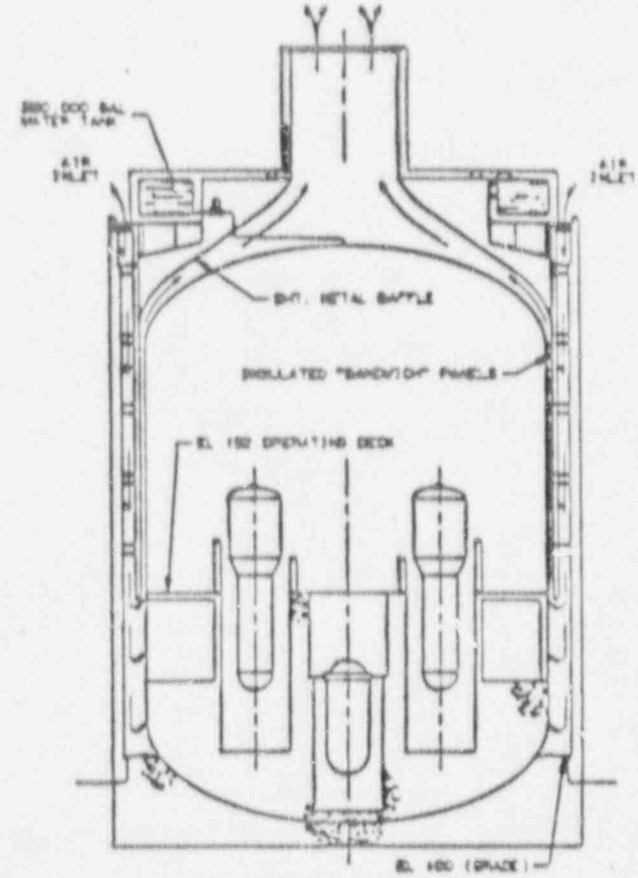


Figure 5. AP600 Passive Containment Cooling (Air/Water Evaporative Cooling)

The passive containment cooling system is initiated automatically by indications of inadequate containment cooling such as high containment pressure or temperature. These signals open air dampers and a few valves to initiate the flow of water on to the containment vessel. The flow of water to the containment is introduced at the very top of the containment dome to ensure that the containment surface is adequately wetted. Circular water distribution weirs placed periodically on the dome ensure adequate distribution. The initial flow of water is about 250 gpm which together with the heat sinks inside of the containment is sufficient to prevent over pressurization of the containment following design basis accidents. As the decay heat level becomes less with time and the mass/energy release from the RCS reduces, the water flow to the containment shell is reduced. This is achieved through the use of passive devices taking advantage of the drop in water level in the tank.

The elevated passive containment cooling system water storage contains sufficient water for three days of operation. Within these three days the operators are expected to take action to replenish this water supply through either the normal filling lines from the demineralized water system or from temporary sources. This action would increase the water flow up to the higher initial flow rates which would result in a reduction of the containment pressure. In the unlikely situation that the operator did not take any action and the water storage tank emptied, the natural convection of air would be sufficient to prevent containment failure although the pressure would rise slightly above the design pressure.

Since the normal containment fan coolers are not required for accident mitigation they will not be safety grade or covered by technical specifications in the AP600. They will be optimized for normal operation which current studies indicate should be two 50% cooling coils each with redundant fans. Chilled water is used for cooling to reduce the surface area requirements.

6.2 Passive Containment Spray

In order to reduce the offsite doses for design basis accidents it is necessary to reduce the concentration of iodine and cesium that is in the containment atmosphere. The AP600 has an accumulator type containment spray system that is designed for this function. It consists of two tanks containing borated water and several tanks containing compressed nitrogen. These tanks are located outside of the reactor containment.

The system is actuated by the presence of high activity in the containment. The actuation signal opens the normally closed valves that isolate the nitrogen tanks from the water tanks. The nitrogen pressure is reduced from the initial storage pressure of about 2400 psig to about 200 psig which is sufficient to force the required spray flow into the containment when it is pressurized at its design pressure. The passive spray system will provide a spray flow of 1100 gpm for at least 30 minutes with the spray continuing for another 15 minutes at reduced flow rates.

The passive spray system replaces the containment spray pumps and their associated valves, pipe and instrumentation. Spray additive tanks are not required.

4. Non-Safety Systems

The design of the AP600 passive safety systems has eliminated some of the systems that are currently used in PWR's. The auxiliary feedwater system, the residual heat removal system, the essential service water system, and the boron recycle system have been eliminated. The safety functions of the auxiliary feedwater system and the residual heat removal system have been replaced by the passive RHR heat exchanger. The non-safety functions are provided by the startup feedwater system and a modified spent fuel cooling system. The safety functions of the essential service water system is provided by the passive RHR heat exchanger and the passive containment cooling system; the non-safety functions are met by the normal service water system. The recycling of boron and water back to the chemical and volume control system has been eliminated by a reduction in effluents produced and taking credit for better fuel performance (fewer fuel failures). The effluents have been reduced by providing plant load follow with control rods, longer fuel cycle, and by the use of canned reactor coolant pumps.

The characteristics of the passive safety injection system and the passive containment cooling system allow the support systems such as the cooling water systems, the HVAC, and the AC power systems to be non-safety and simplified. Other characteristics of the AP600 plant, such as the use of canned reactor coolant pumps, have allowed for the simplification of other systems like the waste processing systems.

7. Summary

The AP600 low power density core design provides improved fuel reliability, reduced fuel enrichment and fuel cycle cost, increased plant availability, and extended reactor vessel life.

The AP600 Reactor Coolant System design provides improved operational transient margins, improved safety, reliability, availability and operational simplicity, reduced capital cost and inservice inspection requirements, improved natural circulation capability, and improved steam generator reliability.

The AP600 RCS configuration, passive safety systems and non-safety systems designs provide for substantial simplification of the plant. The following table summarizes simplifications that have been made to AP600 systems associated with the nuclear island, relative to current plants.

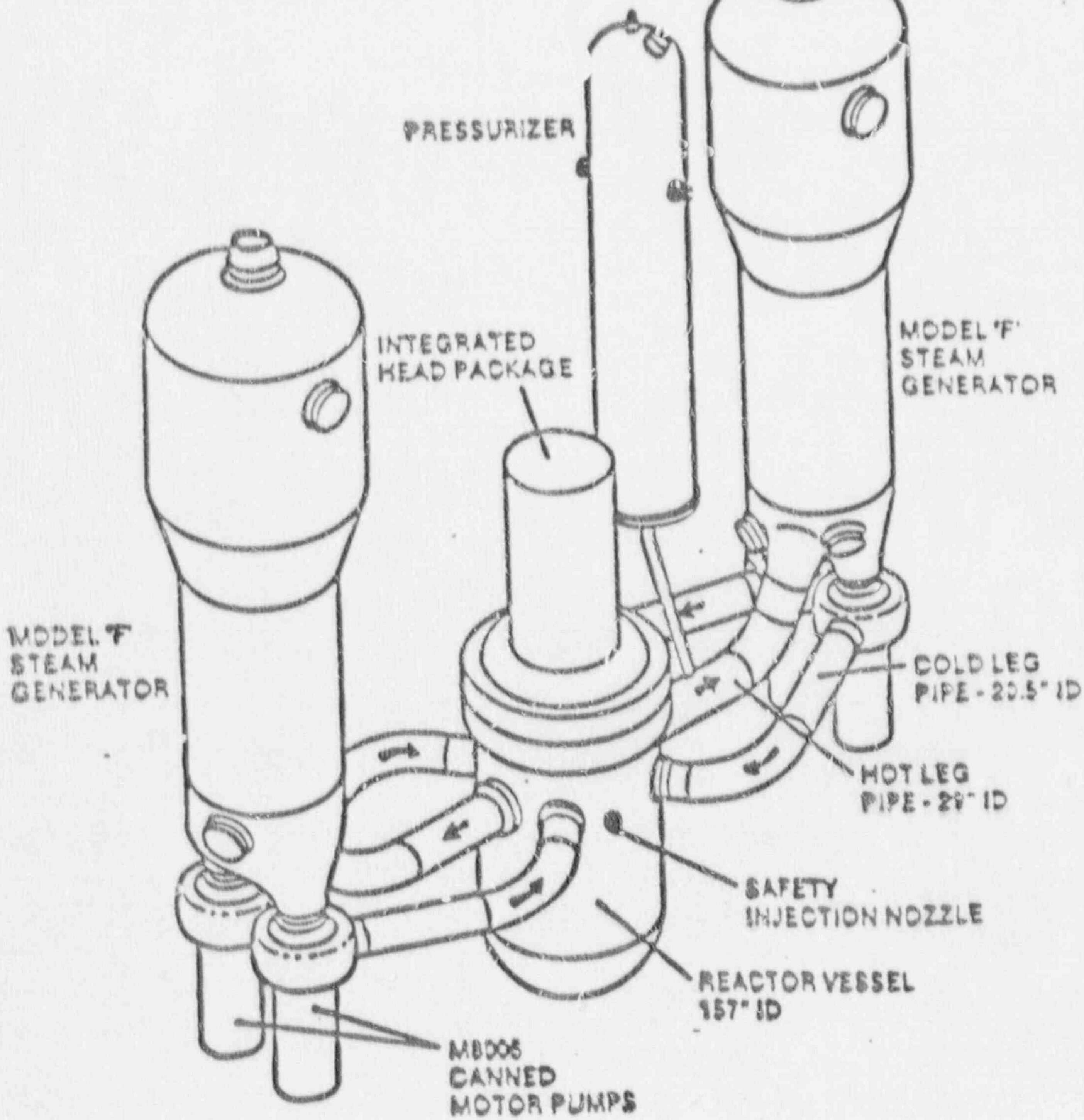
Equipment	Current Plants	AP600
Pumps-Safety	25	None
-Non-Safety	23	22
Tanks	42	27
Heat Exchangers	14	8
Valves-Remote	350	130
-Manual > 2"	700	250
Pipe Length > 2"	31000 ft	11000 ft
Evaporators	2	None
Diesel Generators-Safety	2	None
-Non-Safety	None	1

These simplifications will help the AP600 meet its major design goals of reduced capital cost, reduced construction schedule, increased public safety, reduced occupational radiation exposure, increased plant availability, reduced maintenance and inspection.

AP-600 DEVELOPMENT

- DOE sponsored with EPRI support
- Passive safety systems design
- Plant-wide simplifications
- NRC safety review in 1988

AP-600 REACTOR COOLANT SYSTEM

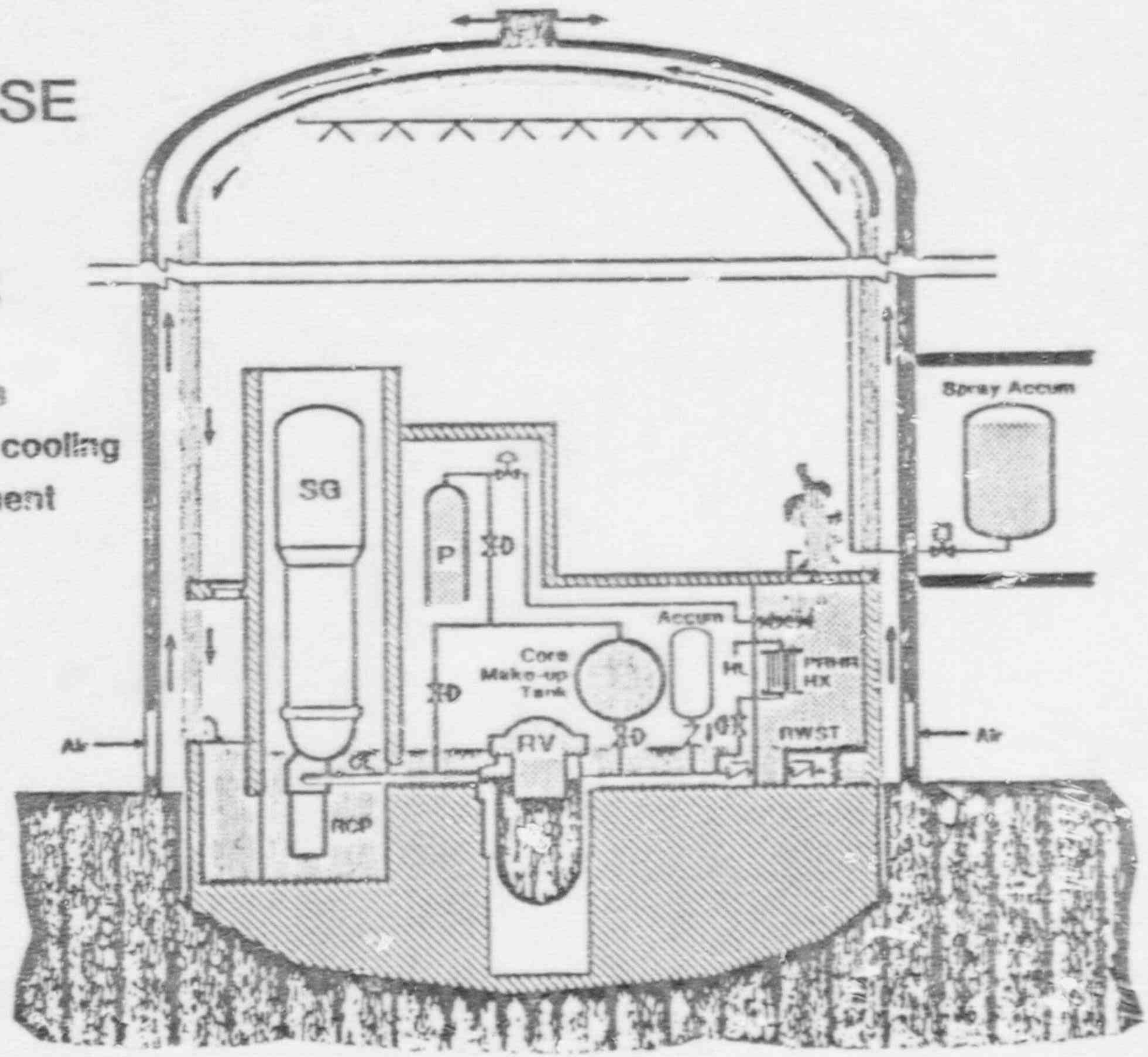


WESTINGHOUSE AP-600 PASSIVE SAFEGUARDS

- Battery operated valves
- Natural circulation core cooling
- Air convection containment cooling

Result in

- No ECCS pumps
- No spray pumps
- No cooling systems
- No Class 1E diesels



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PIUS

The PIUS Arrangement Principle for a Simplified Nuclear Advanced Light Water Reactor Plant

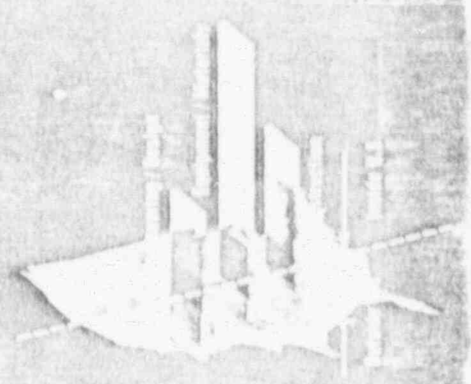
PIUS is a design of completely passive safety incorporated into well proven nuclear reactor technology. Conceived with the goal of simplicity, PIUS is an uncomplicated, understandable, and manageable arrangement that relies on immutable forces of nature, rather than mechanical devices or operator intervention, to achieve a reactor shutdown or reduction in power.

With more than a decade of engineering and testing behind it, PIUS is ready to be considered in the United States.

A subsidiary of Asea Brown Boveri (ABB), ABB Atom (formerly Asea Atom), of Sweden, deals solely with the development, engineering, design, and construction of light-water nuclear power reactors; the fabrication and delivery of reactor fuel; nuclear plant services; and advanced nuclear systems development.

ABB Atom and United Engineers & Constructors Inc. (UE&C), of Philadelphia, Pennsylvania, have agreed to form a new company to market the PIUS reactor in the United States. UE&C will assume much of the responsibility to design the balance-of-plant, support detailed design of the nuclear island, and provide other technical and administrative expertise to the further development of the overall concept. A natural partner for ABB Atom, UE&C provides a strong track record in the U.S. nuclear industry, past technical work on the PIUS design, and a position of leadership in the modular design of both nuclear and fossil power plants.

Where will capital for PIUS be spent?

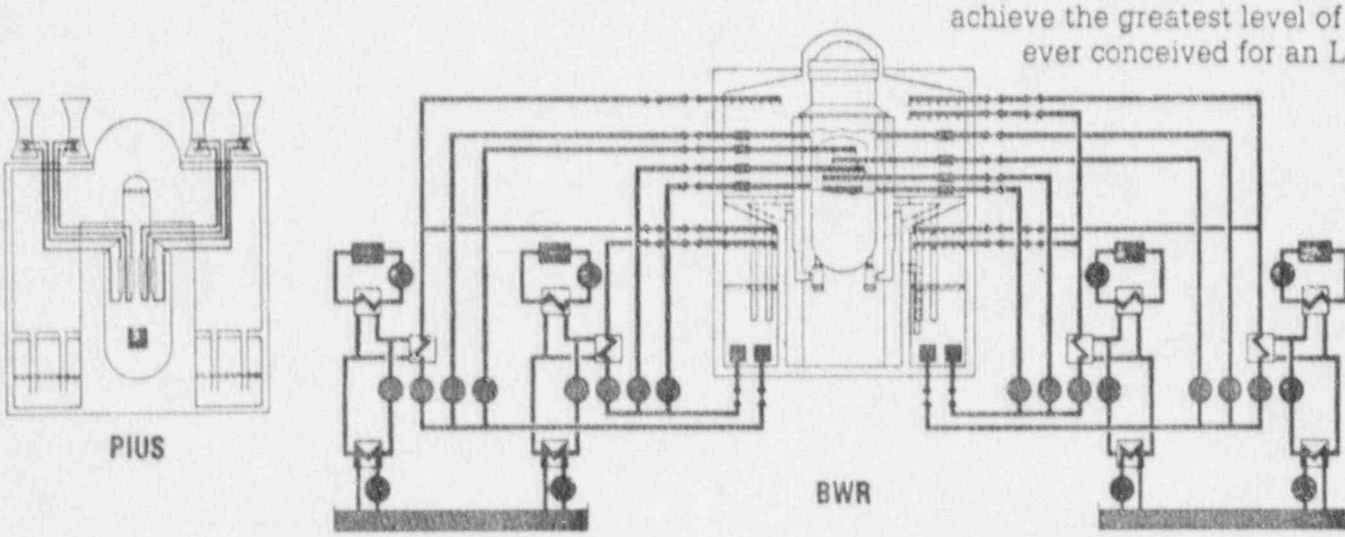


U.S. expenditures are the maximum possible. Bars total 100% of the capital costs of a finished project.

It will be among the highest priorities of the joint venture company to perform the bulk of the design work, to contract construction labor and materials, and to purchase most of the equipment in the United States.

To a nuclear reactor designer, the ultimate safety goal is to prevent the release of radiation from the primary loop, thereby protecting the plant operating personnel and the public. The most straightforward way of accomplishing this is to prevent damage to the reactor core. This, in turn, retains the radioactive products of the fission process safely in the fuel rods and preserves the integrity of the plant. The present generation of light water reactors (LWRs) has achieved that goal with great success. Their record of safety can be traced to a great extent to the design philosophy of defense-in-depth, that is, the use of multiple barriers and back-up systems. The operation of these units depends, however, on given automatic systems or operator actions that – although it is a very low probability – could fail in an emergency.

ABB Atom was convinced more than a decade ago that safety could be enhanced further, that greater simplicity could be achieved, and that emergency planning could be reduced substantially by replacing operator actions and automated equipment with immutable laws of physics and passive safety functions. The pursuit of this belief first resulted in a district heating reactor, and then a significantly improved version of its own boiling water reactor (BWR) design. However, not satisfied with this achievement and certain that the amount of additional improvement possible with the BWR was limited, ABB Atom decided to apply the new and innovative approach used for the district heating reactor to the design of a new configuration of nuclear power reactors, based on well established LWR technology. Supplementing this decision, ABB Atom set a second, prudent goal: to make use of tried and true equipment. The result is PIUS—standard PWR fuel with increased margins and proven equipment, arranged in a new and simple way, to achieve the greatest level of safety ever conceived for an LWR.



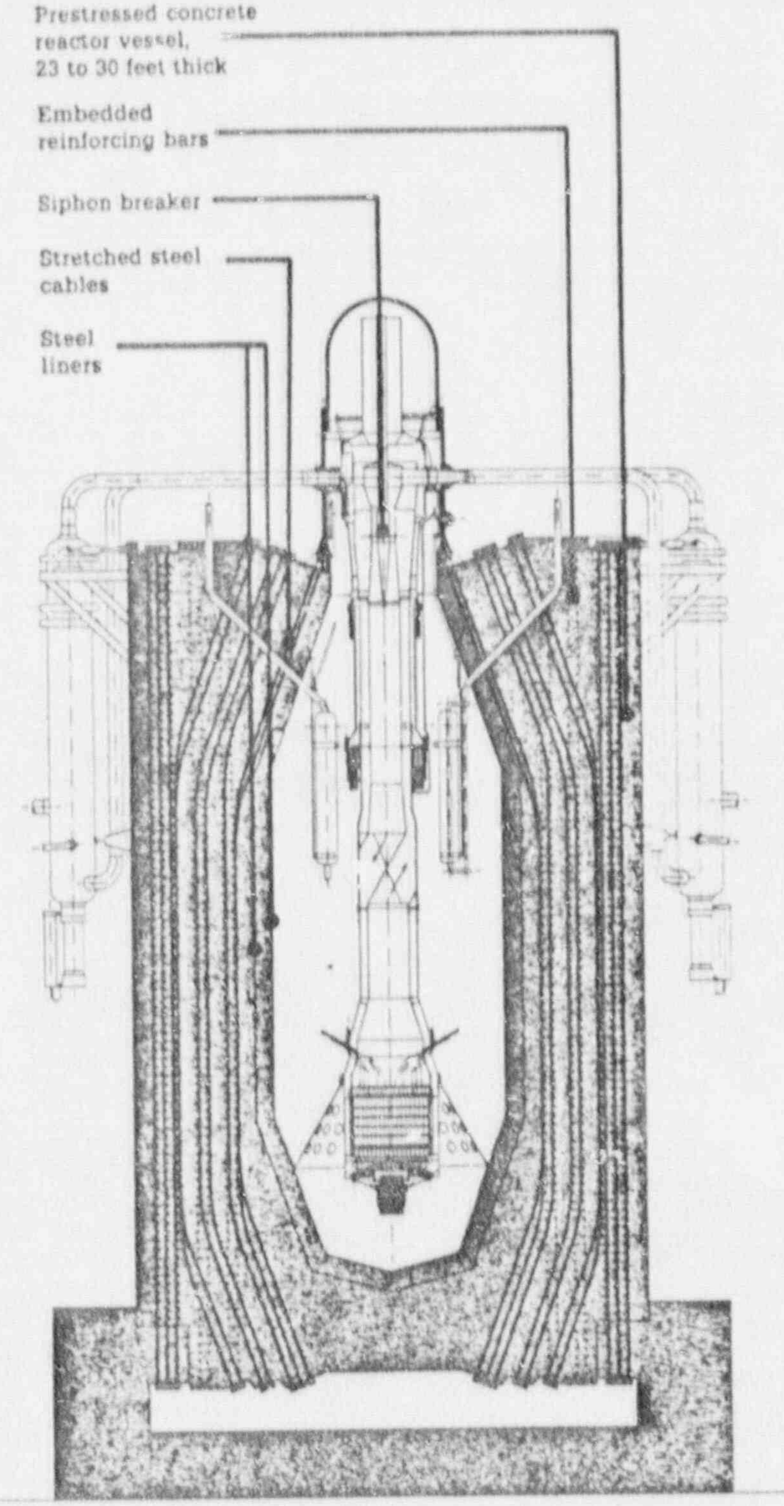
Simplicity is at the heart of the PIUS design. For example, the emergency cooling system of a standard ABB Atom BWR (which incorporates American codes) is a complex configuration of defense-in-depth; whereas, the cooling system of PIUS removes excess heat in the pool water directly to air-cooled heat exchangers by natural circulation, without pumps or instrumentation.

For any LWR, including PIUS, the most direct way of ensuring safety (keeping the core intact) is to fulfill two conditions:

- the core must remain submerged in cooling water at all times
- the combined effect of heating from the nuclear reaction (or its decay products) and cooling from the submerging water must be such that temperatures do not exceed those that will damage the fuel rods.

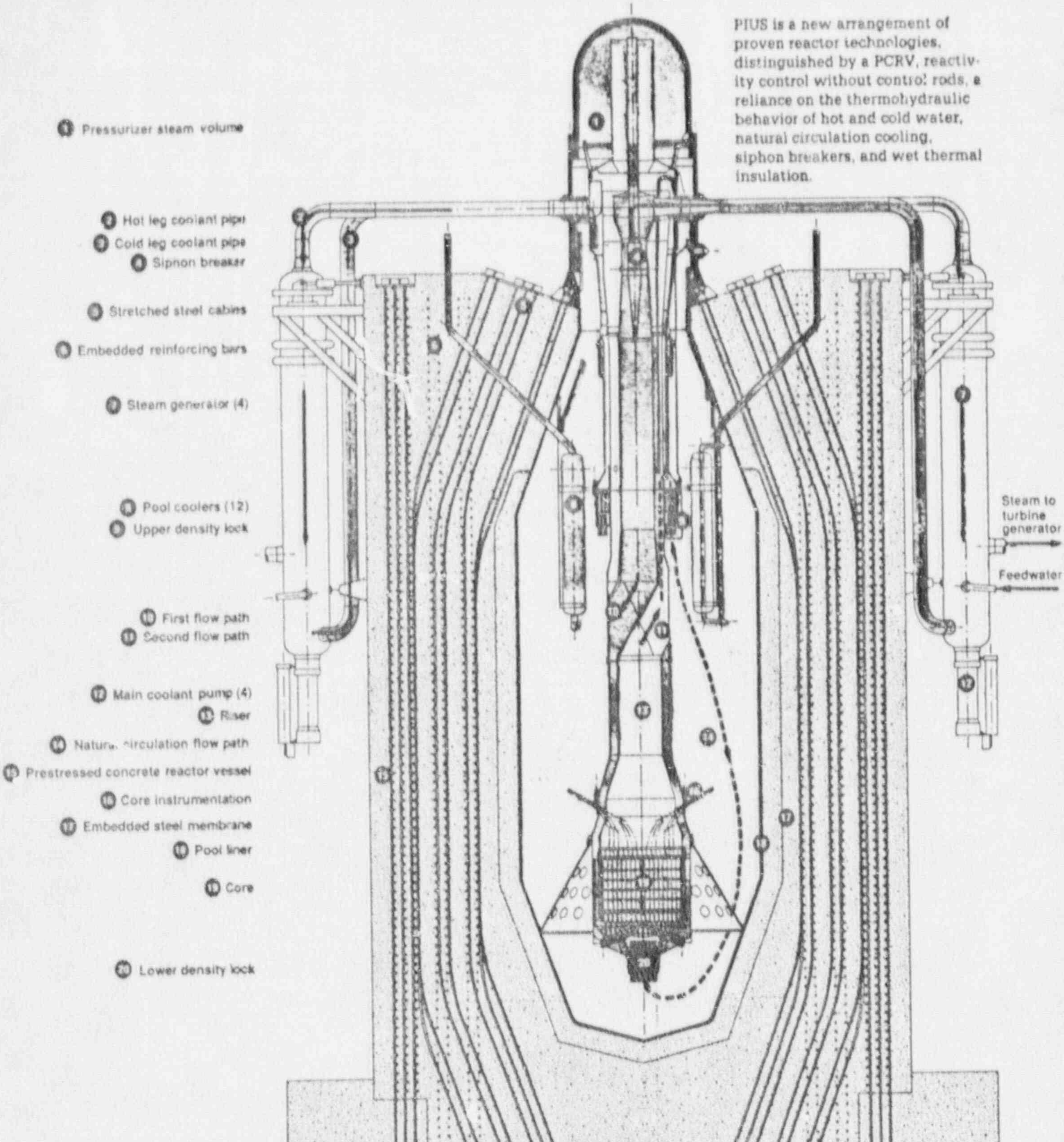
The first condition is met in PIUS by placing the reactor core in a large pool of water contained by a large prestressed concrete reactor vessel (PCRV). The viability of a PCRV in a nuclear reactor has been demonstrated in nearly 30 applications on gas-cooled reactors throughout the world. PCRV technology is well understood and proven.

A PCRV with a steel lining on its interior cavity is virtually leak tight. The vessel is strengthened by embedded reinforcing bars, like those used in common building construction, and by hundreds of stretched steel cables tightly compressing the concrete. These cables, deeply embedded in tubes within the concrete, are similar to those used to support roadways on the world's great suspension bridges, the Golden Gate Bridge, for example. Despite this assurance against leaks, another level of protection is provided in PIUS through a second liner embedded in the concrete. Furthermore, PIUS incorporates special design features that ensure that water cannot be siphoned out of the vessel.



The PIUS PCRV is designed to conform with all of the relevant in-service and extreme environmental and failure load conditions of the ASME III code.

PIUS is a new arrangement of proven reactor technologies, distinguished by a PCRV, reactivity control without control rods, a reliance on the thermohydraulic behavior of hot and cold water, natural circulation cooling, siphon breakers, and wet thermal insulation.



- The second condition for safe LWR operation (that is, maintaining operating temperatures at levels that do not threaten the integrity of the fuel rods) is met by providing three distinct design features that are innovative and unique to PIUS: boron in solution in the pool water, the density locks and the associated core cooling principles, and natural circulation cooling of the submerging pool.

Boron Solution

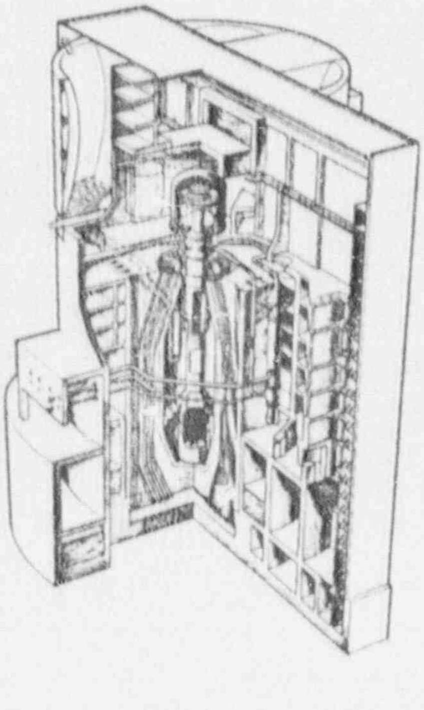
Boron is a "poison" to a nuclear chain reaction. That is, it absorbs neutrons, thereby removing them from the chain. If dissolved in the pool water in sufficient concentrations, boron then is readily accessible to the core (as described in the next section) and an immediate safeguard to shut the reactor down quickly.

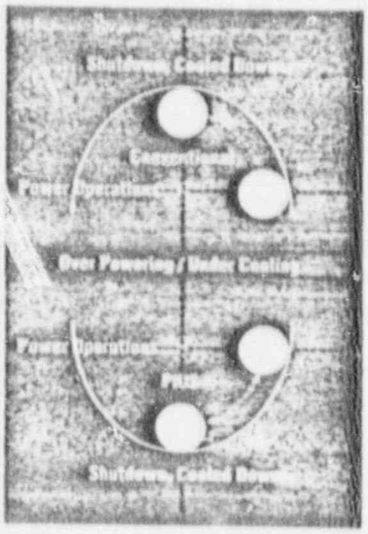
Density Locks and Safety Functions

Inside the PIUS reactor vessel is a cylindrical structure extending from the bottom of the core, near the bottom of the vessel, to the top enclosure (surrounding the pressurizer steam volume) of the vessel. This structure forms, with its internal walls, two flow paths (to and from the core), which are isolated from the rest of the pool except in the areas of the density locks. At the top enclosure, these flow paths connect to external reactor piping.

Under normal operating conditions, reactor core water, having just given up heat to make steam in the steam generator, is pumped, in the first flow path, through the cold leg of the reactor coolant piping to the top enclosure of the vessel. Entering the vessel, it continues through a separate, designated path within the cylinder to flow downward toward the reactor core. Turning at the bottom of the structure, it flows upward into the core, where it is heated. The cylinder above the core acts much like a chimney, pulling the heated water up in the second flow path in the cylinder to the top enclosure. Entering the hot leg of the reactor coolant piping, it then completes its circuit to the steam generator.

The density locks are honeycombs of tubes connecting the water in the flow path described above with the pool. In these density locks, hot water is layered stably on top of cold water; this is known as a thermal barrier. When the pumps are not running, the water heated by the core rises in the "chimney" and flows to the pool through the upper density lock. Cold water from the pool enters the core through the lower density lock, cooling the core and, with its dissolved boron, shutting down the chain reaction in the core. Thus, with no pumps to carry reactor heat away, all reactor heat will be removed by the flow of the water through the density locks, and then through the closed loop pool coolers (as described in the next section). This all occurs naturally, without the use of any electric powered equipment.





The fundamental behavior of PIUS is opposite to that of a conventional reactor. In PIUS, controlled outside forces (e.g., electrical power) must be present to achieve stable power operations; otherwise, the reactor tends toward its natural state – shut down and cooled down. Thus, left to its own, PIUS is safe. Conventional reactors, on the other hand, are dependent on controlled outside forces to move them to a safe condition – they will not do so on their own.

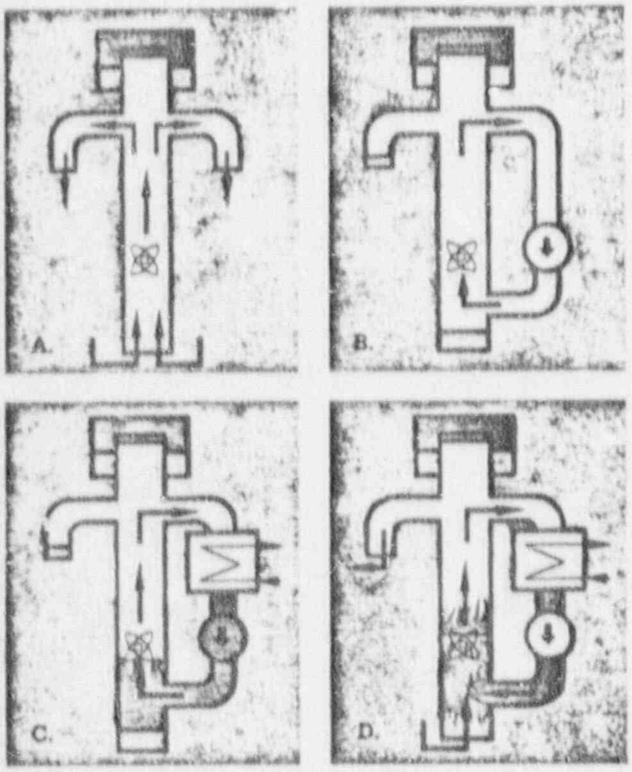
When the primary coolant pumps are engaged in the startup process, they redirect the water that is rising through the "chimney" to the steam generators and, thereby, stop any natural circulation through the density locks. When this hydraulic balance is achieved, and the heat added by the reactor is the same as that removed by the steam generators, flow through the density locks will stagnate and form the distinct and stable thermal barrier between cold, highly borated pool water and hot, less borated water in the reactor circulating loop. Any situation that could potentially result in core damage in a conventional reactor (such as a loss of a pump, loss of steam demand, or a reactor overpower excursion) will break down the established hydraulic and thermal balance, causing the safety response of PIUS – cold borated water rises quickly into the core, shuts it down, and cools it. Yet, normal disturbances in the density locks (for example, variations in the level of the thermal barrier caused by changing demands) will be accommodated by a wide buffer zone incorporated into their design. Analysis and testing (described later) have shown that stable power operations, extended load following, and quick cool-down and safe shutdown in emergencies can be achieved reliably.

Control of the reactor during normal operations is maintained by regulation of the boron concentration in the reactor circulating loop and by the "negative temperature coefficient" designed into the system. Having a negative temperature coefficient also allows the reactor power level to follow load quickly without operator action, making the reactor very responsive to the demands from the electrical grid.

Natural Circulation Cooling

Any heat added to the pool by the reactor core is removed by both active and passive systems. The active system uses electrically driven passages to pass the water through heat exchangers, where it is cooled. The passive system is a closed loop composed of coolers submerged in the pool, air-cooled heat exchangers (similar to automobile radiators – excluding the fans) in cooling towers located above the PCRV, and interconnected piping. Flow in this passive loop is by natural circulation; water made lighter (less dense) by heat in the pool rises to displace water made heavier (more dense) when it is cooled in the air-cooled heat exchangers. This natural circulation is the same process that occurs daily in millions of home hot water heating systems, where lighter hot water rises from the furnace to the radiators throughout the house and the heavier water cooled in the radiators returns to the furnace by gravity. Should power be lost to the pump in the active loop, cooling will be maintained by the passive loop. If both the active and passive loops are incapacitated, the water inventory will keep the core cooled for several days.

All of these PIUS features, which are fundamental to the design, equal or exceed EPRI's proposed designs for passive reactor safety systems—and they do so entirely without movement of mechanical systems, external electrical power supply, sophisticated instrumentation and control, or operator action. These features, based completely on the ordinary principles of nature, make PIUS a concept that can be presented to, understood by, and trusted by the layperson. Furthermore, the layperson can witness the principle of PIUS because the sequence of safety reactions is easily, and visibly, demonstrable with scale models, as discussed in the following section.



The Passive Shutdown Principle

A. A vertical pipe with openings at the top and bottom is placed in a large pool of water. A (nuclear) heat source inside, near the bottom of the pipe, heats the adjacent water, making it less dense than the water outside the pipe. This heated water is buoyant and rises, creating a natural circulation through the pool as it is replaced at the bottom of the pipe with the cooler pool water.

B. If a pump then is operated in a parallel leg connected to the vertical pipe, the pump can drive the water through the outer leg to maintain the flow in the loop while stopping flow to and from the pool. A distinct border between hot and cold water, known as a thermal barrier, is formed. In this configuration, the temperature of the water in the loop will continually increase and the pump speed will require continual adjustment to compensate for this decreasing density - if the thermal barrier is to remain intact.

C. Making this pumped segment of the loop into a power production loop requires the addition of a steam generator. When the heat inputs and outputs are matched, it is possible to create a stable circulating loop, requiring only small variations in pump speed. Thermal barriers are established at the ingress and egress (called "density locks") of the pipe.

D. With this configuration, any set of conditions that could lead to overheating and damage to the reactor core (e.g., significant reactor overpower, steam generator under cooling, loss of a coolant pump, or loss of coolant) will also upset the hydraulic balance that keeps the pool and loop waters separated. Such a condition will cause the temperature of the water in the pipe to rise, thus decreasing its density. Pool water will then rush in through the lower density lock, thereby re-establishing a natural circulation, as illustrated in A, at left. The pool water contains boron, a poison to a nuclear chain reaction, thus shutting down the reactor as well as cooling it.

BECHTEL

Stearns-Roger

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BORG WARNER

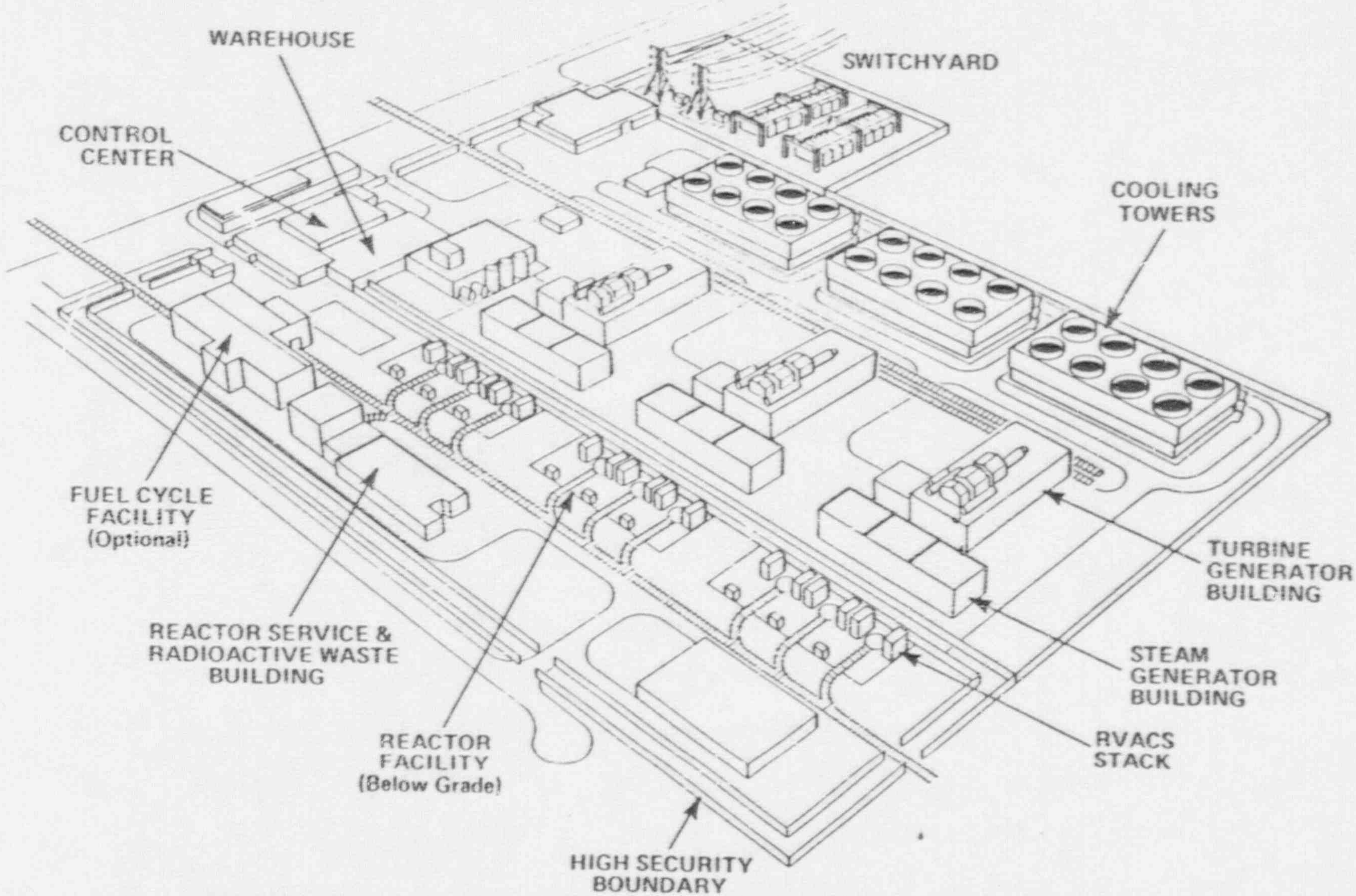
PRISM™

HEDL ETEC INEL ANL ornl

PRISM DESIGN APPROACH

- O COMPACT POOL-TYPE REACTOR MODULES SIZED TO ENABLE:
  - FACTORY FABRICATION WITH MINIMUM SITE INSTALLATION LABOR
  - ECONOMICAL SHIPMENT TO INLAND AS WELL AS WATER-SIDE SITES
  - ECONOMICAL FULL-SCALE TEST FOR CERTIFYING STANDARD DESIGN
- O CAPABILITY FOR INCREMENTAL POWER BLOCK ADDITIONS AND MODULE REPLACEMENT
- O SAFETY-RELATED EQUIPMENT LIMITED TO REACTOR MODULE AND SERVICE SYSTEMS
- O INHERENT, PASSIVE SHUTDOWN HEAT REMOVAL FOR LOSS-OF-COOLING EVENTS
- O INHERENT, PASSIVE REACTIVITY SHUTDOWN IN FAILURE-TO-SCRAM EVENTS
- O LOW PRESSURE REACTOR COOLANT (~ATMOSPHERIC) ALLOWING SIMPLIFIED CONTAINMENT

PRISM POWER PLANT (3 POWER BLOCKS)



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PRISM DESIGN DATA

✱ OVERALL PLANT	
– NUMBER OF REACTOR MODULES	NINE
– NET ELECTRICAL OUTPUT	1395 MWe
– NET STATION EFFICIENCY	32.9%
– TURBINE THROTTLE CONDITIONS	955 psia/540^{o}F (SAT'D)
✱ REACTOR MODULE	
– THERMAL POWER	471 MWt
– PRIMARY SODIUM INLET/OUTLET TEMPERATURE	625^{o}F/905^{o}F
– SECONDARY SODIUM INLET/OUTLET TEMPERATURE	540^{o}F/830^{o}F
✱ REACTOR CORE	
– FUEL	METAL (Oxide Backup)
– REFUELING INTERVAL	18 MONTHS
– BREEDING RATIO	>1

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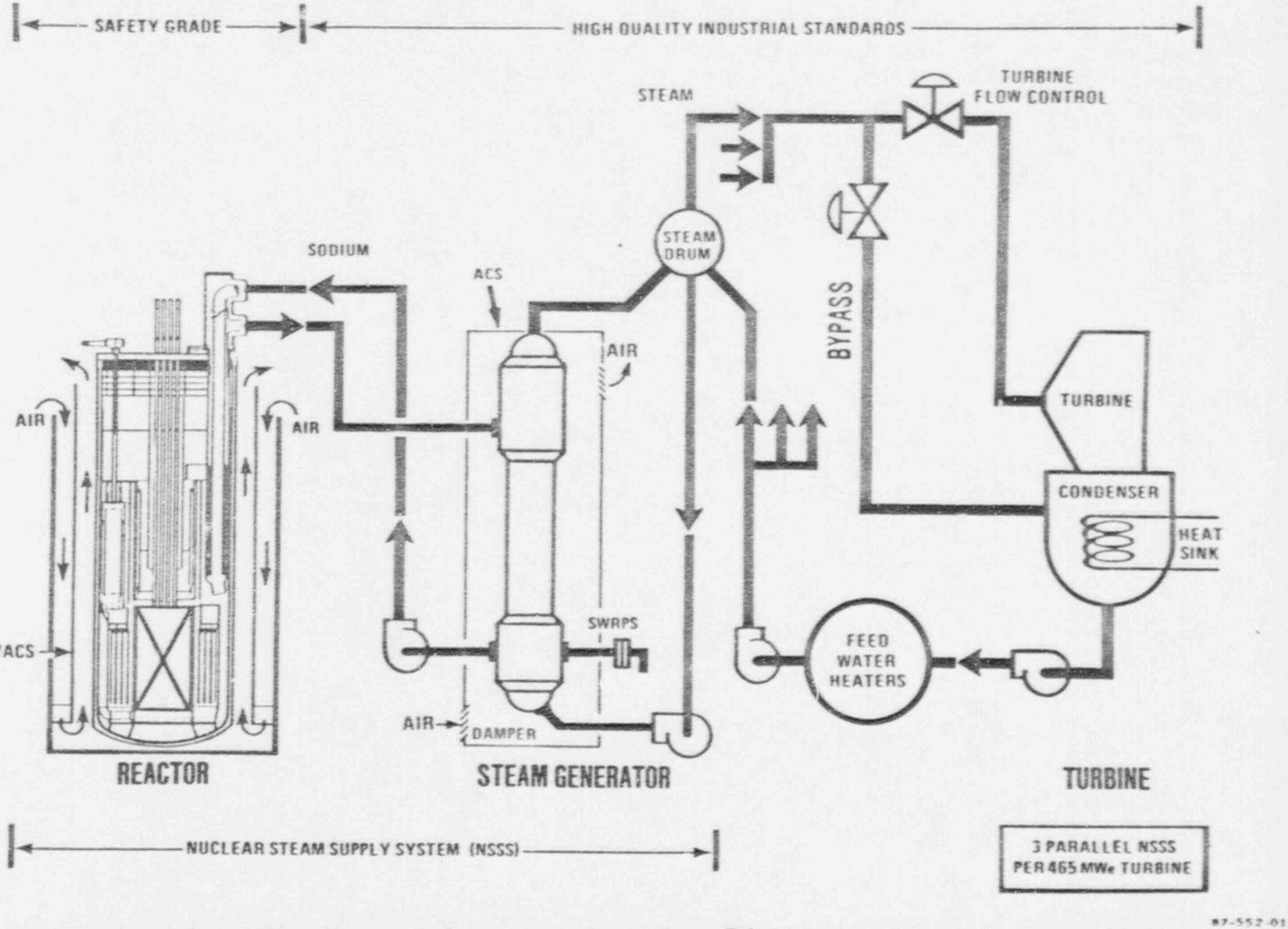
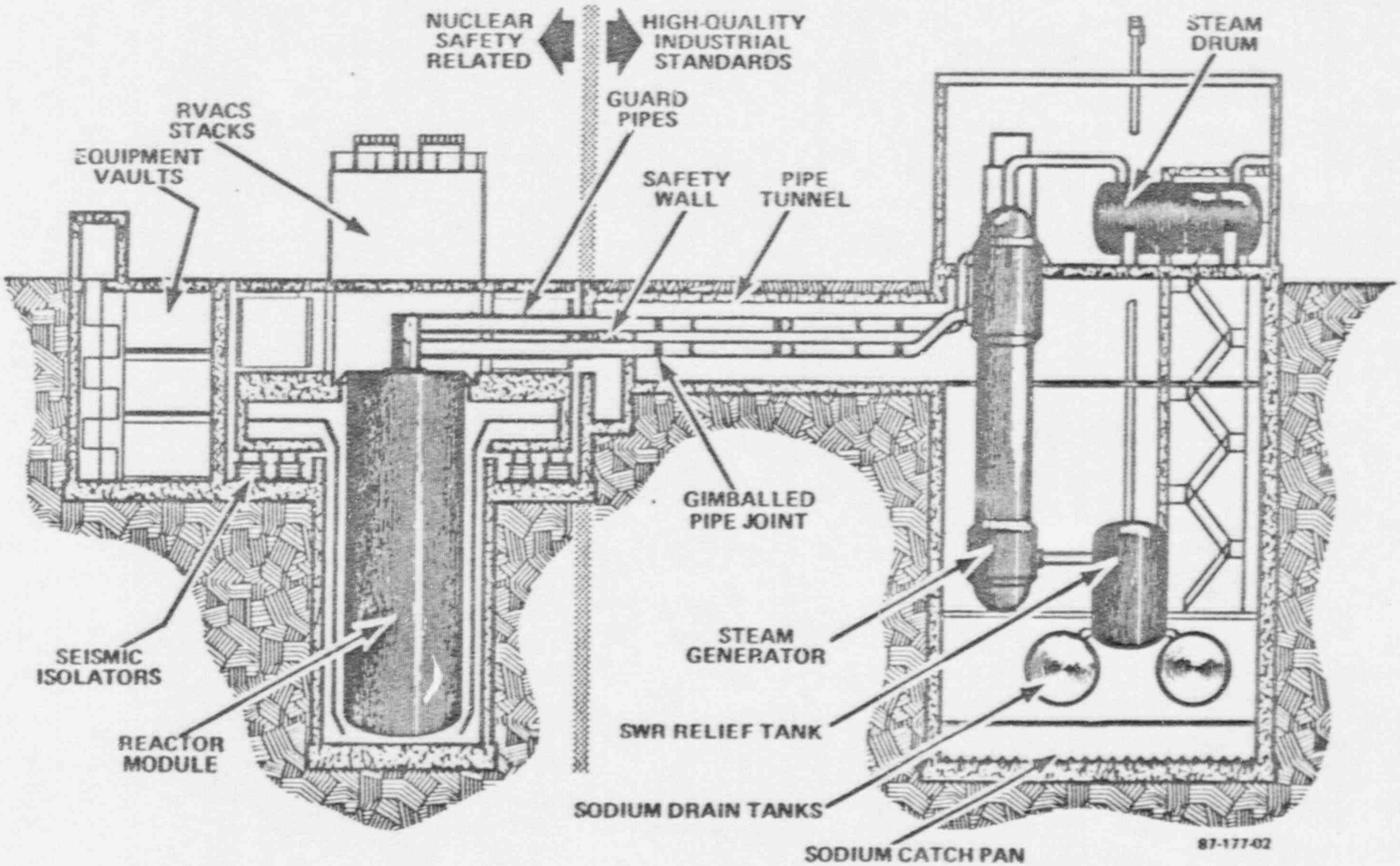
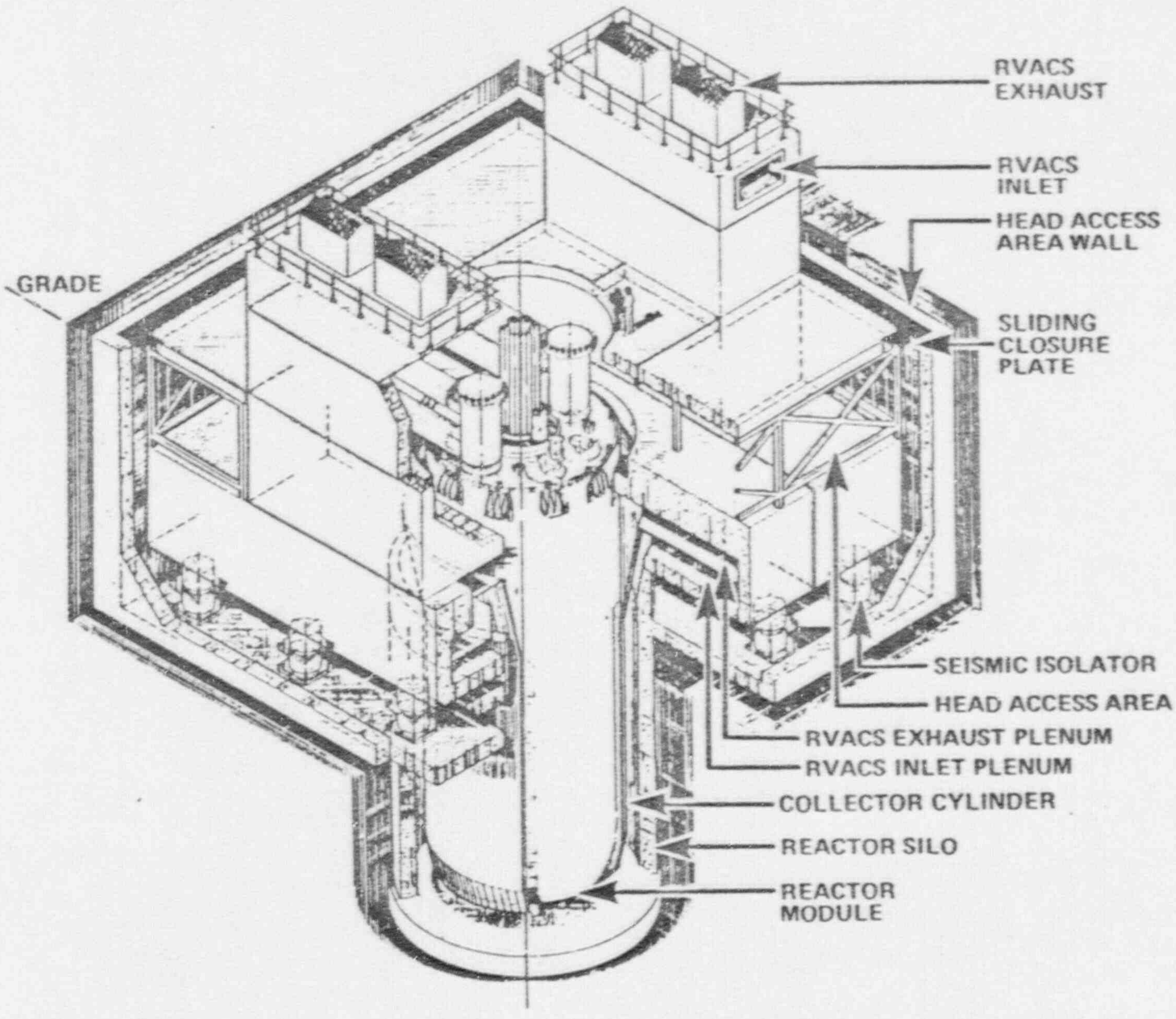
PRISM MAIN POWER SYSTEM



PRISM NUCLEAR STEAM SUPPLY SYSTEM



PRISM REACTOR FACILITY ARRANGEMENT



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RADIOLOGICAL SABOTAGE RESISTANCE

SABOTAGE THREATS:
- o HAND-PLACED EXPLOSIVES
- o HAND-HELD ROCKET LAUNCHER

MITIGATORS:
- o BELOW-GRADE REACTOR
- o PASSIVE REACTOR SHUTDOWN/COOLING
- o PHYSICAL PROTECTION SYSTEM
- o RESPONSE TEAM

TARGETS: ABOVE GRADE SAFETY-RELATED FEATURES
- o RVACS STACKS
- o HAA ROOF DECK

THREAT CONSEQUENCES

- o HAND-PLACED EXPLOSIVES -- ANALYSIS SHOWS RESPONSE TEAM EFFECTIVE BEFORE EXPLOSIVE PLACED AND DETONATED

- o ROCKET-DELIVERED EXPLOSIVES -- ATTACK FROM OUTSIDE PROTECTED AREA CAN DAMAGE RVACS STACKS BUT NOT HAA ROOF DECK. EXPECT SOME STACK BLOCKAGE BUT RVACS EXTREMELY TOLERANT

DESIGN APPROACH FOR MULTI-REACTOR CONTROL

* ONE CONTROL ROOM WITH CAPABILITY GOAL OF ONE OPERATOR PER POWER BLOCK, PLUS SUPERVISOR

  - O OPERATE EACH POWER BLOCK AS A UNIT TO MEET PLANT LOAD DEMAND

  - O OPERATE REACTORS IN A POWER BLOCK AS A GANGED TRIO (OR PAIR) NORMALLY; EXCEPTION IS WHEN ONE IS BEING STARTED UP OR SHUT DOWN

* HIGHLY AUTOMATED CONTROL SYSTEM

  - O LOAD FOLLOWING RANGE, 25-100% (EACH POWER BLOCK)

* SEPARATE REACTOR PROTECTION SYSTEM

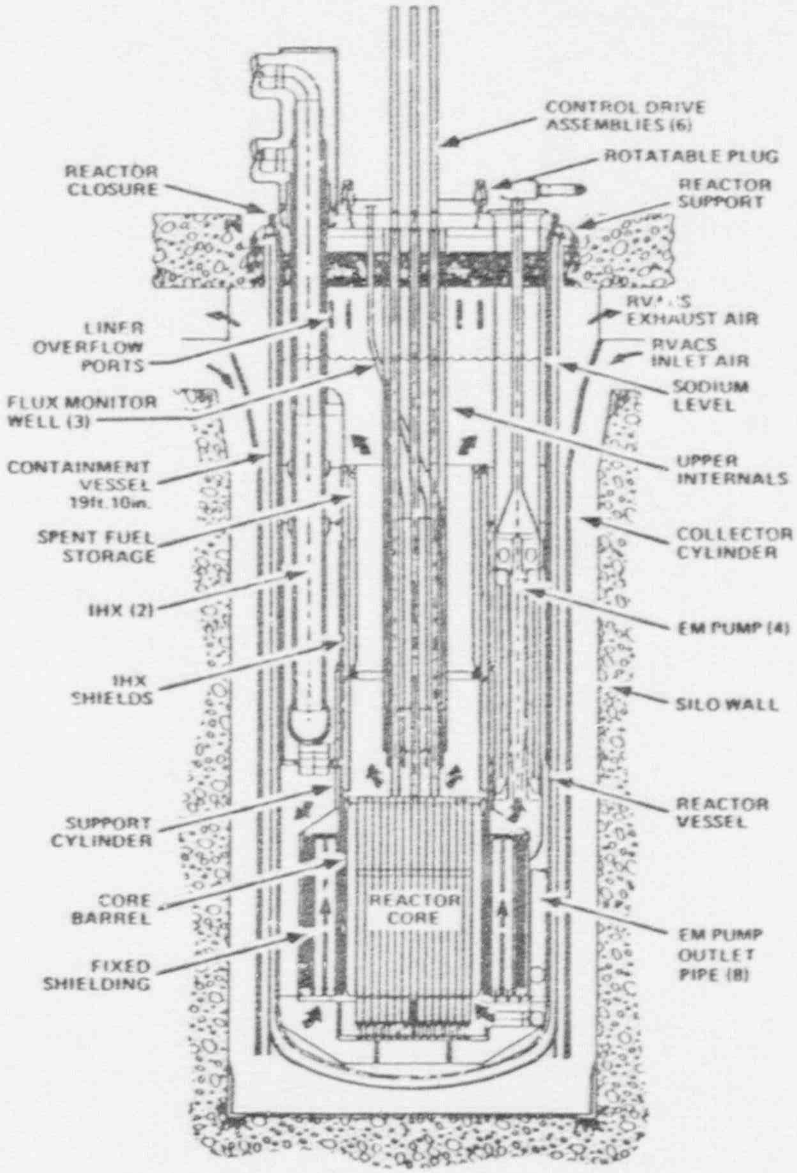
  - O SAFETY RELATED FUNCTIONS IN HARDENED VAULTS AT REACTOR MODULE

  - O NOT DEPENDENT ON CONTROL ROOM

  - O FULLY AUTOMATIC

PRISM IS EXPECTED TO HAVE EXCELLENT OPERABILITY

ADVANTAGES OF PRISM REACTOR CONFIGURATION



- LONGER, SLENDER REACTOR DESIGN GIVES IMPROVED NATURAL CIRCULATION FOR SHUTDOWN HEAT REMOVAL
- LONGER, SLENDER IHX GIVES IMPROVED INTERNAL FLOW DISTRIBUTION
- SMALLER REACTOR SIZE IMPROVES TEMPERATURE MARGINS FOR RVACS OPERATION
- SMALLER VESSEL DIAMETER PERMITS A SIMPLE, STIFF HEAD CLOSURE
- SMALLER REACTOR SIZE PERMITS DIRECT, SIMPLE VESSEL SUPPORT
- LARGE DISTANCE FROM SODIUM TO HEAD CLOSURE REDUCES TEMPERATURE TRANSIENT EFFECTS AND PERMITS SIMPLER HEAD INSULATION
- HIGH VERTICAL STIFFNESS ELIMINATES NEED FOR SEISMIC VERTICAL ISOLATION; HORIZONTAL ISOLATION IS EASIER
- SMALLER REACTOR SIZE IMPROVES FACTORY FABRICATION AND SHIPPING

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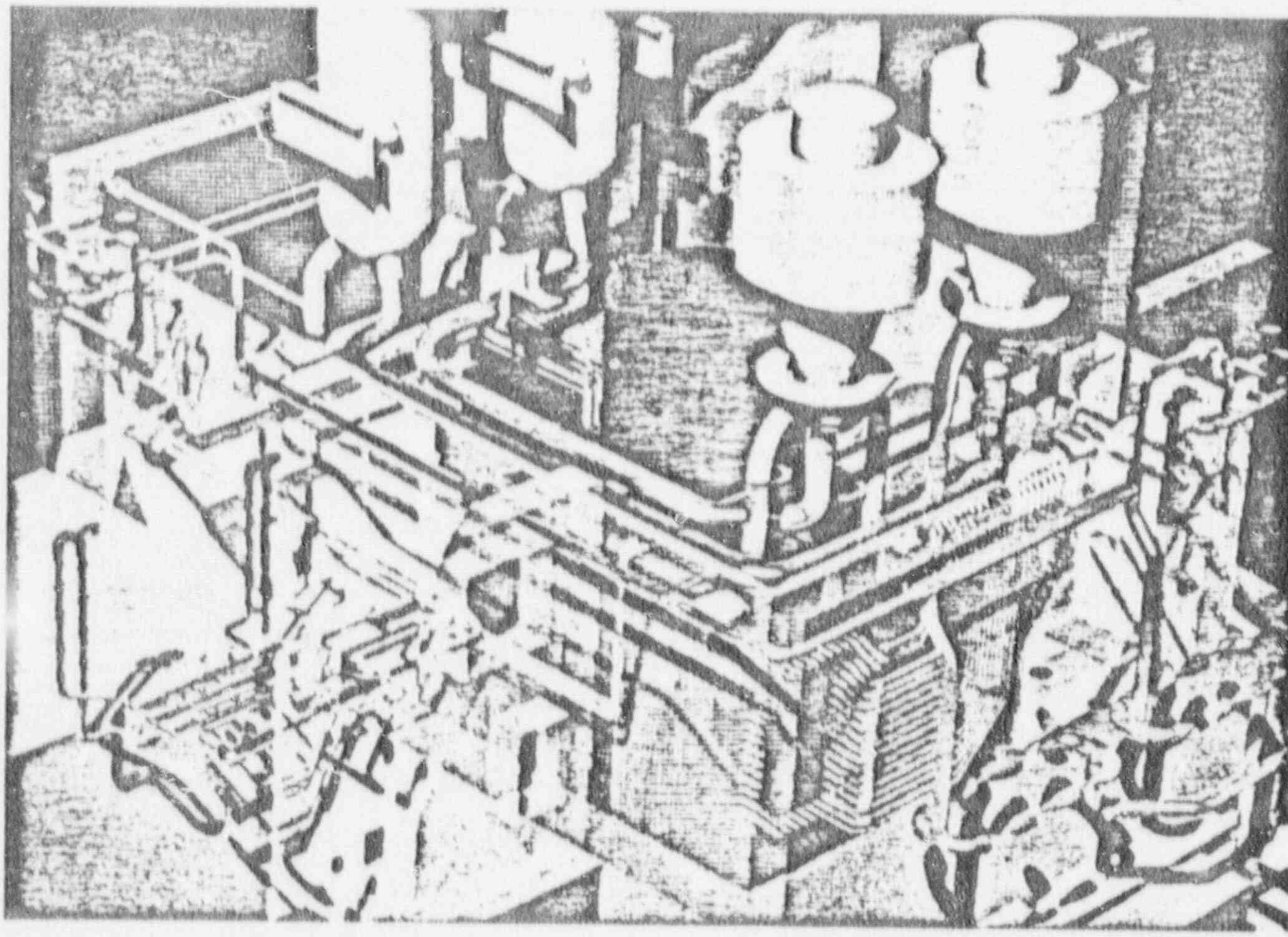
nuclear engineering INTERNATIONAL

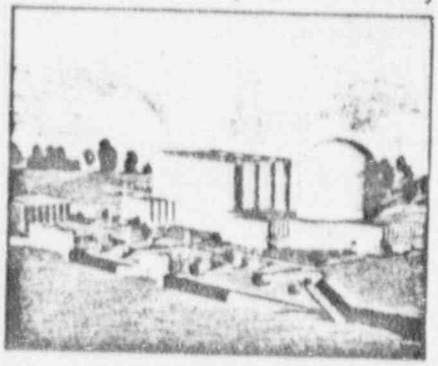


CANDU 3 combines safety with simplicity – wallchart inside
Remote technology
Non-destructive testing

CANDU 3 aims to provide a smaller, cheaper, more reliable alternative

By K. R. Hedges and E. M. Hinchley

Work is well under way on the detailed design stage of the CANDU 3, a new plant which will combine a simpler design with improved passive safety. AECL says that CANDU 3 will be faster and cheaper to build, simpler and safer to operate, and easier to maintain than earlier units.



Atomic Energy of Canada Limited's new smaller (450MWe) standard series of CANDU reactors is now well into its detailed design phase. More than 200 AECL designers and additional outside specialists have brought the CANDU 3 to a stage where the first plants can be built for service later this decade.

Key features and targets of the design include:

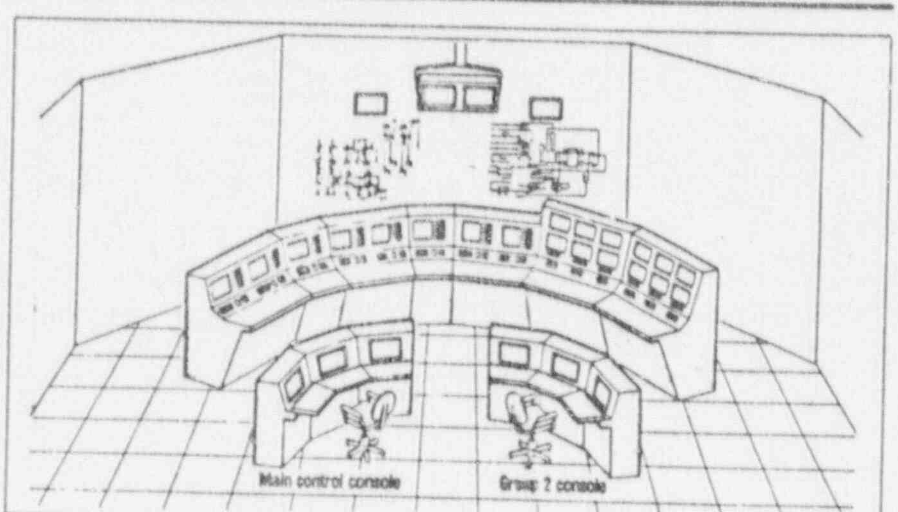
- Standardization to suit a wide variety of site conditions, with minimum site-specific design.
- Simplification, with fewer components.
- An advanced design process ensuring standardized, fully integrated computer databases for drawings and documents.
- Substantial licensing acceptance by the regulatory authority in advance of project commitment.
- One hundred year lifetime for any component that cannot be replaced within a 90 day shutdown.
- A more open layout to facilitate fast construction (open top construction) and easy maintenance.
- A 94 per cent capacity factor target, based on 18 day maintenance outages every two years and a 90 day extended outage at most every 12 years.
- Simplifications and improvements in the safety system, with a target frequency of 10^{-6}/y for severe core damage accidents.
- A more passive containment system, incorporating a steel liner and no active sprays.
- A layout which enables the plant to be built either by conventional methods or using shop-assembled modules.
- Pressure tubes and calandria tubes which have been manufactured as single fuel channel assemblies for simpler initial construction and easy replacement.
- A simpler, single-ended, on-line refuelling system.
- A revised safety system grouping philosophy, which makes qualification easier and simplifies protection of critical equipment from harsh environments, earthquakes and tornadoes.
- Simplified equipment, such as using mechanical control absorbers instead of liquid absorbers.
- Replacement of earlier centralized control and monitoring computers by a redundant distributed control system and modernized plant display system.
- A consistent, logical approach to the control room design, human factors, alarm handling, automation and event management.

STANDARDIZATION

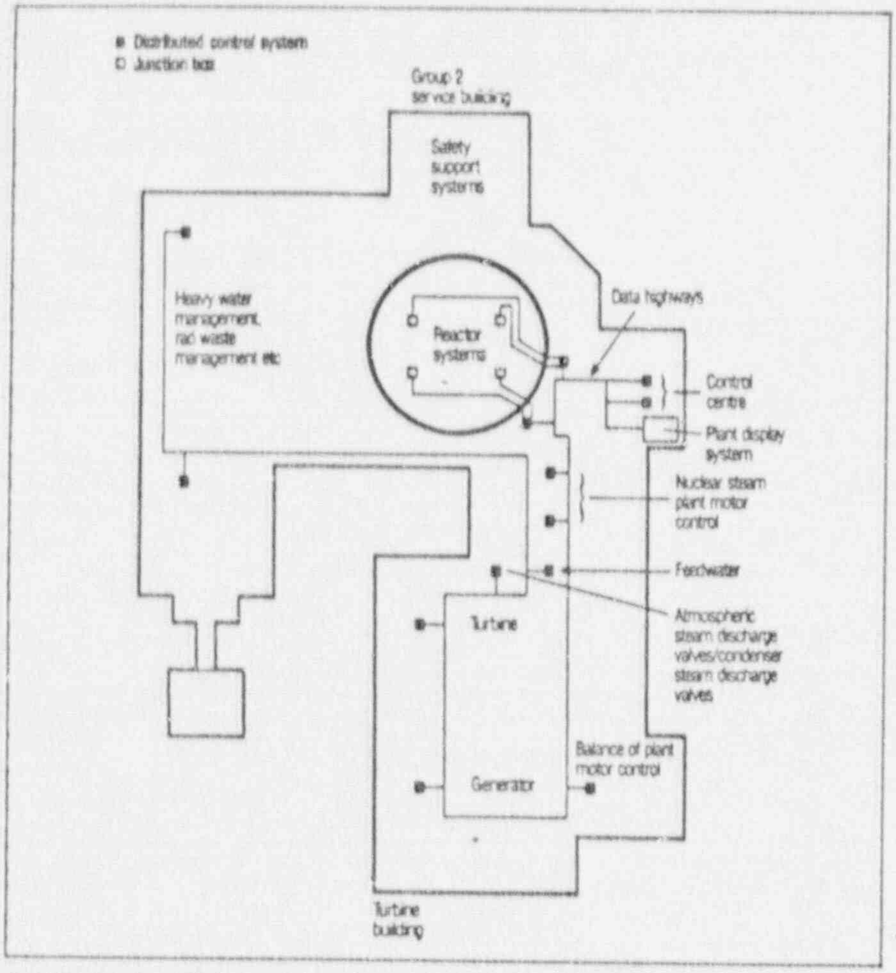
K. R. Hedges is General Manager and Project Director, CANDU 3, and E. M. Hinchley is Manager, Engineering 3, CANDU 3, Sheridan Park Research Community, Mississauga, Ontario, L5K 1B2, Canada.

Comparison of CANDU 3 with present CANDU 6

	CANDU 6	CANDU 3
Net electrical output	665MWe	450MWe
Separate loops in heat transport system	2	1
Number of steam generators	4	2
Number of heat transport pumps	4	2
Number of fuel channels	380	232
Number of fuel bundles per channel	12	12
Number of elements per fuel bundle	37	37
Number of fuelling machines	2	1
Number of reactor control zones	14	8
Number of rods in shutdown system number 1	28	24



▲ The CANDU 3 control room features colour graphic mural mimics, context sensitive displays, and consistency of layout, architecture and ergonomics.



▲ The distributed control system, which replaces the control trunk cabling, control distribution frame, relay logic, analog controllers and comparators used in previous CANDUs.

AECL is designing the CANDU 3 initially as a stand-alone unit suitable for a wide variety of sites. This means that most of the detailed design work would be completed before construction begins. Licensing of the standardized design would be completed, except for site-specific issues, during the initial design period.

The seismic analysis is based on an envelope encompassing ground accelerations up to 0.3g and allowing siting at most potential sites without re-analysis. The degree of tornado protection allows siting in most locations without redesign.

To help with standardization, much of the major equipment is being preselected in advance of the actual award of a contract to build a plant. This has allowed manufacturers to review specifications and suggest design improvements. A series of formal and informal design reviews has also been held with utility staff beginning in conceptual stages. Utility staff have spent extended periods working in the design offices and in return, CANDU 3 designers have had extended assignments in operating CANDU stations to ensure the design meets utility requirements.

FUEL MANAGEMENT

The initial units will use the proven 37 element natural uranium fuel bundles used in existing CANDU 6 and other CANDU reactors. However, in parallel with CANDU 3 detailed design, AECL is developing a new 43 element fuel bundle which will provide further performance improvements from the present reference. It will be particularly well suited to the extended burnup associated with some future alternative CANDU fuel cycles which could use spent fuel from LWRS or slightly enriched uranium. No CANDU 3 design changes will be needed to use these alternative fuels.

DOSE MANAGEMENT

The CANDU 3 design target for occupational radiation doses is no more than 0.40 person-Sv/y in the initial years and up to 0.75 person-Sv/y in final years. The target represents a significant improvement over CANDU experience and an even larger improvement relative to non-CANDU reactor experience. The improvement is being obtained through close attention to material selection, layout of equipment and shielding design for maintenance. Maintenance procedures are reviewed and factored into the design. Improvements have been made in ventilation and air cooling systems to minimize tritium hazards. The usual CANDU feature of on-line refuelling allows immediate removal of any defective fuel.

MAN-MACHINE INTERFACE

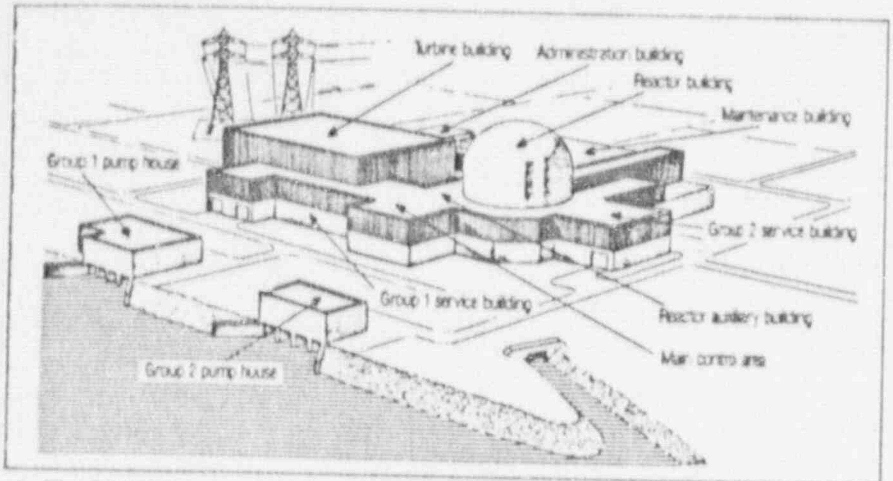
The CANDU reactors make extensive use of redundant computer systems for direct digital control of the main plant functions and for the display, recording and organization of alarms and plant operating information. Since the most recent CANDU plants were designed during the 1970s, there has been significant progress in understanding human behaviour as a component in information processing systems. The CANDU 3 control room design and layouts outside the control room are based on a more task-related approach to the operator's needs, and make cost-effective use of modern equipment.

Design features include:

- Keyboards.
- CRTs and large dynamic colour graphic mural mimics.
- Context sensitive displays.
- Consistency of layout, architecture and ergonomics from a procedure-based design process.
- Intelligent handling of alarm annunciation through categorization, packaging and prioritization of alarms.
- Computer-assisted procedures.
- More computer-assisted testing and transfer of low level tasks to the computers.
- Implementation throughout the plant of a distributed control system based on redundant data highways.

The distributed control system (DCS) replaces the control trunk cabling, control distribution frame, relay logic, analog controllers and comparators used in previous CANDU control systems as well as the central computers used for signal multiplexing and direct digital control. Powerful microprocessors are distributed within the DCS for control of all major functions. The system configuration uses a three channel, dual-redundant, fault-tolerant arrangement similar to previous CANDU computerized control systems.

The plant display system (PDS) provides the main control room interface. The PDS receives information from both the DCS and from other systems, such as safety systems which are not connected



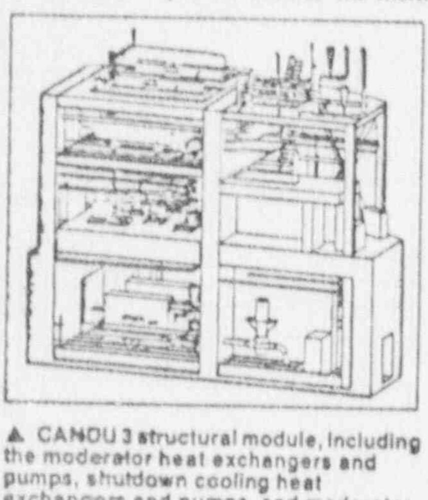
▲ The CANDU 3 layout has been designed to pro de good system separation, access for maintenance and operation, and flexible construction sequences.

to the DCS. The PDS provides various operator displays, an intelligent alarm system, logs and records of operating data, and a plant database.

SAFETY ASPECTS

The Canadian safety philosophy for common cause events assigns all plant systems to one of two groups (group 1 and group 2). Each group is capable of shutting the reactor down, cooling the fuel and monitoring the plant. In the CANDU 3 adaptation of this approach, group 1 systems are those primarily dedicated to normal plant power production, while the physically separate group 2 systems include safety and support systems and have an additional role in mitigating the effects of any accident.

The principal station structures are largely self-contained with minimal connections to other structures. Seismic and tornado qualification is restricted to areas containing group 2 equipment and a few other key areas such as the main control room and the irradiated fuel bay.



▲ CANDU 3 structural module, including the moderator heat exchangers and pumps, shutdown cooling heat exchangers and pumps, and moderator poison. The use of open-top construction, a very heavy lift crane, and externally assembled modules has shortened the predicted construction schedule to 35 months.

The reactor control system can shut down the reactor automatically and completely for all but the most severe postulated accidents. Each of two separate, independent and redundant shutdown systems is capable of immediate and complete automatic shutdown for all accidents.

- Shutdown system number 1 actuates a set of 24 fail-safe spring-assisted rods which drop into the reactor under gravity. These fast-actuated rods drop into the low pressure moderator, which allows them to be simple and reliable. The testable three-channel system is designed to be fully capable even with some of the rods disabled or with groups of sensors unavailable.
- Shutdown system number 2 independently injects a neutron absorbing poison (gadolinium nitrate) into the low pressure moderator. It is triggered by a completely separate and diverse set of channelized sensors and logic. The injection energy is stored in gas-filled tanks and multiple nozzles are used.

Both systems are fail-safe on loss of electrical power and are designed for full testability with good operator interfaces. The latest quality assurance methodologies are used for both hardware and software.

The CANDU 3 emergency core cooling system also uses stored energy from gas-filled tanks for reliable high pressure actuation. The directed injection feature removes any dependency on continuing operation of the main pumps and provides a more flexible response. Automatic signals for all breaks in the primary circuit mean that no operator actions are needed to ensure emergency cooling.

The containment system has been simplified and made more passive than in previous designs by eliminating a water dousing system featured in other CANDU containments and by adding a steel liner.

SEVERE ACCIDENTS

Estimates of severe accident frequencies in CANDU reactors are significantly lower than for most LWRS. For instance, the automatic reactor control system in combination with two simple, independent, rigorously tested shutdown systems has led to a frequency estimate of 3×10^{-8} events/y for a power excursion coupled with a failure to shutdown (ATWS).

The overall probability of severe core damage in recently built CANDU reactors is less than 10^{-5}/y. The CANDU 3 is aiming for further improvements in this low frequency and has set a target of less than 10^{-6}/y.

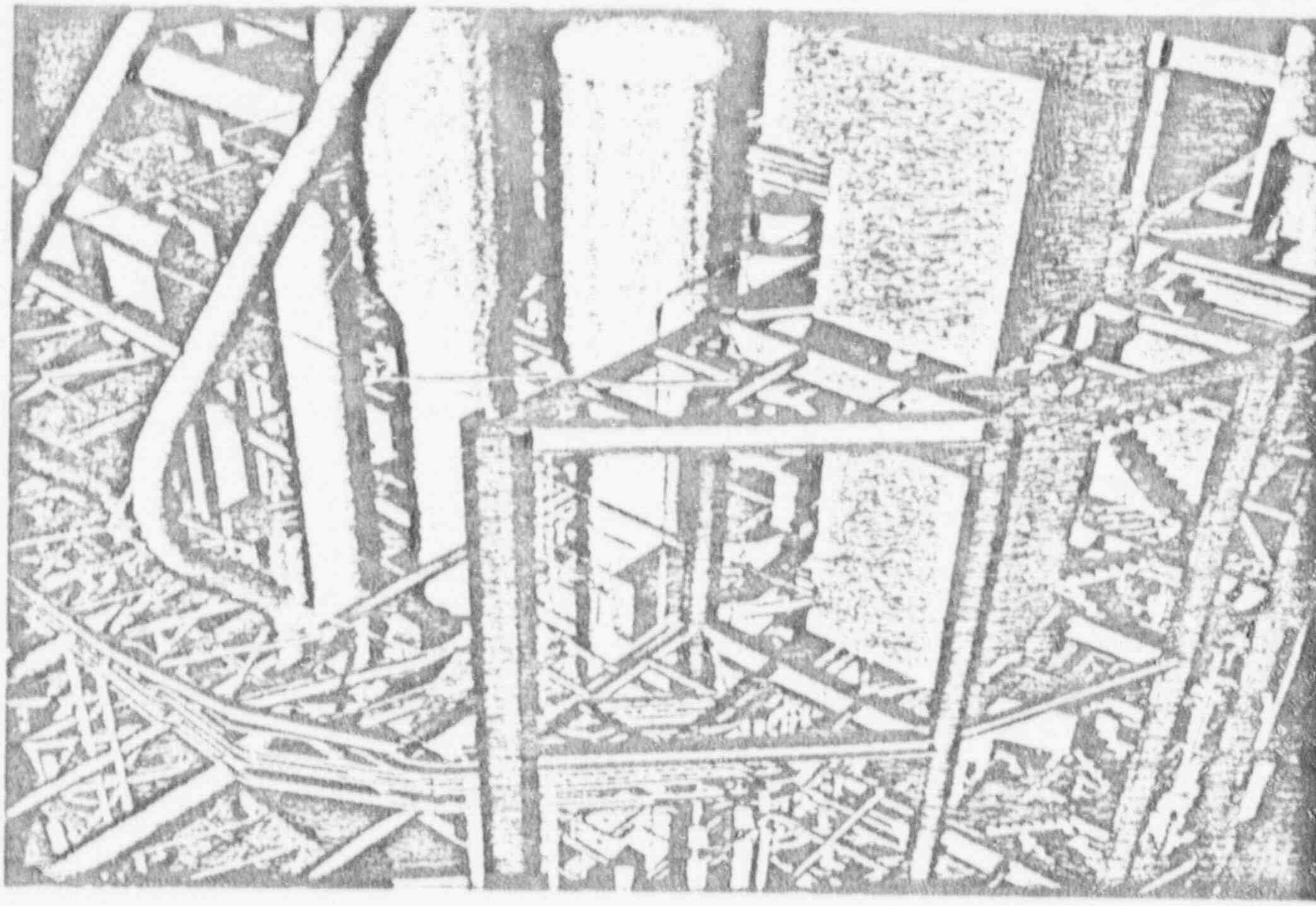
In addition to a low frequency of severe accidents, CANDU plant response to increasingly severe accidents is gradual, without sudden changes in behaviour. In a severe loss of coolant accident, combined with a postulated failure of emergency coolant injection, the moderator system acts as a dispersed emergency heat sink for fuel heat, preventing fuel melting and maintaining the fuel channels intact even if the channels contain no coolant at all. If in turn the moderator is in some way lost, the water-filled shielding system can (in most failure sequences) prevent or delay for many hours melt-through of the shielding system.

CANDU 3 safety levels have been calculated as consistent with so-called "inherently safe" reactors. Further enhancements have been investigated for later versions of the CANDU 3. Natural circulation systems for decay heat removal from the moderator and from the shutdown cooling system are being investigated. Passive isolation and containment pressure suppression systems are also possible for later versions.

LAYOUT AND CONSTRUCTION

The station layout has been planned to provide good system separation, access for maintenance and operation, and flexible construction sequences. The amount of equipment needing seismic or environmental qualification has been reduced through careful segregation in the layout. Layout will also contribute to dose reduction in operation.

During construction, the layout accommodates many separate contractors and flexible material handling without significant interferences. The use of open-top construction, a very heavy lift crane and externally assembled modules has shortened the predicted construction schedule to 35 months (plus contin-



▲ Computer generated illustration of the CANDU 3 design.

gency) for the first CANDU 3 (first concrete to in-service).

Vertical installation of a component, such as a steam generator, through the open roof of the reactor building shortens installation time compared to horizontal access methods from two weeks to one day. Other systems are assembled either off-site or on-site into modules weighing from 10t to 500t, which can then be lifted into place. About 30 heavy lifts will be made into the reactor building, divided more or less equally between modules and major components.

Additional studies have been done on offsite construction techniques based on much larger building-sized modules which could be brought near a site by ship and moved with special multi-wheeled transporters. This technique is well-suited to some sites, but will not be used for the first units. The CANDU 3 layout uses dimensions and building separations adaptable to this large scale modular construction.

KEEPING COSTS DOWN

The CANDU 3 cost targets have aimed to keep energy costs competitive with larger nuclear units and competitive with fossil units of a similar size. Capital costs are now estimated at about US $1600/kW. In combination with the traditional low fuelling costs of the natural uranium CANDU fuel cycle, and projected economies in operation and maintenance, the CANDU 3 will be cost competitive in most markets.

The capital cost savings come in engineering, equipment and construction. Significant savings in engineering costs accrue from the accelerated use of integrated project-wide computer aided design methods. Moreover, the standardized design approach allows the engineering costs to be amortized over several CANDU 3 units to be built in the next few years.

Equipment cost savings are being made through simplification (fewer components) and by careful review of requirements for the equipment.

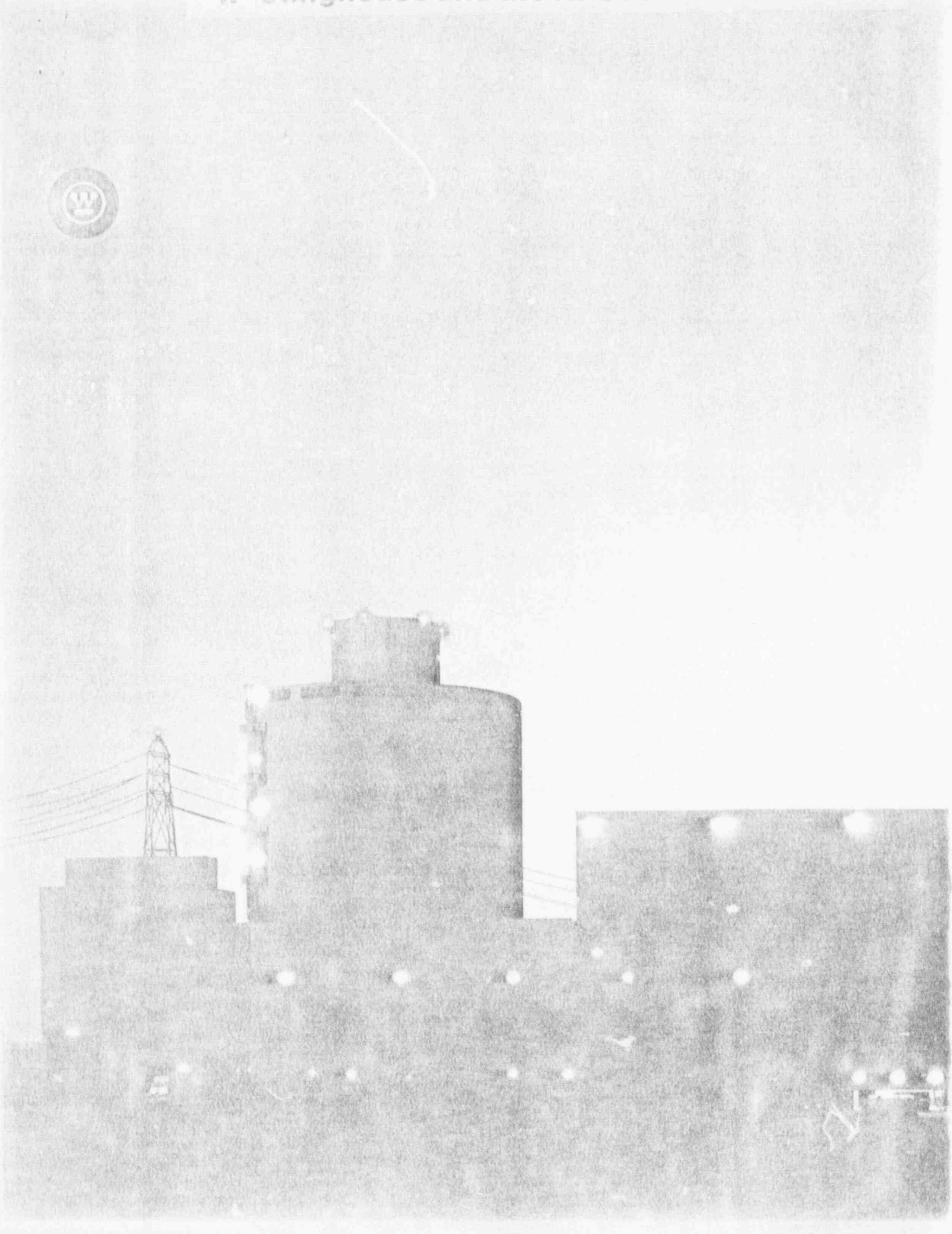
Construction costs are being reduced by designing for faster, easier construction. The improved layout and modularization techniques significantly shorten the construction schedule and lead to lower charges for interest during construction. The designs of internal concrete structures have been greatly simplified through the use of more steel platforms and supports.

The distributed control system has reduced the amount of cabling and the construction effort needed for cable installation and termination. The computer techniques used in design are linked directly to the construction management to give a more efficient and error-free transfer of information from design. In addition, The substantial completion of the design and detailed reviews throughout the design process ensure savings from much less rework during the construction phase.

CURRENT STATUS

Early conceptual work on the CANDU 3 began in 1982 and by 1987 a substantial design team was committed to the standardized design. Now in mid-1990 the conceptual design work on all systems is essentially complete and detailed design is under way. The new design tools and software are in place and working in a production mode. Vendors are being selected for major components and vendors, consultants and utilities are directly involved in most design areas. The standardized design is scheduled for completion in early 1992. The schedule for the first anticipated CANDU 3 unit foresees a contract effective in 1991, first concrete in 1993 and in-service in 1996. □

stinghouse and the AP600



AP600

Westinghouse AP600, the plant of the future, has a solid pedigree which reflects cooperative efforts in vision and development. Over the past five years, the Electric Power Research Institute (EPRI) and the U.S. Department of Energy (DOE) have sponsored us in creating and in refining the AP600 concept and its design.

Recognizing the trend toward smaller generating units and the demand for enhanced safety assurance and operability, EPRI funded an initial Westinghouse program in 1985 to study the potential for an innovative advanced light water reactor. In 1986, DOE added support; together, the programs have produced a complete conceptual design for a 600 MWe pressurized water reactor featuring a simplified reactor coolant loop, passive safety systems, and modular construction.

With 100 percent of the conceptual design effort completed, the AP600 is ready to undergo detailed design, vigorous NRC review, and design certification. Westinghouse looks forward to continued DOE and EPRI backing to restore safe nuclear power to the nation's future energy options.

To restore the nuclear option to U.S. utilities by the mid-1990s–in time to help satisfy growing demand

Westinghouse AP600 will meet the challenge for safe, clean power

Safe nuclear power in America's near-term energy future will help satisfy the rapidly growing demand for electricity, curb the alarming rise of oil imports, and alleviate environmental damage caused by burning fossil fuels.

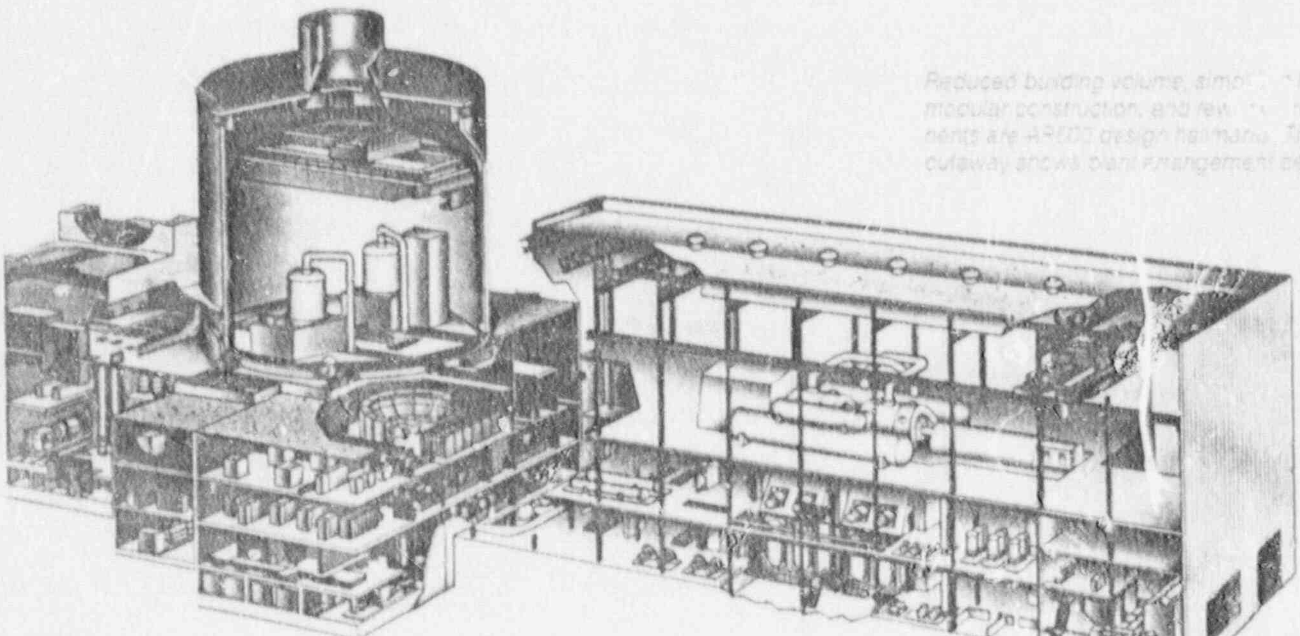
The new Westinghouse 600 MWe AP600 plant can be available by the mid-1990s to fulfill this role. It is designed to be so safe, affordable, reliable, and simple to build and operate that the public, utilities, regulators, and financial community alike will approve it.

AP600 merges the best elements of our world-standard Pressurized Water Reactor (PWR) technology with innovations to improve safety and performance, minimize costs, construction time, and investment risk–and potentially revolutionize the way nuclear plants are built.

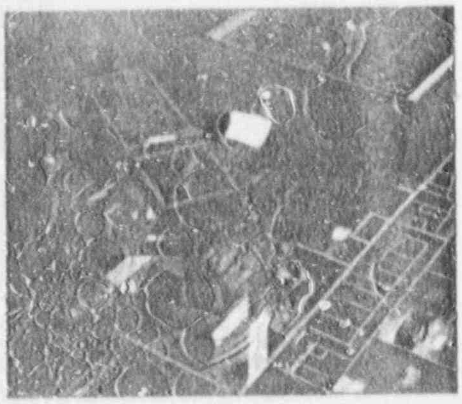
AP600 is the right new plant at the right time, with the right attributes to restore confidence in nuclear power:

- Safety first
- Major plant simplification
- Credentials to earn NRC design certification
- The right team with the most experience and resources
- Dedicated Corporate support

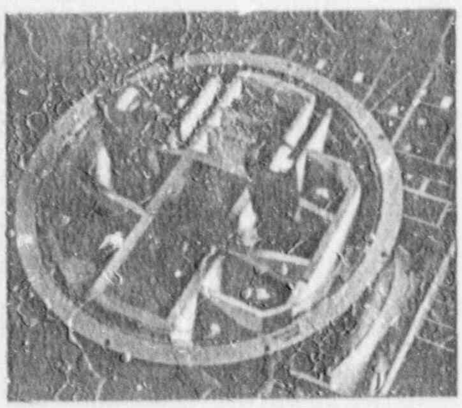
Reduced building volume, simpl... layout, modular construction, and fewer compo-nents are AP600 design hallmarks. This cutaway shows plant arrangement deta...



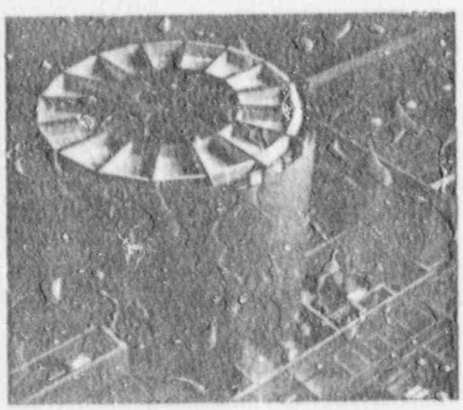
Westinghouse sets the standards for the nuclear industry



A view of the upper part of the containment building above the operating floor shows the steel containment vessel, airflow baffles and polar crane.



Looking inside the reactor, steam generators, pressurizer and core makeup water storage tanks are visible.



Cooling water storage tanks and air inlets for the containment cooling system are located in the top portion of the containment building.

Safety motivates our design work

Westinghouse puts safety first. We help set the standards for the nuclear industry on reactor safety and service, disciplined operator training, and safe waste management. Safety motivates the AP600 design. Simplification brought many economies and efficiencies–but never by compromising safety.

AP600 is conservatively based on proven, licensed technology, with a new emphasis on safety features that rely on natural forces for reduced complexity and improved operability. AP600 incorporates the following:

- Operating experience from 3000 PWR reactor-years
- Improvements and lessons learned from Three Mile Island
- Input from other government advanced reactor programs, utilities, regulators, and the public

Natural forces enhance safety

AP600 safety systems rely on the dependable natural forces of gravity, natural circulation, and cooling by convection and evaporation to protect the reactor in the unlikely event of an accident. Safety injection, residual heat removal, containment spray, and containment cooling perform automatically if they are ever required. These systems are technically described as "passive," since no active or external power sources are required for their effective operation.

These systems eliminate the need for large amounts of piping, safety-related pumps, emergency diesel generators, and many other complex components and systems. Tests, inspections, and maintenance are reduced. This simplicity enhances attention to safety.

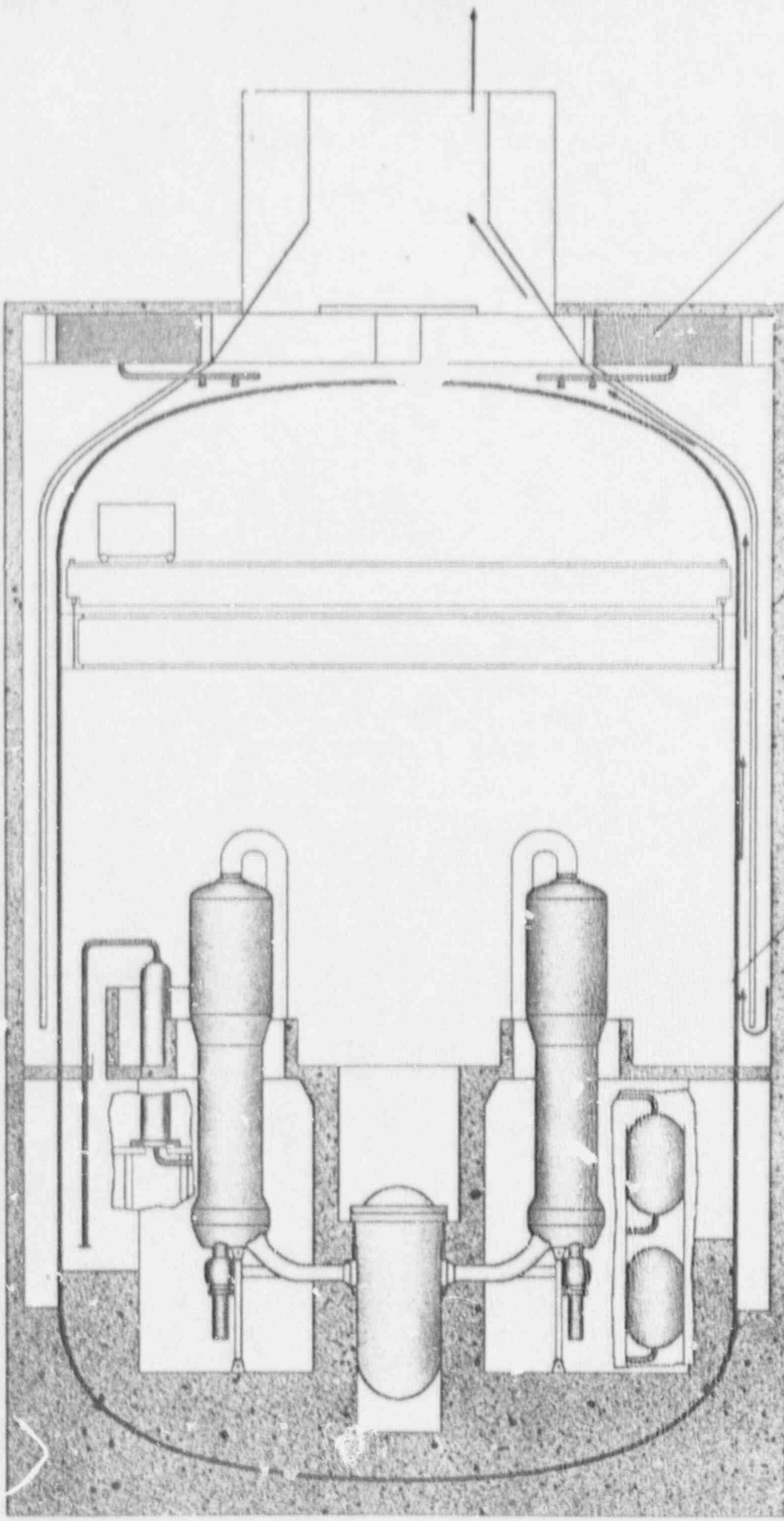
Dependable systems protect plant automatically

Large tanks of cooling water are located above the reactor vessel within the containment to provide water for emergency core cooling injection and decay heat removal. Pipes connect the tanks to the reactor. If ever needed for emergency cooling, fail-safe valves automatically open, allowing water to flow down into the core. The water continues to cool the flooded core indefinitely through natural circulation, which carries heat from the core to the surrounding steel containment vessel.

A concrete shield building surrounds the steel containment vessel, and between building and vessel there is an airflow space. When outside air enters the building, a natural draft transfers containment heat through the airflow space and up through a chimney-like opening in the roof. Cooling is accelerated by spraying gravity-fed water, stored in other tanks at the top of the shield building, over the containment vessel. Over the long term, natural air cooling alone provides all emergency heat removal.

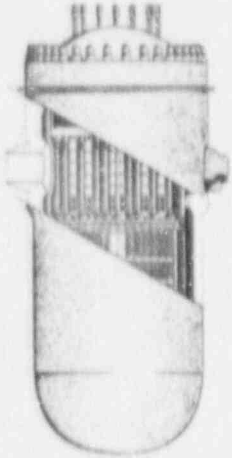
Safety analysis verifies performance

Westinghouse safety engineers are using state-of-the-art Design Basis Accident analysis and Probabilistic Risk Assessment to accurately predict AP600 safety performance. Tests of the features confirm that they will protect the plant and public.


Gravity-fed water, stored in tanks atop the AP600 containment building, sprays over the containment vessel to accelerate initial cooling.
Natural air circulation between the reactor containment structure and surrounding shield building provides dependable containment cooling.
Thick-walled steel containment vessel safely isolates the reactor from the environment.

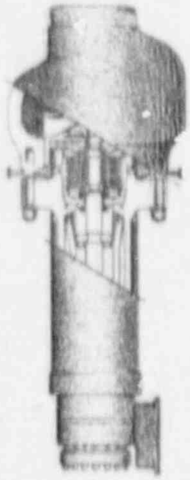
Other AP600 innovations maximize safety and reliability

Low Power Density Core



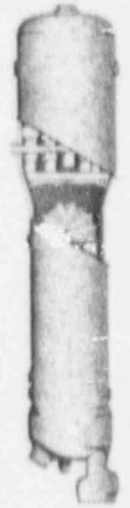
- Extends refueling cycle for increased availability and lower fuel cycle costs (18-24 months)
- Provides increased design margins of safety
- Extends reactor vessel life to 60 years
- Increases fuel cy   fficiency

High-Inertia Canned Motor Pump



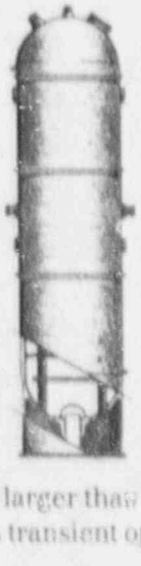
- Incorporates proven reliable motor design
- Eliminates potential for seal leakage
- Reduces maintenance
- Supports cooling flow to coast down reactor on loss of power

Steam Generator



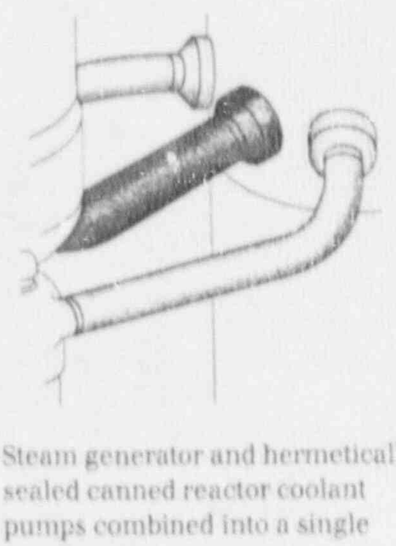
- State-of-the-art materials
- Robotic steam generator inspection and maintenance during refueling
- Reduced worker radiation exposure

Pressurizer



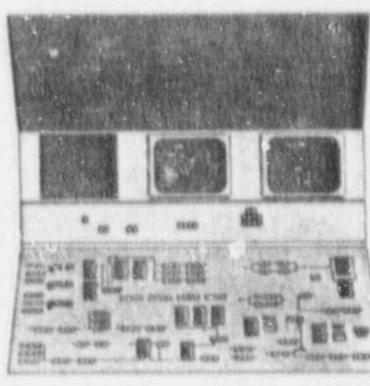
- 30 percent larger than normal
  - –Increases transient operation margins
  - –Improves plant reliability
  - –Eliminates the need for power-operated relief valves

Simplified Reactor Coolant System

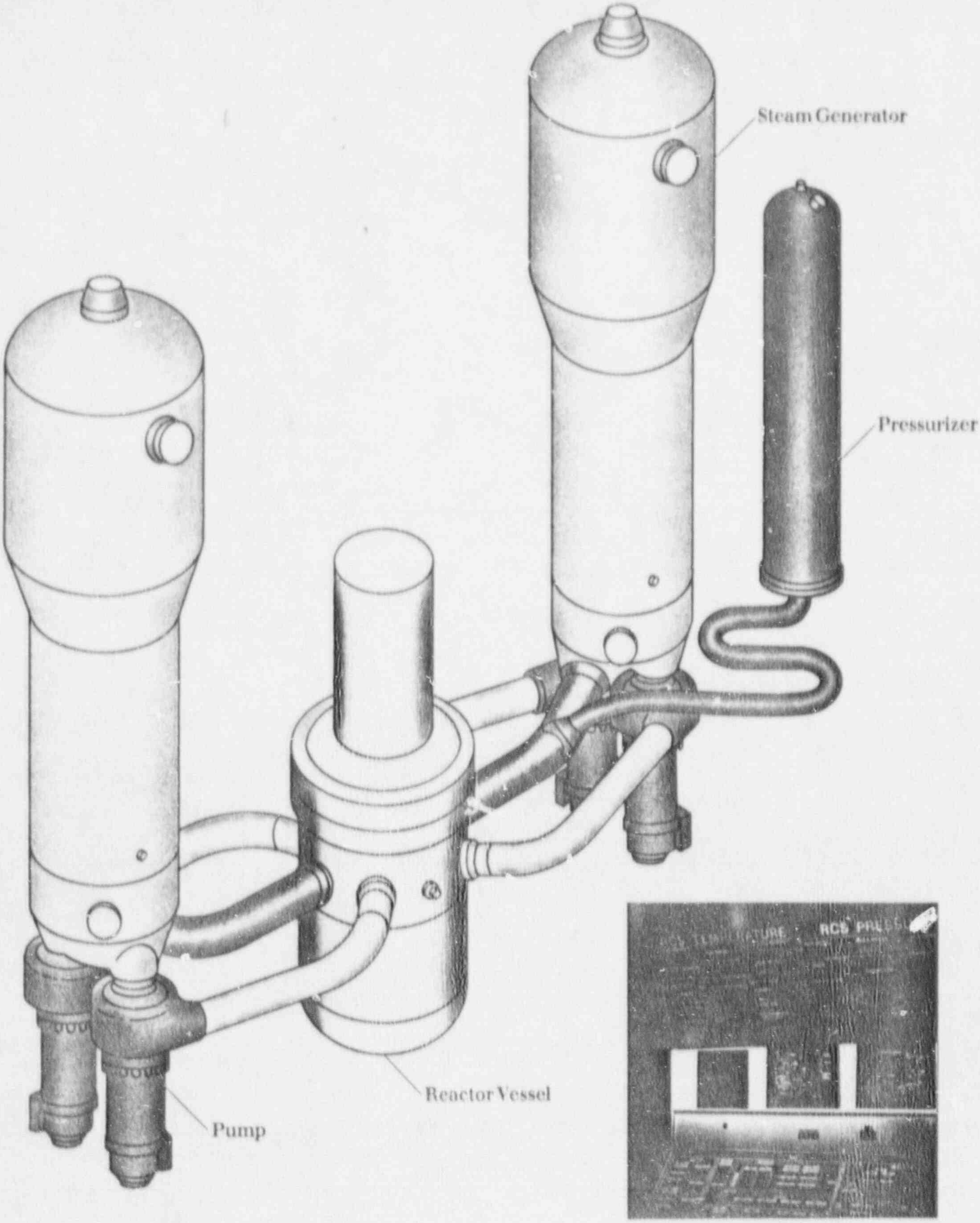


- Steam generator and hermetically sealed canned reactor coolant pumps combined into a single structure
  - –Eliminates need for separate support structures
  - –Increases inspection access

Advanced Instrumentation and Control System



- Digitally multiplexed microprocessor-based controls
  - –Easy to monitor
  - –Improved operability
  - –Reduced chance for operator error
  - –80 percent less control cable

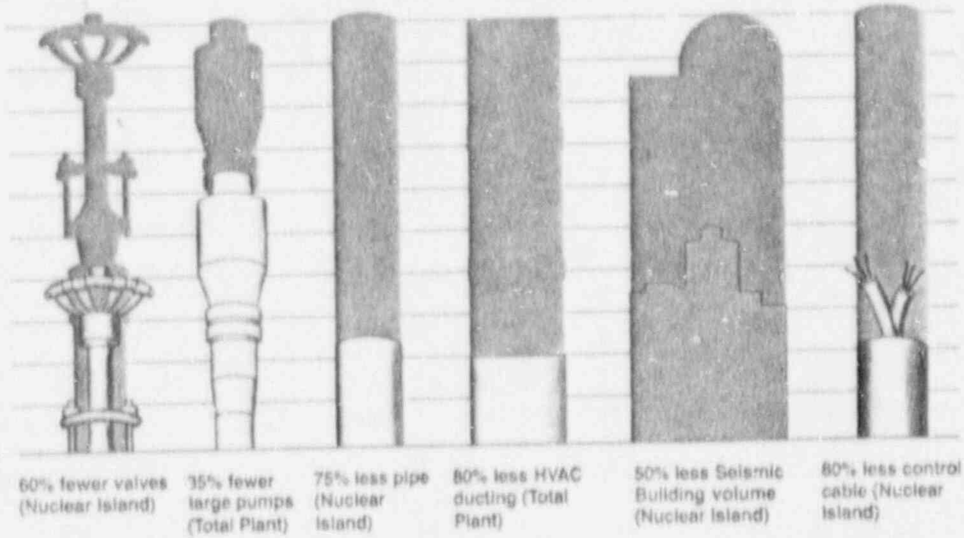


The Westinghouse Advanced Instrumentation and Control System for AP600 will simplify and improve plant controls, displays, and protection systems, for ease of operation.

For unprecedented reliability, operability, constructibility and economy

We simplified and streamlined our world-standard PWR design

AP600 incorporates major simplifications of Westinghouse PWR technology. Compared to a typical 600 MWe plant, the AP600 plant requires fewer valves, pumps, piping, and associated equipment.

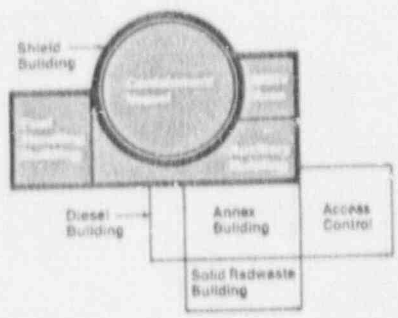


Modularization shortens construction time to 36 months

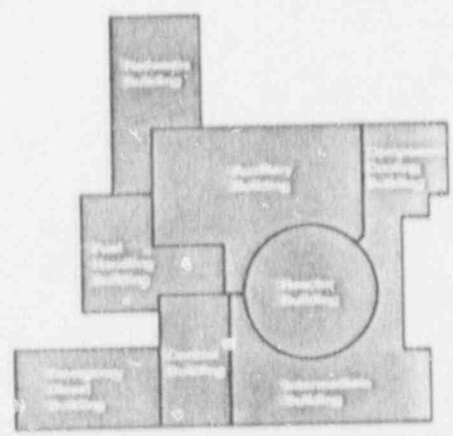
Unique techniques from modular shipbuilding and construction allow large sections of the AP600 to be factory prefabricated in modules. Factory prefabrication enhances quality, increases worker productivity, and reduces capital costs. Modules will be shipped by rail to the site, joined together into larger modules, and inserted into preconstructed areas of the plant.

Modular building methods for field construction allow many areas to be built concurrently. Thanks to this approach, other simplifications, and bulk quantity reductions, AP600 can be built in 36 months.

AP600 Plant Arrangement



600 MWe Reference Plant Arrangement



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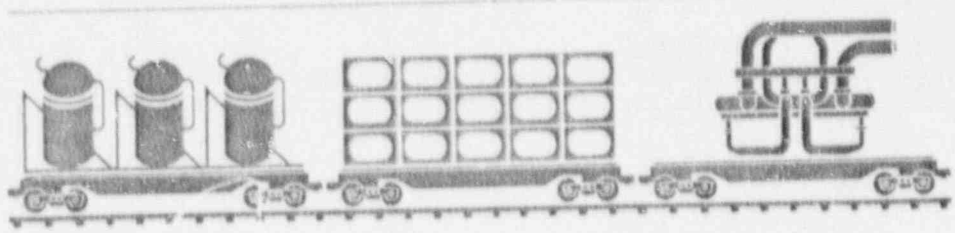
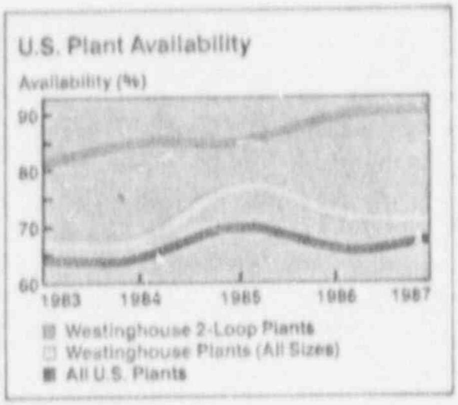


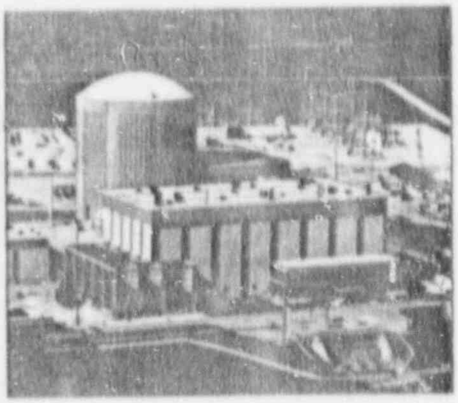
AP600 modules will be [illegible] in 12- by 12- by 80-foot sections, weighing a maximum of 78 tons, to meet all railroad clearance requirements and allow rail shipment to the plant site.

Superior reliability yields 90 percent availability

AP600 is the optimized progression from the Westinghouse two-loop, 600 MWe PWR. Six 600 MWe U.S. plants with Westinghouse nuclear steam supply systems consistently perform 10 percent above the national average in availability. AP600 features a low power density core with more operating margin, proven high-reliability components, and an 18- or 24-month cycle between refueling. These factors support a high plant availability of 90 percent and a capacity factor of 85 percent, resulting in lower electricity costs.



Westinghouse conventional 600 MWe two-loop plants are top performers, with availability 10 percent higher than the national average.



The Kewaunee Power Plant, a Westinghouse 2-loop design, achieves high availability.

Performance meets key requirements

AP600 satisfies key utility and government requirements for a smaller, simplified, naturally safe and affordable nuclear power plant that can be available by the mid-1990s. In addition to safety requirements, Westinghouse fully intends to address and satisfy EPRI's Utility Requirements Document for advanced reactors.

Measurement	Performance Requirement	AP600
Electric Power (MWe)	600	600
Average Annual Availability (%)	87	90
Core Melt Probability (per reactor year)	1 in 100,000	1 in 100,000
Public Risk • Probability of Release (per year) • Magnitude of Release at Site Boundary (rem)	 1 in 1,000,000 25	 1 in 10,000,000 1
Delivery Time (owner commitment to commercial operation, months)	60	60
Occupational Radiation Exposure (man-rem/year)	100	70
Low Level Radwaste (ft³/year)	1750	1750

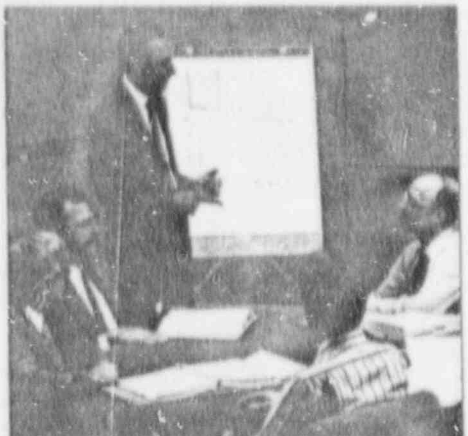
AP600 will meet or exceed safety and utility requirements.

AP600 power rates will be competitive

The cost to produce electricity includes the cost of the capital, the fuel, and operation and maintenance. Detailed preliminary estimates for replicate AP600 plants indicate an overnight capital cost of \$1270 per kilowatt–the cost in 1988 dollars of a completed, licensed AP600 plant exclusive of interest and escalation during construction. This capital cost and the expected lower operating, maintenance, and fuel costs will provide an economically attractive electrical energy source for the future.

Constant attention to financial, schedule, and technical performance data ensures that customer satisfaction is given the highest priority.

The right design to earn NRC approval as a standardized, licensable plant



A dedicated and experienced staff of safety experts will work with the NRC to ensure that AP600 design certification proceeds on schedule.

Westinghouse AP600 working model [illegible] the design offers [illegible] to greater [illegible] and constructibility.

Standardization proceeds with AP600

AP600 represents more than just the best in new nuclear power technology. Its highly engineered, standardized, modular design represents a national commitment to a stable regulatory environment. Under the new 10CFR52 regulations, a combined license can be requested for nuclear plant construction and operation which would be granted after a thorough technical design review. Provisions for full public participation are still included under the new regulations. Accountability for all resulting requirements through construction will be achieved by means of a rigorous "Inspection, Test, Analysis, and Acceptance Criteria" compliance document.

The standardized design offers increased assurance of public safety in the form of highly engineered system simplification and improvements which can be realized through increased regulatory licensing stability.

We will work with NRC to prove that AP600 fully conforms to regulatory criteria

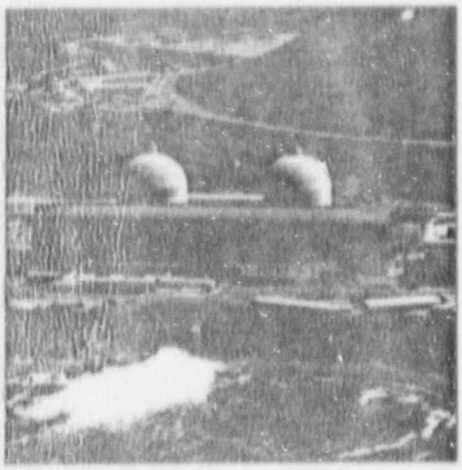
Westinghouse has designed AP600 to meet all applicable codes, standards, regulations, and NRC General Design Criteria. Westinghouse checked the AP600 design against the NRC Standard Review Plan and Regulatory Guides, Unresolved Safety Issues, Generic Safety Issues, and Additional TMI Requirements. As a result, AP600 addresses all the issues and implements all regulatory guidance.

Westinghouse is working closely with the NRC to ensure that the AP600 safety features provide for the protection of public health and safety. In February 1989, we submitted a 900-page plant description to the NRC for an early safety review. This report thoroughly documents key design information for the entire AP600 plant.

NRC position favors moving forward

The NRC has specifically considered future licensing criteria for evolutionary designs such as AP600. The Commission expects that the safety margins of new passive systems can be verified by analysis and testing without a requirement for full plant demonstration before licensing.

NRC



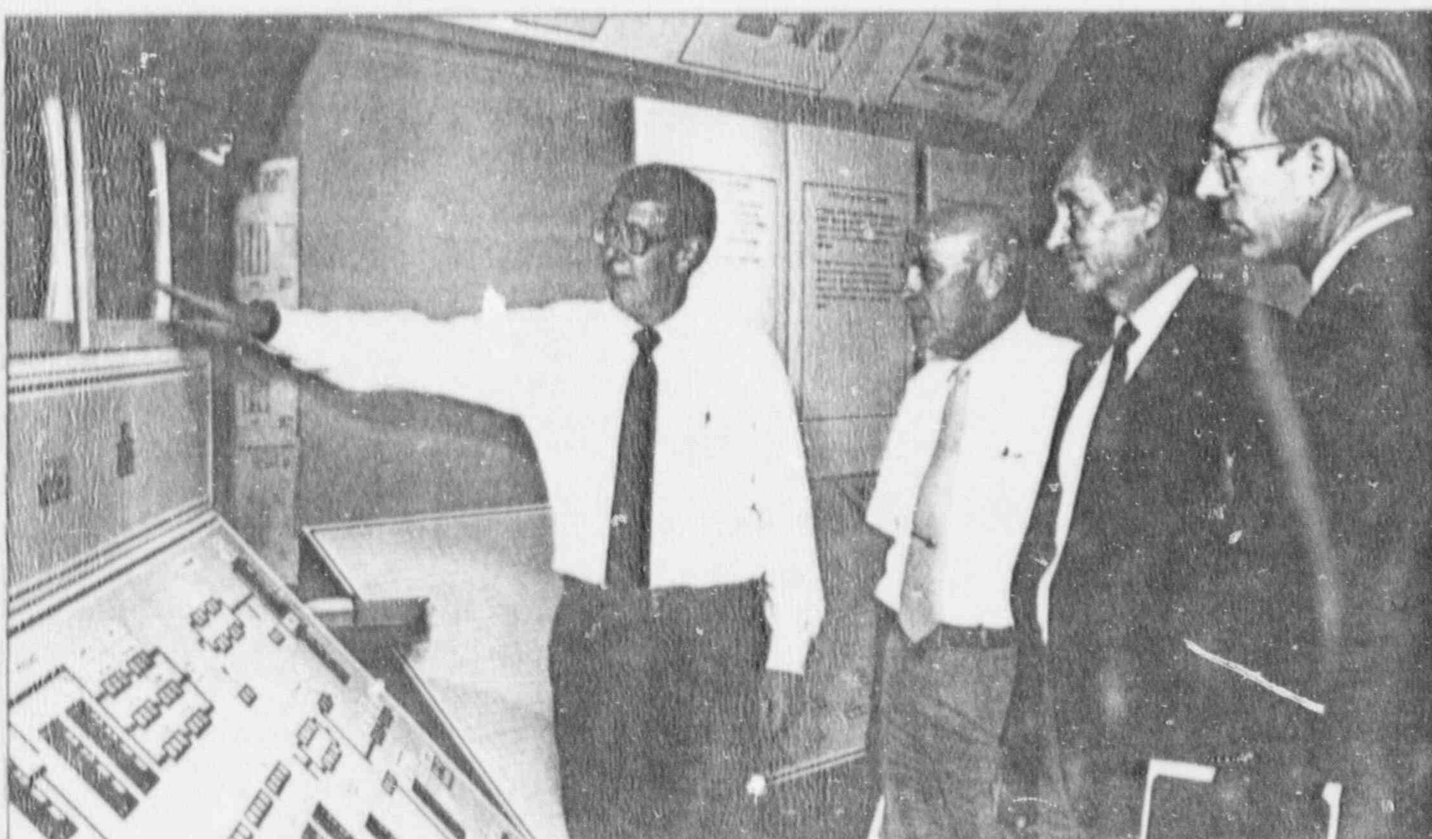
Westinghouse has licensed 12 different PWR designs, representing 173 successful plants.

We successfully manage regulatory issues

Westinghouse has extensive experience in designing and licensing 600 MWe, two-loop plants here and abroad. In all, we have licensed 12 different PWR designs representing 173 plants, including 13 turnkey units. We have contributed to design standardization and streamlined licensing since the early 1970s, when the Westinghouse Reference Safety Analysis Report was introduced to simplify regulatory review of construction permit applications.

Together with NUMARC, the Nuclear Utility Management and Resource Council, Westinghouse successfully led industry efforts to make the regulatory process more effective.

Westinghouse has conducted rigorous safety analyses and probabilistic risk assessments for more than 150 nuclear facilities, and has more than 5,000 man-years of licensing experience before the NRC and its Advisory Committee on Reactor Safeguards. We actively support and implement the requirements of the various codes and standards that apply to nuclear power plant design and construction, including 10CFR50, NQA-1, ANSI and IEEE Standards, NRC Regulatory Guides and the ASME Code.



AP600 program director Howard Bruschi (left) points out the latest Westinghouse nuclear plant control room advantages to (from left) Bill Kinkel, Karl Stahlkopf, and Jack DeVine, members of the EPRI Utility Steering Committee which has performed in-depth reviews and evaluations of the AP600.

Westinghouse uses the latest computer-aided techniques in every facet of AP600 design. Its conceptual design is 100 percent complete. Detailed design and NRC certification will be complete prior to construction.

Westinghouse pioneered nuclear power

Westinghouse designed the first commercial nuclear power plant more than 30 years ago and has been the leader in continued development ever since. We design, license, service, maintain, and inspect reactors; train operators; make fuel; and safely manage wastes. More than half the world's nuclear electricity is produced by reactors built to the Westinghouse design.

Westinghouse has also made a tradition of fulfilling the government's objectives for energy and defense for more than four decades. Our accomplishments range from the first naval nuclear propulsion system to the nation's premier demonstration of safe nuclear waste management at one of the seven complex government facilities we manage and operate for the Department of Energy.

Our team unites the most expert nuclear designers

Our expert AP600 Project Team reflects this tradition and experience. It includes the most qualified nuclear designers in the world, from our Energy Systems Business Unit and other supporting Westinghouse nuclear and research divisions. They have worked together on AP600 conceptual design since its inception, exemplifying the right mix of technical and leadership skills. Our team is further strengthened by experts in architect-engineering, modular prefabrication, field construction, and nuclear component fabrication.

The Nuclear Fuel Division, which is responsible for AP600 core design, recently won the government's first Malcolm Baldrige National Quality Award for excellence as the best manufacturing facility in America. Westinghouse will apply this same dedication to Total Quality throughout AP600 design and construction.

Utility feedback optimizes AP600 design

AP600 is designed to achieve the best safety and performance record. A Westinghouse Owners Group, representing utilities who operate our PWR plants around the world, has collected data on operations, maintenance, and performance for the past decade. We reviewed these data in detail, with a total commitment to optimize AP600 component and system reliability, simplify and improve maintenance, and significantly extend plant life.

AP600 also incorporates feedback from our worldwide network of expert field engineers and service personnel who assist domestic and international utility customers and rapidly respond to their needs.

Westinghouse personnel strengthen the AP600 Project Team with more than 30 years of preeminent nuclear design expertise.

To achieve U.ᶜ energy independence through a nuclear powe, renaissance

The Westinghouse Energy Center, with more than 2100 employees, is the focus of our domestic and international nuclear power business.

Westinghouse issues [illegible] through [illegible] communications.

Westinghouse assigns top priority to bring AP600 on line by 1999

Westinghouse is dedicated to secure America's strength, environmental well-being, and future energy independence through a mix of resources that includes nuclear power and AP60O, as a vital ingredient. Full Corporate support is committed to bring this power plant on line by 1999. As the 27th largest company in the nation with more than $12 billion in sales, Westinghouse has the human resources, financial strength, and technical capability to make AP600 a reality.

Continued cooperation is needed for nuclear renaissance

To stimulate a nuclear power renaissance, government, industry, and utilities must share in development of a new plant. EPRI is planning to continue their support of the AP600 with staff and funding for detailed design and certification. Westinghouse has invested substantial amounts of engineering, testing, and computer analyses on advanced reactor plant designs including the AP600. We will continue to agressively pursue design details of the AP600 and advance its development and acceptance.

ʼWe build public acceptance for AP600

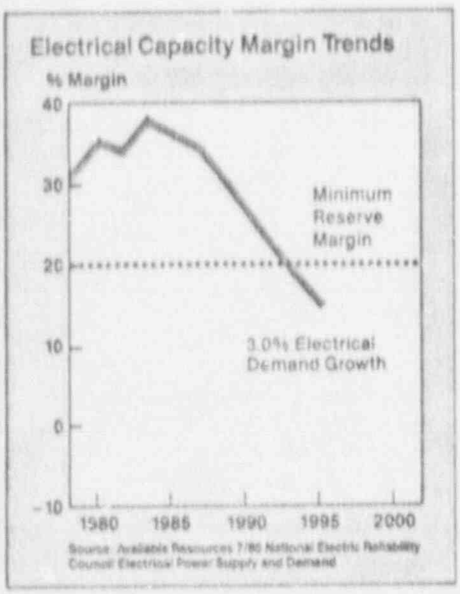
Westinghouse develops effective public information, public acceptance, and community involvement programs to support first-of-a-kind nuclear energy projects. We have described the AP600 in many forums to raise public awareness and get reaction and fee.:back. We communicate and interact with opinion leaders, citizens, and the news media; respond frankly to sensitive issues; and then sustain communications through meetings, newsletters, technical papers, and speakers' bureaus. These activities will support accep.ance of the AP600.

Westinghouse Corporate leadership [illegible] endorses the AP600 program. From left: D[illegible] Stember, Energy Systems Business Unit vice president and general manager; Howard Bruschi, AP600 program director; John Marous, Westinghouse Chairman and Paul Lego, president.

To take charge of our energy destiny

We must begin now to address America's future energy needs

America will suffer unless we begin immediately to develop safe, economic, and environmentally sound energy sources like AP600, to provide essential electricity for industry, business, and home. Since the 1973 oil embargo, U.S. electricity use has doubled. Utilities have not ordered new nuclear plants due, in part, to public uncertainty about safety, economic, and regulatory issues.



These attitudes are changing. President Bush is on record as firmly endorsing the need for clean, safe nuclear power. Even many critics are beginning to refocus attention on nuclear power's environmental benefits.

Nuclear can make greater contributions

Nuclear currently provides 19 percent of U.S electricity–second only to coal. Factors that will likely influence its role in the future include the following:

- Shrinking electric utility reserve margins
- Pressure to shift away from fossil fuels to alleviate damage from acid rain and the climate-altering "greenhouse effect"
- The need to reduce oil imports, which now account for one-third of our trade deficit

As a nation, we are making steady progress toward safe management and disposal of radioactive waste and spent fuel from power generation. Most states have joined together in compacts to develop waste disposal facilities, and a location was recently designated for the national spent fuel repository. Westinghouse will participate in its design.

Recently completed nuclear plants have been delayed for years during construction, while interest on loans mounted, tripling final costs. Licensing reform is now under way to equitably resolve safety and environmental questions without such financial dislocation. One-step licensing will reduce investment risk, by assuring commitment to all approved requirements before construction begins.

Westinghouse prepares for commercialization

Westinghouse has aggressively advanced the AP600 since 1985, completing conceptual design with the full support of the U.S. Department of Energy and EPRI. With continued support, we project a detailed, licensed design by December 1994. With an advanced customer commitment, the first AP600 can be built and producing power by mid-1999.

Westinghouse will ensure that AP600 has a short plant construction schedule and low capital costs by the following:

- Completing 100 percent of design and design certification before construction begins
- Gaining financing, regulatory, utility, and public approvals before groundbreaking
- Including incentives in all contracts to meet cost, schedule, and quality commitments

1990 · 1992 · 1993 · 1994 · 1995 ▸ ▸ ▸ ▸ ▸ ▸ ▸ 1999

Congressional Support and Funding · Utility Customer Commitment · Long Lead Procurement · NRC Design Certification · Site Engineering and Licensing · Site Development · Modular Construction · Plant Operation

That America is on the move again

Now is the time [illegible]
To help ens[illegible] [illegible]ply of energy for America, we must restore confidence in nuclear power. AP600 has great potential to resolve this multidimensional challenge. With government, utility, and public support, this plant can be built, sending a message to the world–that America has not lost its "can do" spirit.

By making the first AP600 a showcase in safety, reliability, and economy, we prove that America is on the move again–in charge of its own energy destiny, and able to offer safe, simple nuclear power for worldwide use.

Westinghouse

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