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

May 24, 2001

MEMORANDUM TO: Dr. Thomas Kress, Chairman
Advanced Reactors Subcommittee

FROM: Michael T. Markley, Senior Staff Engineer

SUBJECT: **STATUS REPORT** FOR THE MEETING OF THE ACRS SUBCOMMITTEE ON ADVANCED REACTORS, JUNE 4-5, 2001, IN ROCKVILLE, MARYLAND

The purpose of this memorandum is to forward additional written materials for your use in preparing for the meeting of the ACRS Subcommittee on Advanced Reactors, June 4-5, 2001. These materials include the revised agenda, status report, and background materials. **Please note that some of these documents are pre-decisional provided for internal ACRS use only and should controlled accordingly.**

Attendance by the following Members is anticipated and reservations have been made at the following hotel for June 3-4, 2001, as indicated:

Apostolakis	Four Points Sheraton	Shack	Four Points Sheraton
Bonaca	Four Points Sheraton	Sieber	Ramada Inn
Ford	Four Points Sheraton	Uhrig	Four Points Sheraton
Kress	Four Points Sheraton	Wallis	Four Points Sheraton
Leitch	Ramada Inn	Garrick (ACNW)	Doubletree
Powers	Ramada Inn		

Please notify Ms. Barbara Jo White at (301) 415-7130 if you need to change or cancel the above reservations

Attachments

1. Subcommittee agenda.
2. Subcommittee status report.
3. ACRS reports dated February 19, 1993; July 20, 1988;June 9, 1987; April 16, 1986; and October 16, 1985.
4. Draft Memorandum dated May 1, 2001, from William D. Travers, EDO, NRC, to The Commissioners, Subject: Staff Readiness for Future Licensing Activities. **(Pre-Decisional Draft)** provided separately
5. Draft Memorandum dated April 25, 2001, from William D. Travers, EDO, NRC, to The Commissioners, Subject: SECY-01-0070 - Plan for Preapplication Activities on the Pebble Bed Modular Reactor (PBMR). **(Pre-Decisional Draft)** provided separately
6. Letter dated May 10, 2001, from James A. Muntz, Exelon Generation Company, to Thomas L. King, Office of Nuclear Regulatory Research, NRC, Subject: Regulatory Issues related to the Pebble Bed Modular Reactor (PBMR).

5/8

7. Memorandum dated February 12, 2001, from Thomas L. King, Office of Nuclear Regulatory Research, NRC, Subject: Meeting with Exelon Generation Company and Other Interested Stakeholders Regarding the Pebble Bed Modular Reactor. (Publicly Available)
8. Handouts from May 7, 2001 meeting, concerning International Reactor Innovative and Secure (IRIS), by M.D. Carelli, Westinghouse Electric Corporation.
9. Handouts from March 2001 meeting, on Gas Turbine-Modular Helium Reactor (GT-MHR): Commercialization Program Briefing, by General Atomics.
10. Handouts from International Symposium on the Role of Nuclear Energy in a Sustainable Environment, presentation entitled, "The GenIV Nuclear Energy System Program: Expectations and Challenges," by Professor Neil E. Todreas, April 20, 2001.
11. Letter dated January 12, 2001, from William D. Travers, EDO, NRC, to James A. Muntz, Exelon Generation Company, Subject: Response to Letter dated December 5, 2000. (Publicly Available)
12. Letter dated December 5, 2000, from James A. Muntz, Exelon Generation Company, to NRC Document Control Desk, Subject: Pebble Bed Modular Reactor Review Requirements.
13 Memorandum dated May 17, 1994, from James M. Taylor, EDO, NRC, Subject SECY-94-133 - Updated Commission Policy Statement on Advanced Reactors to Reference the Commission's Metrication Policy.
14. U.S. Nuclear Regulatory Commission, NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants.

cc:	ACRS Members
cc w/o Attach.	J. Larkins
	ACRS Staff and Fellows

ACRS WORKSHOP
NRC Auditorium
Two White Flint North
11545 Rockville Pike
Rockville, MD. 20852

REGULATORY CHALLENGES FOR FUTURE NUCLEAR POWER PLANTS

JUNE 4- 5, 2001

FIRST DAY, June 4—9:00 A.M. to 7:00 P.M.

1. Introduction G. Apostolakis and T. Kress — 9:00 a.m. - 9:15 a.m.

2. Keynote Address by Commissioner Diaz — 9:15 a.m. -10:00 a.m.

BREAK --- 10 00 a m - 10:15 a m

3. DOE Presentations

Overview and Introduction to Generation IV Initiative--- W. Magwood (DOE)	10:15 a.m.-10:40 a.m.
Generation IV Goals and Roadmap Effort--- R. Versluis (DOE)	10:40 a.m.-11:00 a.m.
Near-Term Deployment Efforts---T. Miller (DOE)	11:00 a.m.-11:25 a.m.
Generation IV Concepts--- R. Versluis (DOE)	11:25 a.m.-11:40 a.m.
Next Steps Generation III+/IV--- S. Johnson (DOE)	11:40 a.m.-12.00 p.m.

LUNCH — 12 00 p m - 1:00 p m

4. Generation IV Design Concepts

Pebble Bed Modular Reactor --- W. Sproat (Exelon)	1:00 p.m. - 1:45 p.m.
International Reactor Innovative and Secure --- M. Carelli (Westinghouse)	1:45 p.m. - 2:30 p.m.
General Atomic- Gas Turbine / Modular Helium Reactor --- L. Parme (General Atomics)	2:30 p.m. - 3:15p.m.

BREAK --- 3:15p.m - 3.30 p m

General Electric-Advanced Liquid Metal Reactor and ESBWR designs —A. Roa (General Electric) 3:30 p.m. - 4:15 p.m.

5. NRC Presentations

NRC Response to 2/13/2001 SRM on Evaluation of NRC Licensing Infrastructure (NRR/RES/NMSS) ---M. Gamberoni (NRC-NRR) 4:15p.m. - 5.15p.m.

Planned RES Activities--- A. Thadani (NRC-RES) 5:15 p.m. - 6:00 p.m.

6. Panel Discussion on Industry and NRC Licensing Infrastructure Needed for Generation IV Reactors 6:00 p.m. -7:00 p.m.

Panelists:
A. Thadani, NRC
S. Johnson, DOE
W. Sproat, Exelon
M. Carelli, Westinghouse
L, Parme, General Atomics
A. Roa, General Electric

End of the First Day

SECOND DAY, June 5 — 8:30 A.M. to 6:45 P.M.

1. Introduction G. Apostolakis and T. Kress 8:30 a.m. - 8:45 a.m.

2. NEI Advanced Reactors Initiatives 8:45 a.m. - 9:30 a.m.
Address by R. Simard, NEI

3. Technical Presentations 9:30 a.m. -4:00 p.m.

Safety Goals for Future Nuclear Power Plants --- N. Todreas, MIT 9:30 a.m.- 10:30 am

BREAK--- 10:30 a m - 10:45 a.m

Licensing by Test --- A. Kadak, MIT 10:45 a.m. - 11:45 a.m.

NERI Project on Risk-Informed Regulation **11:45 a.m. - 12:45 p.m.**
--- G. Davis, Westinghouse and M. Golay, MIT

LUNCH --- 12 45 p.m - 2 00 p m

Advanced Safety Concepts **2:00 p.m. - 3:00 p.m.**
--- C. Forsberg, ORNL

Regulatory Framework for Future Nuclear Power Plants--- A. Heymer, NEI **3:00 p.m. - 4:00 p.m**

BREAK--- 4.00 p m. - 4 15 p m

4. **ACRS and Panel Discussion with Audience Participation — The Most Important Regulatory Challenges for the Licensing of Future Nuclear Power Plants** **4:15 p.m. - 6:30 p.m.**

Panelists:
- **N. Todreas, MIT**
- **R. Barrett, NRR**
- **E. Lyman, NCI**
- **R. Simard, NEI**

5. **Conclusions** **Apostolakis, Kress, et al** **6:30 p.m. - 6:45 p.m.**

End of Workshop

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE SUBCOMMITTEE ON
ADVANCED REACTORS
JUNE 4-5, 2001
ROCKVILLE, MARYLAND

PURPOSE

The purpose of this meeting is to discuss regulatory challenges for future nuclear power plants.

BACKGROUND

The ACRS previously reviewed issues related to advanced reactor designs and provided reports to the Commission dated February 19, 1993; July 20, 1988;June 9, 1987; April 16, 1986; and October 16, 1985 While the technology of advanced reactors and risk analysis have progressed substantially, many of the issues noted in these past ACRS are still applicable to the generation of plants being discussed during this meeting. These issues include: early staff interaction as provided in the 1997 Commission Policy Statement on the Regulation for Advanced Nuclear Power Plants, application of Safety Goals, accident evaluation, source term, site selection, containment, emergency planning, reactivity control, operator staffing and function, residual heat removal, low-power and shutdown operations, fire protection, mitigation systems and classification of structures, systems, and components (SSCs). The question of adequate protection and "How safe should these plants be?" was also discussed.

DISCUSSION

In order to manage time and allow for maximum member/presenter participation and sharing, the Subcommittee/Workshop has been designed with a few protocols. These protocols will also facilitate ample opportunity for public/stakeholder participation. Dr. Kress has agreed and plans to read these guidelines in his opening remarks These guidelines are:

- Presenters should be allowed to make their presentations without substantial interruption ACRS members should hold questions until the end of the individual presentations
- Questions from the audience/stakeholders will be entertained at the end of presentation sessions, not the individual presentations.
- Members of the public/audience should use question cards provided. The ACRS staff facilitator (Mike Markley) will collect comment cards, group like comments as practicable, read them into the record, and refer questions/comments to presenters and/or panel participants, as appropriate.
- It may not be possible to respond to all questions and comments. However, all questions/comments will be listed in the meeting proceedings (NUREG report) following the workshop.

EXPECTED FUTURE ACTIONS

The Subcommittee should identify issues to discuss during the July 12-14, 2000 ACRS meeting.

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

February 19, 1993

The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: ISSUES PERTAINING TO THE ADVANCED REACTOR (PRISM, MHTGR, AND PIUS) AND CANDU 3 DESIGNS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

During the 393rd and 394th meetings of the Advisory Committee on Reactor Safeguards, January 7-8 and February 11-13, 1993, we reviewed a draft Commission paper on the cited subject. Our Subcommittee on Advanced Reactor Designs also met on January 6, 1993, to discuss this matter. We had the benefit of discussions with representatives of the NRC staff, the Department of Energy, and the preapplicants: Atomic Energy of Canada, Limited, Technologies (AECLT), General Electric Nuclear Energy (GE), and General Atomics (GA). We also had the benefit of the referenced documents.

The draft Commission paper lists ten issues that need policy direction from the Commission for proposed deviations from existing regulations. These deviations arise either because existing regulations are generally specific to light water reactors (LWRs), or because the criteria proposed by the designers of the four reactor types listed are significantly different from those in the existing regulations. The draft paper also classified these ten issues into two categories: (1) those issues for which the staff agrees that departures from current regulations should be considered and (2) those issues for which the staff does not believe a departure from current regulations is warranted at this time. Not all of these issues are relevant to each reactor type; the draft paper contains a matrix identifying plant applicability. The paper contains some general comments and recommendations, as well as specific comments and recommendations on each of the ten issues.

Everything we say is predicated on our understanding of the applicable safety policies, which we would describe as follows:

- The safety objective for the nuclear enterprise was described in the 1986 Policy Statement on Safety Goals, and has not been rescinded. There is no distinction drawn in there between existing plants and new plants.

- The ACRS has recommended that the principal use of the goals be to judge the effectiveness of the entire enterprise, including regulation, in producing a plant population consistent with the goals. The Commission has never rejected that view.

- If the industry chooses to do better, we can only applaud its zeal, but ought not to stifle initiative by transforming initiatives into requirements.

Our views on the various items in the referenced draft paper are given below.

GENERAL COMMENTS

1. We find that the identified issues are important and that the staff should receive guidance from the Commission. (There are other policy issues affecting these reactor designs that are being addressed in connection with the evolutionary and passive LWR designs.) There may well be additional policy issues that appear during the preapplication review process. The staff has committed to identify any such issues in subsequent Commission papers.

2. The staff has grouped these ten issues into the two categories described above. We note that all of the affected preapplicants who appeared before us would treat Issue I (Control Room and Remote Shutdown Area Design) as a Category 1 issue, whereas the staff proposes it as a Category 2 issue. We will discuss this difference of opinion below in our opinion on Issue I.

3. For Category 1 issues, the staff proposes more conservative alternatives than the preapplicants propose, in order to account for uncertainties associated with the conceptual design. We are concerned that such an approach might well freeze an unnecessarily large degree of conservatism into the designs, and the preapplicants would have great difficulty persuading the staff to relax this conservatism on the basis of more precise information available in the final design.

4. We support the staff recommendation that "a prototype CANDU 3 is not required for design certification."

5. We support the staff intention to notify the Commission if its position on any of these ten issues should change, or if new issues are identified.

6. We have no objection to the staff recommendation that the highest priority be given to issues that are applicable to the PRISM design.

7. We understand and sympathize with the staff recommendation to defer decisions on generic rulemaking on these ten issues. Nevertheless, we urge the Commission to address these decisions in the near future. (The generic rulemaking question may arise in connection with passive LWR designs.)

8. In several places in the draft Commission paper, there occurs qualitative language, e.g., "appropriate conservatisms" or "credible severe accidents." This language must ultimately be translated into quantitative guidance. We believe that the quantitative guidance is, to a large measure, policymaking, and should not be relegated to low-level reviewers.

SPECIFIC COMMENTS

Category 1 Issues

A. Accident Evaluation

The staff proposal to develop a single approach with certain specified characteristics appears reasonable. We would like to review that approach when it is ready. We believe, however, that the staff should identify at an early stage quantitative guidelines and criteria for accident selection and evaluation. We note that AECLT has taken exception to some of the statements in the draft Commission paper that relate to its approach to this issue. We believe that this disagreement can be resolved by AECLT and the staff.

B. Source Term

The staff proposal to base the source terms on mechanistic analyses appears reasonable, although it is clear that the present data base will need to be expanded. We note that the staff is now developing for LWRs a revision to the TID-14844 source term. It will be appropriate for the staff to consider using the newer approach when it develops source terms, and to take specific account of the unique features of each of the reactor types.

C. Containment

The staff proposal "to postulate a core damage accident as a containment challenge ..." appears reasonable. We would like to review the list of postulated accidents when it is ready.

D. Emergency Planning

The staff proposes that advanced reactor licensees be required to develop offsite emergency plans which will include a requirement for onsite and offsite exercises. This proposal appears reasonable under the present circumstances, except that we would follow existing LWR guidance that permits the omission of offsite exercises when it can be shown that the design would preclude any accidental release exceeding the EPA Protective Action Guides. The staff has agreed to consider, after a review of Accident Evaluation (Issue A, above), whether some relaxation from current requirements may be appropriate. We urge that work on Issue D be closely correlated with work on Issues A and B, in order to avoid unnecessary conservatism.

E. Reactivity Control System

The staff proposal that the absence of control rods need not disqualify a reactor design, provided that an applicant can show a level of safety in reactor control equivalent to that of a traditional rodded system, appears reasonable. We note that this issue is applicable only to the PIUS concept, and that we have not yet had the benefit of presentations by the PIUS designers.

F. Operator Staffing and Function

The staff intends to review the justification for a smaller crew size by evaluating the function and task analyses for normal operation and accident management. This intention appears reasonable, although we believe that particular attention needs to be given to multiple module designs. We note that this issue is related to a similar issue for passive reactors. We believe that the Commission policy should be the same for the advanced reactors and CANDU 3 as it is for the passive reactors.

G. Residual Heat Removal

The staff belief that reliance on a single, completely passive, safety-related residual heat removal (RHR) system may be acceptable appears reasonable, although we would have liked to see the criteria to be used by the staff in deciding acceptability. We agree with the staff that NRC regulatory treatment of non-safety-related backup RHR systems for these reactors should be consistent with design requirements (not yet identified) for passive LWRs.

H. Positive Void Reactivity Coefficient

We agree with the staff that the existence of a positive void reactivity coefficient is a significant concern, but that it should not necessarily disqualify a reactor design. The burden of showing that the consequences of those accidents that would be aggravated by a positive void reactivity coefficient are either acceptable or could be satisfactorily mitigated by other design features surely falls on the preapplicant. On the other hand, the staff should state the criteria it will use to judge "acceptable" or "satisfactorily."

Category 2 Issues

I. Control Room and Remote Shutdown Area Design

We do not agree with the staff decision to treat this issue as a Category 2 issue, and the concomitant recommendation to apply current LWR regulations and guidance until passive LWR policy in this area is finalized. We believe that this issue should be a Category 1 issue, and that the preapplicants should accept the burden of convincing the staff that a proposed design is satisfactory, according to some criteria that should be specified by the staff.

J. Safety Classification of Structures, Systems, and Components

This issue is relevant only to the MHTGR concept. GA makes a persuasive case that the MHTGR is sufficiently different that the LWR criteria for identification of safety-related structures, systems, and components should not arbitrarily be applied to the MHTGR. We concur with this view and believe that Issue J should also be classified as a Category 1 issue. This would not preclude coordination of the policy for passive reactors with the policy for the MHTGR.

Our interest in all these matters continues. We would like an opportunity to review any significant change in staff or preapplicants position, as well as any significant developments in the implementation of the policies.

Dr. Thomas S. Kress did not participate in the Committee's deliberations regarding issues related to the MHTGR.

Sincerely,

Paul Shewmon

Paul Shewmon
Chairman

References:

1. Memorandum dated December 2, 1992, from James M. Taylor, Executive Director for Operations, NRC, for the Commission, transmitting Advance Information Copy of Forthcoming Commission Paper - Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements
2. Letter dated January 28, 1993, from David P. Hoffman, Gas-Cooled Reactor Associates, Management Committee, for D. M. Crutchfield, Office of Nuclear Reactor Regulation, NRC, Subject: Commission Papers on Policy Issues Concerning the Preapplication Reviews of Advanced Reactors
3. Letter dated January 25, 1993, from Peter M. Williams, Department of Energy, to J. Donohew, Office of Nuclear Reactor Regulation, NRC, commenting on the draft Commission Paper
4. Letter dated January 25, 1993, from N. Grossman, Department of Energy, to S. Sands, Office of Nuclear Reactor Regulation, NRC, Subject: Commission Papers on Policy Issues and Schedules Concerning the Preapplication Reviews of Advanced Reactor and CANDU 3 Designs

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

July 20, 1988

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: REPORT ON KEY LICENSING ISSUES ASSOCIATED WITH DOE SPONSORED REACTOR DESIGNS

During the 339th meeting of the Advisory Committee on Reactor Safeguards, July 14-16, 1988, we met with members of the NRC Staff and the Department of Energy (DOE) Staff and reviewed a draft Commission Paper on "Key Licensing Issues Associated with DOE Sponsored Reactor Designs," dated February 9, 1988. This subject was also considered during our 334th, 335th, 336th, and 337th meetings on February 11-13, 1988; March 10-12, 1988; April 7-9, 1988; and May 5-7, 1988, respectively. Our Subcommittee on Advanced Reactor Designs met on January 6, 1988 to discuss this matter. We also had the benefit of the documents referenced to this letter.

The Commission, in a letter dated July 9, 1987, instructed the staff to develop such a key-issues paper in advance of projected safety evaluation reports on each of the three conceptual designs being proposed by DOE and its contractors. The Committee believes this was a wise decision; it is appropriate to confront and attempt to resolve the most important safety and licensing issues in a general and direct way, rather than only by reacting to design proposals. In doing this, the NRC Staff has undertaken an important and difficult task. It can be viewed as an attempt to create, from the top down, a comprehensive rationale for licensing requirements. This would be very different from the existing body of regulations for light water reactors (LWRs), which has grown an element at a time in a more reactive and pragmatic fashion.

The nation has more than thirty years of experience in the development and realization of practical nuclear power. The DOE sponsored designers have made use of this experience and of associated research and analytical development to create three conceptual designs which they believe offer significant advantages over existing LWR plants.

Similarly, the NRC should take advantage of experience in the regulation and safety analysis of plants to create an improved approach to the specification of safety requirements. In doing this, care must be taken that regulatory requirements do not unnecessarily frustrate the development of advanced reactors. The regulations should permit the application of innovative reactor concepts while protecting the health and safety of the public. We believe this can be done, but additional effort on the part of the Commissioners and the NRC Staff will be required. False urgency should be avoided; it is more important to do the job right than to do it soon.

The staff effort so far has been thoughtful and productive, and provides appropriate preliminary guidance. They have identified four key issues as a basis for review of the design proposals:

- Accident selection
- Siting source term selection and use
- Adequacy of containment systems
- Adequacy of off-site emergency planning.

We believe these are important issues, but they do not adequately encompass the full set of concerns. We comment below on these issues and then discuss several additional issues that we believe are also important and deserve further development. We suggest that the staff's key-issues paper be regarded as preliminary guidance and that a continuing program of development and dialogue is necessary before criteria are considered final.

ACCIDENT SELECTION

The staff has proposed four event categories for selection of design basis events based on estimates of the probability of events that might challenge a given system and on past practice and engineering judgment.

For the second of these event categories (EC-II), the staff would require that there be tolerance for single failures, that only safety-grade systems should be credited in meeting the event challenge, and that reactor plant systems should continue to operate normally in response to the challenge. We believe this general approach is sound, but requires two caveats:

° Credit for performance of nonsafety grade equipment in this class of events should be permitted when this can be justified. Designation of a component or system as safety grade is intended to ensure it has certain specific attributes. Among these are the ability to resist certain seismic events, ability to function within certain harsh environments, and a high level of reliability (supposedly guaranteed by a quality assurance program). Not all postulated initiating events are challenges to all of these attributes. Selectivity should be permitted when sufficient information is available about the nature of the design basis event.

° We agree there should not be complete dependence on probabilistic arguments. Although estimates of probability are a proper first-cut approach to the definition of event categories, uncertainty in these estimates is large. Judgments are needed about whether and how to include as design criteria the capability to accommodate phenomena and sequences that are not specifically indicated to be necessary by probabilistic estimates.

CONTAINMENT SYSTEMS

Containment structures clearly are intended to restrict release to the environment of radioactive materials resulting from a severe accident. For LWRs, although the design bases for containments have included a source term related to severe accidents, the design pressures and temperatures have been those related to a large-break LOCA rather than those resulting from an accident involving severe core damage. Whether this seemingly inconsistent but pragmatic approach has served the nuclear power enterprise well can be debated. On the one hand, some of the severe accident issues facing the NRC and the industry today are a legacy of that approach. On the other hand, such a containment performed very well in the TMI-2 accident. Research over the past few years indicates that most existing containments would be reasonably effective in reducing the consequences of severe accidents.

The staff proposal for severe accident and containment requirements for advanced reactors seems to be taking a different, but not necessarily better approach, than that used for LWRs. Their contention is that, if the early lines of defense, namely:

- prevention of challenges to protection systems, and

- prevention of core damage by protection systems are effective enough, then the next two lines of defense, namely:

- a conventional containment structure, and
- an emergency plan for the area around the site,

are not necessary.

The so-called prevention and protection attributes of the three designs being proposed by DOE and its contractors are indeed impressive. The modular high temperature gas cooled reactor (MHTGR) has no conventional containment structure, but relies instead on the capacity of its unique fuel particles to retain fission products, even at abnormally high temperatures, with high reliability. The two liquid metal reactor (LMR) designs have containers around the reactor vessels, but these have low volume and pressure capacity. It is unclear how they would accommodate a challenge greater than minor leakage of sodium coolant.

Accidents can be postulated that would challenge the defense-in-depth concepts being advanced. For the LMRs, a contemporaneous failure of the guard vessel and the reactor vessel, coupled with a sodium fire, would seem to lead to severe consequences. For the MHTGR, a fire in the graphite moderator, perhaps permitted by massive failures of the reactor vessel and core support, might also have severe consequences. Whether these or other accidents could be effectively mitigated by a containment enclosure, or a filtered vent, has not been determined.

We note that in all three designs, absence of containment helps to make feasible one of the major safety advantages, passive systems for removing decay heat. In each case, the reactor vessel surroundings are designed so that air from outside the plant will flow by natural buoyancy through the reactor vessel cavity and thereby remove decay heat. This seems to be a highly effective heat transfer means if the reactor vessel and core are intact. If they are not, this ready supply of oxygen and access to the environment might be a problem. This seems to be a major safety trade-off.

We are not prepared at the present time to accept these approaches to defense in depth as being completely adequate. Further, we are not prepared at this time to accept the arguments that increased prevention of core melt or increased retention capacity of the fuel provide adequate defense in depth to justify the elimination of the need for conventional containment structures. This is not to say that we could not decide otherwise in the future, in response to an unusually persuasive argument.

EMERGENCY PLANNING

We agree with the present approach of the staff's proposal. However, we believe that emergency planning should be reexamined in an effort to describe an approach that would be applicable to all types of reactors.

ADDITIONAL ISSUES

How safe should these plants be?

We believe the debate about how safe is safe enough is concluded. The safety goal policy is in place. That should stand as the definition of how safe these advanced reactors, as well as future LWRs, should be. There are, of course, matters of interpretation and implementation with regard to safety goal policy. These need to be dealt with for all types of reactor plant designs. The focus of licensing and regulation for advanced reactors should be consistent with the safety goal policy; no more, no less, no enhancements, no compromises.

The Advanced Reactor Policy states that advanced reactors must be at least as safe as the current generation of LWRs. The staff interprets this to mean the "evolutionary" generation of LWRs now being reviewed by the NRC for preliminary design certification.

We believe the Advanced Reactor Policy requires no more than, and should require no more than, the level of safety called for in the safety goal policy. Reactor developers, i.e., DOE and the industry, may seek a design that is safer than the safety goal would suggest as necessary, or whose safety is more readily apparent to the public. Those are not unreasonable goals for a developer in seeking public acceptance or more economic operation. However, it seems to us inappropriate for the NRC to ratchet on the standard of safety it has established as necessary and sufficient.

To what extent should regulatory requirements accommodate public perception?

The draft paper states that the staff has incorporated only technical considerations in the development of its proposed positions. In particular, they have not attempted to accommodate external factors, such as public perception. We applaud this restraint. And we counsel the Commission to keep safety regulations unambiguously related to protection of the public health and safety.

Extra capacity in decay heat removal and scram systems

The three DOE designs provide much more capacity in decay heat removal and scram systems than are provided in present LWRs. While these important systems in LWRs must be tolerant of single failures, the advanced reactors go well beyond that. The reason for this is the intent to build more robustness into the first two layers of defense in depth and thus permit less in the last two layers, containment and emergency planning.

Two independent scram systems are provided in two of the three proposed designs. Each system is somewhat diverse in design and tolerant, within itself, of single failure. All three design proposals have multiple systems for decay heat removal. In addition to being diverse and resistant to single failure, the extra systems have inherent passive attributes. They apparently will function effectively without motive power or operator intervention.

However, a caution is necessary. Experience in operation and analysis has indicated that redundancy, i.e., extra systems or components, is not as powerful in improving reliability as might be expected. Too often the nature of initiating challenges, or of the complex sequence of events in accidents, seems to cause the extra parts of a system to be faulted along with the main system. The diverse and passive nature of the three designs being considered might ameliorate such unwanted interdependency, but further study is warranted. In addition, while the three proposed designs have these positive features, it is not clear that the NRC's proposed requirements would provide assurance that these desirable diverse and passive attributes would be guaranteed.

Need for prototyping

The staff proposes only modest requirements for prototype testing of the advanced reactor designs. Although, they have recently added a proposed requirement that any designs not incorporating a containment must be tested in prototype at a remote site, we question whether this is enough to carry the process to a point at which the NRC would be willing to license an unlimited number of new power plants. For example, the metallic LMR cores are claimed to have very favorable, inherently stable characteristics in responding to possible transients. These characteristics were not well understood a decade ago.

An excellent experimental and analytical program by ANL with the EBR-II reactor at INEL has effectively demonstrated that the EBR-II system does exhibit such inherently stable and predictable behavior. However, it is not yet clear that such characteristics can be assured for the larger and different LWRs to be used in commercial electric power production. We believe that a more and extensive series of prototype tests will be necessary before design certification could be granted.

Use of cost-benefit analysis

The staff paper proposes that prospective licensees should be required to demonstrate through cost-benefit analysis that design features alternative to those being proposed are not warranted. Presumably, the NRC staff would review such analyses and perhaps suggest alternatives. We believe this is an unworkable and unnecessary strategy. The NRC should concentrate its efforts on specifying design requirements that will result in plants that are in conformance with the safety goal. Consideration of alternatives and costs is properly a function of the designer and owner of a plant. The NRC should have enough confidence in its safety goal that it does not feel the need for the proposed approach.

Design for resistance to sabotage

It is often stated that significant protection against sabotage can be inexpensively incorporated into a plant if it is done early in the design process. Unfortunately, this has not been done consistently because the NRC has developed no guidance or requirements specific for plant design features, and there seems to have been no systematic attempt by the industry to fill the resulting vacuum. We believe the NRC can and should develop some guidance for designers of advanced reactors. It is probably unwise and counterproductive to specify highly detailed requirements, as those for present physical security systems, but an attempt should be made to develop some general guidance.

Operation and staffing

Little is said in the staff paper about requirements for operation and staffing of advanced reactors. We find this to be a serious oversight. Experience with LWRs has shown that issues of operation and staffing are probably more important in protecting public health and safety than are issues of design and construction. The designers of the three reactor proposals seem to be claiming that the designs are so inherently stable and error-resistant that the questions of operation and staffing, so important for LWRs, are unimportant for the advanced reactors. And that, in fact, the advanced plants can be operated with only a very small staff. We believe these claims are unproven and that more evidence is required before they can be accepted.

The two major accidents that have been experienced in nuclear power, those at TMI-2 and Chernobyl 4, were caused, in large measure, by human error. These were not simple "operator errors" but instead were caused by deliberate, but wrong, actions. There are some indications that the advanced reactor designs being considered have certain characteristics tending to make them less vulnerable to such maloperation. But, this has not been demonstrated in any systematic way. The traditional methods of PRA are not capable of such analyses; but, we believe a systematic evaluation should be made. There seems little merit in making claims for the improved safety of new reactor designs if they have not been evaluated against the actual causes of the most important reactor accidents in our experience.

Will regulatory criteria evolve?

The Staff proposal provides for a future milestone in the ongoing design-review-licensing process at which the NRC will step back and make sure that the agreements reached early in the process are still valid, given possible new information and understandings. We believe this is wise and necessary, although it does place a potential licensee at some risk. It should be recognized that this milestone activity might have to include the possibility of changes in the actual requirements, as well as interpretations of requirements.

Focus on the most important residual uncertainties

Although the staff paper discusses uncertainties relative to the development of requirements and designs, it should provide a clearer statement of what the staff believes to be the most important of these. This would assist policymakers in making judgments about the designs and requirements and, perhaps, about whether certain avenues of research should be further pursued before or in parallel with licensing.

Additional comments by ACRS Member Carlyle Michelson are presented below.

Sincerely,

WKerr

Willliam Kerr
Chairman

Additional Comments by ACRS Member Carlyle Michelson

It is not clear to me that the safety goal in its present form was intended to apply to advanced reactors which do not have conventional containment systems. The guidelines for regulatory implementation might have been different if the Commission had considered that the defense-in-depth approach might not include a containment system on future plants.

It would be unfortunate if the frequency of large release criterion suggested in the present guidelines is used as a basis for justifying the omission of a containment system for an advanced reactor plant at a time when advanced LWRs which might be able to meet the same criterion are required to have containments.

References:

1. Draft Commission Paper from Victor Stello, Jr., for the Commissioners, Subject: Key licensing issues associated with DOE sponsored advanced reactor designs, dated February 9, 1988
2. U.S. Nuclear Regulatory Commission, NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," published June 1988

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

June 9, 1987

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: ACPS COMMENTS ON DRAFT NUREG-1226, "DEVELOPMENT AND UTILIZATION OF THE NRC POLICY STATEMENT ON THE REGULATION OF ADVANCED NUCLEAR POWER PLANTS"

During the 326th meeting of the ACRS, June 4-6, 1987, and in our 325th meeting, May 7-9, 1987, we discussed NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants." A Subcommittee meeting was also held to discuss this NUREG with the NRC Staff on April 24, 1987. During our discussion, we had the benefit of the documents referenced and also of earlier meetings with the NRC Staff. We had previously reviewed the Advanced Reactor Policy Statement and had commented on the statement in a letter to Chairman Palladino dated October 16, 1985.

When the Advanced Reactor Policy Statement was issued, in July 1986, the Commission directed the NRC Staff to prepare a document that would describe its development. Later the purpose of the document (which became NUREG-1226) was extended to include factors important to implementation of the policy. Our comments will be limited to the implementation aspects of the document. We are in general agreement with the implementation approach, but have several comments.

The early interactions between the Staff and an applicant are to be concerned with review of conceptual design, well in advance of any formal application for a construction permit or a design certification. The Staff reported that it intends to assure a conceptual design that looks ahead to possible future standardization. We concur.

The implementation plan encourages, but does not require, the development of new designs based on building and operation of prototypes. We believe that operation of prototypes prior to certification of designs should be the norm and the only exceptions should be made in carefully evaluated cases, where there exists a sufficiently well-developed experience base.

NUREG-1226 uses the terms "defense-in-depth" and "design-basis accident." These are time-honored terms, but they are inexact as concepts. For example, there is a requirement to consider "beyond design basis" scenarios in the design. This presents, at minimum, a serious semantic problem. We believe the Staff needs to clarify its use of these terms.

The policy statement encourages use of "performance-based" rather than "prescriptive" requirements. Again we have concerns that these terms are used without being well defined. For example, 10 CFR 50.46 is certainly a performance-based requirement for the design of an Emergency Core Cooling System (ECCS), but prescriptions for analyzing performance are given in excruciating detail in Appendix K. We believe there is a need to clarify both of these terms and concepts.

We believe the attribute "simplicity" is not always a virtue to be encouraged in future nuclear power plants. From the perspective of safety it is important to have plant systems designed to be easy to operate, easy to maintain, easy to understand, and capable of accommodating a broad spectrum of challenges. However, simplicity does not always provide these characteristics. As an example, increased automation, as a means to make a plant easier to operate, may actually make the design more complex. The history of the evolution of engineered systems indicates they often become more complex as they are improved in reliability and performance, including safety performance.

We believe that NUREG-1226 should provide more definitive guidance for sabotage-protection considerations for advanced plant designs. We recognize this as a difficult issue, and it is for this reason that the Staff should give it additional attention.

Additional remarks by ACRS Member David Okrent are presented below.

Sincerely,

WKerr

William Kerr
Chairman

Additional Remarks by ACRS Member David Okrent

I believe that defense-in-depth should be maintained such that an appropriate containment or other system intended to mitigate severe core melt accidents will be provided.

References:

1. U.S. Nuclear Regulatory Commission, NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," draft published May 5, 1987.
2. U.S. Nuclear Regulatory Commission, SECY-85-279, Subject: "Revised Advanced Reactor Policy Statement," dated August 21, 1985.
3. U.S. Nuclear Regulatory Commission, "Regulation of Advanced Nuclear Power Plants, Statement of Policy," 51 FR 24643, dated July 8, 1986.

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

April 16, 1986

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS COMMENTS REGARDING NRC REVIEW OF ADVANCED REACTOR DESIGNS

During its 311th and 312th meetings, March 13-15 and April 10-12, 1986, the Advisory Committee on Reactor Safeguards heard presentations by the NRR Staff, DOE personnel, and DOE industrial subcontractors on one advanced gas-cooled reactor (GCR) design and two advanced liquid-metal reactor (LMR) designs. These designs are in their early stages, and a unique feature of the design efforts is that NRR personnel have provided safety input very early in the conceptual design stage. This approach, which is in accord with the NRC Advanced Reactor Policy Statement, contrasts with that followed in the design of most of the current generation light water reactors (LWRs) wherein a finalized design was presented to NRC for review and approval (or disapproval). The ACRS believes that significant safety benefits can result from an early interaction between the NRC and the designers and that NRC can have a fundamental influence on the safety aspects of a design if its input is provided at an early stage when design changes can be made both easily and without substantive cost. This contrasts with the situation wherein a finished design is presented to NRC and the latter has considerable difficulty influencing the safety design of the reactor other than through "patches" or "add ons," as some have described the process. The ACRS has recommended the early-interaction approach in the past, and we continue to support it strongly.

These design efforts are directed toward achieving high levels of safety as well as toward achieving low costs and improved operating features. They are thus aimed toward implementing the policy of the Congress as expressed in the Atomic Energy Act. Many innovative features are evolving. For example:

1. LMR designs are being developed which the designers believe would tolerate, without core melt or significant radiation release, very severe accidents such as loss of flow without scram, power excursion without scram (both commonly referred to as ATWS for LWRs), and loss of heat sink without scram. These designs are being influenced by tests run during the past months on EBR-II in Idaho, which have proved that some LMRs can indeed tolerate such severe accidents without public health effects.

2. The designers believe that the need for emergency evacuation planning for the surrounding population can be totally or almost totally eliminated.

3. The reactors which are evolving are small, modular units that would be built in a central factory and shipped by truck, rail, or barge to a site. With factory fabrication, it should be possible to eliminate most of the QA/QC problems which have harassed the current LWRs. With small units, the capital costs per unit should be low, a feature attractive to prospective purchasers.

4. Designs may evolve for which no operator actions would be required in the case of some severe accidents, fires, or types of sabotage for at least several hours.

These and many more innovative features are evolving. However, in order to optimize a design, it may not be necessary to incorporate safety features which would be required in a current LWR. The designers believe that they cannot be innovative in selected areas only; they believe they must be innovative across the board if they are to succeed.

We have been told by NRR Staff that their budget is being reduced drastically and that it may be necessary to terminate the early interactions with DOE. We are also told by DOE that it will be a great loss if this interaction ceases, that DOE and its subcontractors will be unable to proceed effectively without NRC safety input and regulatory guidance. Further, DOE will probably need to share costs with industry, and the latter may be more inclined to provide financial support if DOE can make some sort of statement that NRC considers the designs to be licensable.

We believe that it would be very shortsighted for NRC to terminate this effort for budgetary reasons. We realize that the agency has severe financial problems, but the total amount of resources involved here is very small, and we strongly urge a continuation of this modest effort. If DOE proceeds without NRC input, the NRC may have missed a golden opportunity to influence reactor safety. If DOE stops, the NRC may bear part of the responsibility for failure of the Congressional policy.

Although the comments above have been based on GCR and LMR activities which have been before us recently, the underlying considerations pertain fully as much and perhaps even more to advanced LWRs now being developed and designed by various U.S. organizations.

Sincerely,

David A. Ward

David A. Ward
Chairman

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

October 16, 1985

Honorable Nunzio J. Palladino
Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS COMMENTS ON THE PROPOSED NRC ADVANCED REACTOR POLICY STATEMENT

During its 306th meeting, October 10-12, 1985, the Advisory Committee on Reactor Safeguards reviewed the proposed Statement on the Revised Regulatory Policy for Advanced Reactors as presented in SECY-85-279 dated August 21, 1985. This matter was also discussed during a meeting of the ACRS Subcommittee on Advanced Reactors on September 25, 1985. We also had the benefit of the documents referenced.

In our view the intent of the proposed statement is fully appropriate, including the specific provisions for establishing an early and continuing interaction between the NRC and the designers and others engaged in developing proposals for advanced reactors. We also welcome the intention to stabilize, expedite, and clarify, to the extent possible, the regulatory review process for advanced designs, and particularly the effort to make the findings and decisions readily understandable by all those concerned -- including the general public.

In the "Commission Policy" paragraph of the proposed text (and also in the "Summary"), it is stated that, "The Commission intends to require the same degree of protection . . . as is required for current generation LWRs." It might be better to say "The Commission intends to require at least the same degree of protection . . .," or, alternatively, that "The Commission does not intend to relax in any way the degree of protection required . . . ." It is, of course, true for a number of reasons -- including the advances in technology since the basic present LWR designs were laid down, the contribution of past experience to new designs, and the inherent features of some proposed advanced designs -- that a greater margin of safety or a greater assurance of safety can be expected to be realized. Beyond the general intention to make as full use of these as may be feasible, we however, do not consider it useful or possible in any clearly implementable way to include such an expectation as a requirement.

As an additional general comment, in the section "Proposed Policy," there is a list of eleven attributes, some or all of which, it would seem, must be incorporated to some degree for a reactor design to qualify as "advanced." This list is a rather mixed bag, and some of the stated attributes would appear to be inconsistent with some others. For example, "Simplified safety systems which require . . . the least equipment" may be difficult to reconcile with "sufficient . . . redundancy, diversity," etc. It would seem preferable to have this (or some such) list identified as possibly desirable attributes which could assist in establishing the acceptability or licensability of a proposed design, but not as a set of criteria for the decision of whether or not a reactor is "advanced." Indeed, some of the attributes -- such as reducing radiation exposure to plant personnel, or considering defense-in-depth philosophy -- would scarcely appear to be hallmarks for being advanced. The statement of the particular (desirable) attribute of "providing sufficient inherent safety," tied in, as it is, with the possibly incompatible requirements for redundancy, diversity, etc. might benefit from rewording or rethinking.

Along with these general comments, there are a number of minor items in the proposed text of the statement which we would suggest should be modified or given some further consideration:

- The second sentence of the paragraph "Purpose" would appear to define what is meant by an "advanced reactor." This might better end after, ". . . now under construction or in operation." Though the properties identified in the balance of the sentence -- "providing more margin . . ." -- or "making more use . . ." may indeed be desirable, to include these here makes it less clear than it ought to be whether a particular design is to be considered "advanced," and tends to conflict with the more basic intention to require (at least) the same degree of protection. [The same sentence also appears in the "Summary."]

- In the second paragraph of "Previous Experience" it is stated that the FFTF was "reviewed but not licensed." It might be worthwhile to make it clear that the reason it was not licensed had to do with its not being in the realm requiring a license, rather than because it was not licensable.

- In the last sentence of this same paragraph it seems odd to adduce the previous experience with LWRs.

- In the paragraph following the list of attributes it is said that the number of regulatory requirements would be based on the extent to which a design incorporates the suggested attributes. This sentence might be a candidate for deletion.

- In the next following paragraph it is interesting to learn, as part of the Commission policy on advanced reactors, that early interaction may be accomplished either by means of meetings or in writing. This is another highly deletable sentence.

- In the final sentence of the antepenultimate paragraph it is pointed out that if there should be too many requests for interaction, the Commission may have to limit the number. This might become a fact of life, but is it really a proper part of a Commission Policy Statement?

- Finally, at the end of the third paragraph of the "Summary," the Commission is urged to undertake to keep the public informed of its judgment on all the "known and unknown" aspects . . . . This would seem to be ambitious beyond reason.

Sincerely,

David A. Ward

David A. Ward
Chairman

References:

1. SECY-84-453A, Subject: Regulatory Policy for Advanced Reactors, dated February 26, 1985
2. Letter from R. F. Fraley, Executive Director, Advisory Committee on Reactor Safeguards, to J. E. Zerbe, Director, Office of Policy Evaluation, NRC, Subject: Proposed Regulatory Policy for Advanced Reactors, dated April 15, 1985
3. Memorandum from J. E. Zerbe, Director, Office of Policy Evaluation, NRC, for NRC Commission, Subject: Information Paper - Summary of Comments on Advanced Reactors Policy Statement, dated July 3, 1985

ABSTRACT

On March 26, 1985, the U.S. Nuclear Regulatory Commission issued for public comment a "Proposed Policy for Regulation of Advanced Nuclear Power Plants" (50 FR 11884). This report presents and discusses the Commission's final version of that policy as titled and published on July 8, 1986 "Regulation of Advanced Nuclear Power Plants, Statement of Policy" (51 FR 24643). It provides an overview of comments received from the public, of the significant changes from the proposed Policy Statement to the final Policy Statement, and of the Commission's response to six questions contained in the proposed Policy Statement The report also discusses the definition for advanced reactors, the establishment of an Advanced Reactors Group, the staff review approach and information needs, and the utilization of the Policy Statement in relation to other NRC programs, including the policies for safety goals, severe accidents and standardization. In addition, guidance for advanced reactors with respect to operating experience, technology development, foreign information and data, and prototype testing is provided. Finally, a discussion on the use of less prescriptive and nonprescriptive design criteria for advanced reactors, which the Policy Statement encourages, is presented.

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EXECUTIVE SUMMARY

Advanced reactors have a long regulatory history, but until recently there has been essentially no explicit policy for their regulation other than case-by-case reviews which included determinations about their licensing requirements, including the extent of their conformance with Light Water Reactor (LWR) criteria. Accordingly the Commission has developed a Statement of Policy for Regulation of Advanced Nuclear Power Plants (Final Statement), published on July 8, 1986 (51 FR 24643) which encourages early interaction between NRC and advanced reactor designers to establish licensing guidance applicable to these designs. This report serves to document the comments on the proposed policy (published in the Federal Register on March 26, 1985, 50 FR 11884), to describe the significant changes made to the policy from that proposed to the final version and to provide guidance about implementation of the final policy, staff information needs and the staff approach to be used in the review of advanced reactor concepts under the Final Policy Statement. It is not the purpose of this document to impose technical design requirements on advanced designs. The staff reviews under the Final Policy Statement would occur before any formal application for authorization of construction or for a standard plant review and certification. However, the review principles and results would be expected to be used in the review of that design after a formal application. The key points contained in this document are summarized below:

(1) The Final Policy Statement is applicable to reactors of innovative design but not to designs for which licensing requirements are essentially covered by the LWR-Standard Review Plan (i.e., evolutions from current generation LWRs). The specific determination of which new designs are considered to fall within the Final Policy Statement will be made case by case. At the present time certain high temperature gas-cooled reactor (HTGR) designs, liquid metal reactor (LMR) designs and innovative LWR designs qualify as advanced reactor designs.

(2) Comments received on the proposed Policy Statement (50 FR 11884) were almost unanimous in the support of its objectives. Most commenters, however, stated that the objectives should not be imposed as requirements.

(3) The Policy Statement established a charter for an Advanced Reactors Group (ARG). The ARG function is in the Office of Nuclear Regulatory Research and is located in the Advanced Reactors and Generic Issues Branch, Division of Regulatory Applications. The ARG serves as a project manager coordinating and scheduling activities both within and outside the NRC, as well as performing a significant portion of the technical review itself. In performing this review, use will be made of the existing licensing guidance for LWRs, where practical, and supplemented, as necessary, with additional criteria to address the unique characteristics of the advanced designs.

(4) While the Final Policy Statement encourages innovative reactor designs and safety criteria, the review of advanced reactor designs will still require satisfactory consideration of the Commission's regulations, regulatory guides and other guidelines, such established and developing criteria as the defense-in-depth philosophy, standardization, the Commission's safety goal and severe accident policies, and applicable industry codes and standards.

(5) The Commission and staff expect the licenseability of advanced reactor designs to be supported by technology through a suitable combination of operating experience, the existing technology base, planned technology development, probabilistic risk assessment, applicable information and data from foreign countries, and plant testing. Prototype testing is encouraged.

(6) The use of less prescriptive, nonprescriptive, or performance related licensing criteria will be considered. Designers are encouraged to propose those criteria they believe are applicable to their designs and to address how such criteria will enhance safety and what changes or benefits in the traditional NRC process of regulation are expected from the use of such criteria.

(7) Requests by advanced reactor designers for reviews of advanced reactor conceptual designs should be addressed to:

Director, Office of Nuclear Regulatory Research
USNRC
Washington, DC 20555

1 INTRODUCTION

On May 1, 1986 the NRC approved the issuance of a document entitled, "Regulation of Advanced Nuclear Power Plants; Statement of Policy." This Policy Statement was published in the Federal Register on July 8, 1986 [51 FR 24643] and forms the overall guidance for the NRC's activities regarding advanced nuclear power plants. The Policy Statement is provided in the Appendix to this document.

The Policy Statement calls for early interaction between the NRC staff and advanced reactor designers; encourages greater safety margins through the use of inherent, passive, or other innovative means for safety design; and establishes an Advanced Reactors Group (ARG) as a focal point for its implementation. The Policy Statement originally established the ARG within the Office of Nuclear Reactor Regulation (NRR), but a subsequent NRC reorganization approved by the Commission on February 11, 1987 transferred the ARG function to the Office of Nuclear Regulatory Research (RES).

The final Policy Statement is based on the development and revision of a proposed Policy Statement, published for comment on March 26, 1985 (50 FR 11884), including assessment of public comments.

The stated primary objectives of the Policy Statement are:

(1) "Encourage earliest possible interactions of applicant, vendors, and government agencies, with the NRC;

(2) Provide all interested parties, including the public, with the Commission's views concerning the desired characteristics of advanced reactor designs; and

(3) Express the Commission's intent to issue timely comment on the implications of such designs for safety and the regulatory process."

The purpose of this document is to (1) summarize the public comments received on the proposed version of the Policy Statement, (2) identify the significant changes made in the Policy Statement from the proposed version to the final version and (3) identify the responsibilities, interfaces and other considerations which must be addressed in the implementation and utilization of the final Policy Statement.

2 HISTORY AND BACKGROUND

The NRC and the Atomic Energy Commission before it, together with the Advisory Committee on Reactor Safeguards (ACRS), have a long history of review and evaluation of advanced reactors. Safety reviews for construction and operation of liquid metal-cooled, gas-cooled, and other types of non-water-cooled power reactors performed in the 1950s and early 1960s were similar to those performed for the early commercial Light Water Reactors (LWRs). The reviews performed by the regulatory staff and the ACRS were highly customized and were generally based on the engineering experience and judgment of participating individuals. The regulatory staff and ACRS members worked closely together in the review and assessment of information supplied by the designers, owners and constructors without the availability of the regulatory guidance and structure established later during the course of LWR commercial development. In more recent advanced reactor reviews, explicit use was made of LWR regulatory guidance where applicable, a practice that continues.

The Advanced Reactor Policy Statement identifies previous experience with the regulation of high temperature gas cooled reactors (HTGRs) and liquid metal reactors (LMRs). Construction permits and operating licenses were granted to the helium cooled Peach Bottom-1 and Fort St. Vrain reactors and to the sodium cooled Fermi-1 and the Southwest Experimental Fast Oxide Reactor (SEFOR) reactors. The design of the Department of Energy's (DOE's) Fast Flux Test Facility (FFTF) was given a safety review by the NRC but a license was not required by law. Reviews were also performed on reactor designs that were not subsequently built. For gas cooled reactors these were the Summit and Fulton applications for large HTGRs, the General Atomic Company's standard large HTGR plant (GASSAR), and a conceptual design for a gas-cooled fast breeder reactor (GCFR). With regard to LMRs, the Clinch River Breeder Reactor (CRBR) was reviewed, and a public hearing held, but the project was terminated by Congress in 1983 before a construction permit was issued and general construction began. It should be noted that since the CRBR was to be a power reactor prototype, it was subject to the same regulatory process as any current commercial nuclear power project.

In addition to the background of individual licensing actions, the Nonproliferation Alternative Systems Assessment Program (NASAP) of 1979 provided both a broad policy study and a review of specific safety concepts on reactor regulation. In the NASAP studies the NRC considered the safety and licensability of a variety of advanced reactor concepts ranging from preliminary conceptual designs to variations on existing LWRs.

Table 2.1, "Advanced Reactor Regulatory Experience" provides in summary format further information on previous advanced reactor safety reviews in the United States.

Until the present Policy Statement, the principal statement on advanced reactor review policy was given in the introduction to Part 50 of Title 10 of the Code of Federal Regulations: Appendix A, "General Design Criteria for Nuclear Power Plants." Specifically, this introduction states:

"These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units."

This led to the "comparable level of a safety" philosophy under which HTGRs and LMRs were reviewed for many years; that is, a comparable level of safety would be established for all reactor types, with the recognition that the licensing criteria for advanced reactors could be developed using those for light water reactors to the extent practicable. The implementation of this philosophy took three forms with respect to the existing criteria; direct adoption, suitable adaptation, and recognition of the need for and development of specialized criteria. Direct adoption of the existing criteria was possible in many instances and provided a ready means of ensuring a comparable level of safety.

Examples of direct adoption are numerous and include industry standards for electrical and mechanical equipment and many of the NRC regulatory guides.

For those existing criteria that could not be regarded as unequivocably applicable, suitable adaptations were developed to permit the use of the phrase, "meets the objectives of" or words to this effect. Development of such adaptations was usually a straightforward practice of the applicant identifying and justifying discrepancies from the criteria followed by a staff review of the applicant's approach. An early example of the adaptive approach was the means for conformance of the Fort St. Vrain design to the Commission's General Design Criteria for LWRs.

For those portions of advanced reactor designs that were uniquely different from those of LWR designs (e.g., requirements for handling a sodium coolant or the use of a concrete reactor vessel for HTGRs), adoption or adaptation of existing regulations or standards was not possible or desirable. Such criteria needs were satisfied by engineering judgment and analysis resulting in the development of specialized licensing criteria.

Although the above developments have taken place in the advanced reactor area, they only provide a general background for the scope and intent of the present Advanced Reactor Policy Statement. The first formal development of advanced reactor policy began at a Commission meeting held on November 30, 1983, during which the Commission's responsibilities toward encouraging the development of reactor types of "greater inherent safety" were discussed. NRC's Office of Policy Evaluation (OPE) was asked to prepare an initial draft statement that was to include a discussion of the Commission's role in advanced reactor design in relation to NRC's enabling legislation. This draft was reviewed by the Office of Nuclear Reactor Regulation (NRR) and later discussed with the Commission at a meeting held on February 27, 1984. NRR participated with OPE in the further development of the statement and after substantial Commission and staff review, a statement of "Proposed Policy for the Regulation of Advanced Nuclear Power Plants" was published for comment on March 26, 1985 (50 FR 11884). The proposed Policy Statement included a description of the way the regulation of advanced reactors is guided by the legislative background and noted that the NRC "is precluded from designing, or doing research on, complete new designs for the purpose of establishing or developing their commercial potential." This principle avoids a conflict of interest since the NRC would not be placed "in a position to generate, and then have to defend, basic design data of its own."

A 60-day comment period for the Policy Statement followed its publication and 20 responses were received. These responses are identified and discussed in Section 3, "Abstract of Comments." After consideration of the comments and further review by the Commission and the staff, the final Policy Statement was issued. One of the features of the proposed Policy Statement was the inclusion of six questions on advanced reactor policy. The final Policy Statement restates these questions together with the Commission's own responses. The commenters' responses to the questions are discussed in Subsection 3.5, "Response to Questions." A discussion of the major changes in formulating final Commission advanced reactor policy from that proposed in 1985 is given in Section 4, "Formulation of Final Policy."

Table 2.1 Advanced Reactor Regulatory Experience

Part A - High Temperature Gas-Cooled Reactors
(The General Atomic Company and its successors were responsible for all HTGR designs)

Project Identification	Operational and/or Regulatory Experience	Comments and Remarks
Peach Bottom I - 40MWe, Philadelphia Electric Company, Peach Bottom, Penn.	Construction initiated in 1962. OL granted in 1967. Highly successful operation between 1967 and 1974.	First HTGR in U.S. Demonstrated cermanic (graphite) core design and ceramic fuel. Fuel concept differed from later HTGRs as design provided for fission product release and clean-up. Reactor project terminated for economic reasons.
Fort St. Vrain - 330MWe, Public Service Company of Colorado, Weld County, Colo.	Constructed between 1968 and 1974. OL granted in 1974. Operation sporadic, mainly caused by water ingress from helium circulator bearings.	Provided basis for modern, large HTGR concept through introduction of PCRV, integrated primary coolant system, improved fission product retention in fuel particles through use of silicon carbide layer. Fuel and steam generator performance excellent.
1000 MWe HTGR Study	A 1969 study involving both the staff and ACRS to upgrade HTGR power level. Favorable ACRS letter issued.	LWR type large containment vessel determined to be necessary for an HTGR of this size.
Summit and Fulton Plants, Sited in Delaware and Pennsylvania, but never built, 700-1000 MWe.	Licensing activities 1973 to 1975. Favorable SERs and ACRS letters issued but plants cancelled for economic reasons prior to public hearings and CP issuance.	Design based on 1000 MWe study. Substantial component development program planned.

Table 2.1 (Cont'd)

Part A - High Temperature Gas-Cooled Reactors

Project Identification	Operational and/or Regulatory Experience	Comments and Remarks
Gas Cooled Fast Breeder Reactor - GCFR	Concept reviewed by staff and ACRS between 1971 and 1975. Staff concluded that a demonstration plant, subject to the conditions of its SER, could be built.	Some SER concerns about ECCS were later addressed by use of a natural convection design for decay heat removal when pressurized.
GASSAR - a standard plant review based on Fulton Reactor Design	Staff review initiated 1974, terminated in 1977 with an interim SER.	Detailed review of fission product release from fuel experiments published as NUREG-0111.
Severe Accident Source Term Study - PRA study performed by RES Contractors on 2240MW(t) concept.	Study performed between 1982 and 1984. Inconclusive quantitative results but valuable insights into HTGR severe accidents developed.	Forms a basic starting point for continued HTGR severe accident analysis. Did not consider air and water ingress events.

Table 2.1 (cont'd)

Part B - Liquid Metal Reactors
(Fast Reactors Unless Otherwise Noted)

Project Identification	Operational and/or Regulatory Experience	Comments and Remarks	Designer
EBR-I (Experimental Breeder Reactor) INEL Site, Idaho 1.4 MWt	Plant not reviewed or licensed by NRC. Startup 1951, Shutdown in 1964	NaK cooled, first commercial power generation	Argonne National Laboratory
EBR-II Idaho 62.5 MWt INEL: Site (Experimental Breeder Reactor)	Plant not reviewed or licensed by NRC. Startup 1963, Continues in operation	Has operated successfully for 24 years. Demonstrated inherent safety characteristics of liquid metal reactors and metal fuel	Principal Nuclear Contractor Argonne National Laboratory
SRE Sodium Reactor Experiment Santa Susana, Calif., 20 MWt	Startup 1957, Shutdown 1964	Sodium Graphite Reactor (Thermal Reactor)	Atomics International
Hallam Nuclear Power Facility - Hallam, Nebr. 240 Mwe	Startup in 1962, Shutdown 1964	Sodium Graphite Reactor (Thermal Reactor)	Atomics International
Fermi-I Lagoona Beach, Mich. 200 MWt	Startup 1963, Shutdown 1963	Experienced fuel melting from partial core flow blockage. Returned to service but shutdown for economic reasons.	Power Reactor Development Corp.
SEFOR (Southwest Experimental Fast Oxide Reactor) Strickler, Ark. 20 MWt	Startup in 1969, Shutdown in 1972	Operated successfully until shutdown due to completion of its mission. Demonstrated inherent negative reactivity feedback in oxide fuel.	General Electric
ETEC Facilities - Santa Susana, Calif. (Non-Nuclear)	Sodium equipment test facility	Demonstrated liquid metal component performance.	Atomics International
FFTF (Fast flux Test Facility), Hanford, Wash 400 MWt	Constructed 1971-1980, NRC performed a safety review of the design and issued an SER (NUREG-0365) in 1978.	Plant has operated successfully for 6 years. Has demonstrated oxide fuel system.	Westinghouse
Clinch River Breeder Reactor - Oak Ridge, Tenn. 975 MWt	NRC completed the SER (NUREG-0968) and public hearing for CP in 1983.	Plant never built due to lack of funding. Much R&D done in support of design.	Westinghouse

3 COMMENTS ON PROPOSED POLICY

This section consists of abstracts and discussions of the public comments that were submitted on the Commission's proposed Policy Statement on advanced reactors published on March 26, 1985 (50 FR 11884). The abstracts were prepared from the 20 sets of comments from the organizations listed in Table 3.1. These organizations, which are indicated parenthetically, can be categorized according to the following groups: nuclear utilities (4, 6, 12, 16, 19); nuclear industry, (1, 3, 8, 9, 10, 11, 13, 15, 18, 20); national laboratories (2, 7); academic institution (17); government agency (5); and public interest group (14). The general reactions of the commenters and their responses to the six Commission questions are discussed in the following sections.

The abstracts are intended as accurate as possible representations of the oral and written comments that were received. In the interest of brevity, however, the commenters' reasons for their views are not given in detail; therefore, the abstracts may not be totally accurate. The reader who finds an abstract unclear and wishes to know exactly what the commenter said should consult the original comments; these are available for inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, DC 20555.

3.1 Overall Reaction

The commenters unanimously supported the issuance of a policy statement. All except one (14) endorsed the desirability of simplifying and stabilizing the regulatory process and called for less specificity in the NRC's regulations, although they differed somewhat in the specific details of the process they would endorse. These commenters generally supported NRC's use of top-level public risk objectives, with most explicitly referencing safety goals. Although generally endorsing the Commission's objectives for advanced reactors that were stated in the proposed Policy Statement, all but two (13, 14) felt that they should not be considered as NRC requirements. Most believed that the baseline for acceptability should be the level of safety required of current light water reactors.

Most commenters were supportive of an Advanced Reactors Group and continuing interactions between the industry and the NRC during the development process. However, there were some differences in their views. In addition, there was confusion among the commenters about the type of reactor to which the policy statement was applicable and the extent of the difference from current reactors before a reactor could qualify as an "advanced" reactor; some explicitly suggested that the Commission clarify this point. Summarized below are the comments received on the individual sections of the proposed Policy Statement, including the six questions.

Table 3.1 Table of Commenters

Reference Number	Name	Affiliation
1.	Doan L. Phung	Professional Analysis, Inc.
2.	J. O. Zane	EG&G Idaho, Inc.
3.	John J. Taylor	Electric Power Research Institute
4.	D. W. Edwards	Yankee Atomic Electric Company
5.	James W. Vaughan, Jr.	Department of Energy
6.	H. L. Brey	Public Service Company of Colordao
7.	Herman Postma	Oak Ridge National Laboratory
8.	T. E. Northup	GA Technologies, Inc.
9.	L. D. Mears	Gas-Cooled Reactor Associates
10.	A. E. Scherer	Combustion Engineering, Inc.
11.	R. B. Bradbury	Stone & Webster Engineering Corp.
12.	L. Bernath	San Diego Gas & Electric
13.	John C. Young	International Energy Associates Limited
14.	E. Nemethy	Ecology/Alert
15.	Glenn G. Sherwood	General Electric Company
16.	Hal B. Tucker	Duke Power Company
17.	M. Golay, D. Lanning and L. Lidsky	Department of Nuclear Engineering, Massachusetts Institute of Technology
18.	E. P. Rahe, Jr.	Westinghouse Electric Corporation
19.	J. R. Thorpe	GPU Nuclear
20.	R. P. Schmitz	Bechtel Power Corporation

3.2 Scope

The proposed Policy Statement defined advanced reactors as "reactor designs which are significantly different from the present generation light water reactors." Most commenters (1, 2, 6, 7, 9, 10, 11, 12, 13, 14, 16, 17, 20) either accepted or did not mention the Commission's definition. Some (5, 8) explicitly supported the definition. The Electric Power Research Institute (EPRI) (3) believed that the statement was applicable to both advanced reactor designs based on "evolutionary improvements demonstrated by current light water reactor technology" and to those based on "substantial changes or radical departures from current technologies" and criticized the statement for not defining criteria that distinguished between the two. Similarly, Westinghouse (18) stated that "the policy statement should recognize that future designs do not necessarily require different features to be viable and licensable." Others (4, 15) believed that the scope of the Policy Statement was unclear and needed revision.

3.3 Interaction with NRC

Many commenters (2, 3, 5, 8, 11, 13) supported the earliest possible interaction between the industry and the NRC during the development process, with the NRC Advanced Reactors Group responsible for this interaction. Others (1, 4, 6, 7, 10, 14, 15, 17, 19) did not explicitly discuss this issue. Duke Power Company (16) expressed the opinion that the NRC should be cautious so as not to unduly influence, either positively or negatively, the selection of alternative concepts at the conceptual design stage and should deal with industry in a cooperative but independent manner.

San Diego Gas and Electric (12) was negative in its reaction to the concept of early interaction with the Commission and disclosure of the Commission's safety judgements to the public throughout the process. This commenter stated: "These 'motherhood' statements are antithetical, since premature disclosure of design details, before being fully analyzed and verified, raises expectations, which subsequently may require substantial modification to be viewed by the regulators and the anti-nuclear activists as equivocation. Also, early interaction invites critical assessment before all design features are fully coordinated into a defensible, validated whole. The NRC should take care to minimize opportunities for demagoguery and the fostering of misconceptions."

Gas Cooled Reactor Associates (GCRA) (9) felt that the Policy Statement needed to be revised to "include a statement to the effect that the NRC will actively pursue the development of mechanisms for the timely and effective incorporation of data from other countries into the licensing process." Westinghouse (18) voiced opposition to the aspect of NRC interaction with foreign sources by stating: "We strongly question the USNRC's stated willingness in this policy statement to review designs proposed by foreign vendors. The Atomic Energy Act of 1954, as amended, provides no extraterritorial jurisdiction to the NRC in the review of designs which may neither be manufactured or licensed in the United States. Improper exercise of USNRC jurisdiction could give rise to legal challenges." Westinghouse also felt that technical review responsibilities should rest with the current staff technical organization and not with a new staff group.

3.4 Standardization

Only two comments were received with respect to standardization. The Department of Energy (DOE) (5) stated:

> "The Department considers that it is critically important to improve the efficiency of the nuclear licensing and regulatory process and has had introduced into both Houses of Congress the "Nuclear Facility Standardization Act of 1985" to accomplish that objective. Any policy statement on the regulation of advanced reactors should be supplementary and complementary to that prime objective."

In contrast, the Public Service Company of Colorado (6) stated:

> "As a general comment, PSC supports the Commission's 1985 Policy and Planning Guidance statement that encourages industry to pursue standardization of the current generation of nuclear power reactors. However, the immediate application of this policy to advanced nuclear reactors may be inappropriate, since advanced reactors, by definition, are reactor designs which are significantly different from the present generation of light water reactors and the various advanced reactor concepts ordinarily differ in many ways from one another. Until a particular advanced reactor develops into a proven design that is capable of giving rise to a new family of nuclear power plants, it would be premature to think in terms of standardization for such units."

3.5 Responses to Questions

Question 1 - Regulatory Approach

> "Should NRC's regulatory approach be revised to reduce dependance on prescriptive regulations and instead establish less prescriptive design objectives, such as performance standards? If so, in what aspects of nuclear power plant design (for example, reactor core power density, reactor core heat removal, containment, and siting) might the performance standards approach be applied most effectively? How could implementation of these performance standards be verified?"

All commenters agreed that a less prescriptive approach to regulation (than the current one) is desirable, with the exception of the commenter from Ecology Alert (14), who did not address the issue. Almost all of these expressed the view that advanced reactors should be subject to top-level risk objectives or safety goals concerned with public health and safety and that any subsidiary performance standards should be closely related to showing compliance with these goals: in other words, they did not want regulation to otherwise restrict the design of advanced reactors. Most commenters felt that any design objectives should be broad enough to permit or encourage innovation. EPRI (3) differentiated between designs evolving from current reactors, which it feels should be regulated under an improved version of the current process, and reactors based on radical design approaches, for which it deems performance standards practical. DOE (5) emphasized the importance of a predictable, well-defined licensing process which identified information required and methodology used by NRC to judge compliance with the top-level criteria. Duke Power Company (16) contended that use of performance standards rather than design-oriented regulations is not enough to avoid prescriptive regulation. It also argued that the management structures of NRC and industry, and the interactions between them, must be changed. Oak Ridge National Laboratory (ORNL) (7) suggested establishing performance standards for essentially all aspects of the nuclear steam supply system and all systems which determine the safety of the public. Several commenters (5, 7, 18) stated that, to the extent that more detailed standards are needed, general NRC regulations should be supplemented as necessary by industry standards and codes. Several commenters (4, 7, 10, 15) believed that standardization will reduce the need for prescriptive regulation. Several others (2, 7, 9, 13, 15, 16) discussed the need for standards which permit simple verification and give designers considerable latitude and responsibility for demonstrating compliance.

Question 2 - Inherent Safety

"Should the regulations for advanced reactors require more inherent safety margin in their design? If so, should the emphasis be on providing features that permit more time for operator response to off-normal conditions, or should the emphasis be on providing systems that are capable of functioning under conditions that exceed the design basis."

Commenters were divided in their opinions on whether advanced reactors should be more inherently safe but generally believed that the regulations should not require a degree of supplemental safety (beyond the top-level safety goals). Two (13, 14) believed that regulations should require more inherent safety. Four (3, 7, 8, 15) considered greater safety margins appropriate for advanced reactors and thought that NRC should encourage or give credit for margins incorporated by designers rather than require them. General Electric (15) stated that it would be more appropriate to reduce uncertainty in safety assessments. A number of others (2, 4, 5, 6, 9, 11, 12, 16, 18, 20) believed that a safety margin is not necessary because it would be redundant to a well-conceived design objective, would undermine the objective and lead to additional, unnecessary standards, and would not recognize the adequacy of the current level of safety. Two commenters (16, 19) suggested that a clear definition of design objectives would incorporate safety margins to the extent necessary and that separate margins would not be necessary.

No commenters advocated requirements for systems capable of functioning under conditions that exceed the design basis. Ecology/Alert (14) recommended requiring passive measures. A number of commenters (1, 2, 5, 6, 7) did not express a view as to which safety approach should be emphasized, but advocated leaving the choice to designers. A number of others (3, 8, 9, 10, 13, 14, 15, 16) suggested that designs should incorporate passive features which permit more time for operator response, but none stated a preference for requiring this.

Question 3 - Simplified Designs

"Should licensing regulations for advanced reactors mandate simplified designs which require the fewest operator actions, and the minimum number of components needed for achieving and maintaining safe shutdown conditions, thereby facilitating operator comprehension and reliable system function for off-normal conditions?"

While all commenters (except Ecology/Alert (14), who did not comment on Question 3) expressed the view that simplicity of design should not be a regulatory criterion, there was strong support for encouragement of simplicity in design (7, 8, 15). International Energy Associates Limited (IEAL) (13) stated that it is unnecessary for NRC to require simplicity; rather, inherent safety will yield simplicity. ORNL (7) believed that simpler designs are likely to make safety more predictable and verifiable and reduce burdens on both the operator and the regulator.

ORNL (7) gave further support to this concept by stating that facilities to enhance operator comprehension and understanding and to achieve reliable system functions should be required for both normal and off-normal conditions. It noted that these may be achieved by simplification of design to require fewer operator actions e.g., by providing the operator with automated assistance, improved information display and more extensive analytical systems.

Some commenters (2, 5, 8, 18) stated that the designer must be free to balance safety and ease of operation with plant availability, to balance greater time for operator action against plant economics, or to balance the extent of operator action against the degree of design complexity. DOE (5) further stated that regulatory policy should encourage flexibility.

Other views included the statement of GCRA (9) that additional hardware complexity should be avoided where increased operator understanding can achieve a net gain in safety. Westinghouse (18) stated that reducing the number of operator actions results in more system complexity because it requires more automatic functions. IEAL (13) said that NRC should consider a goal for advanced reactors of "walk away" safety--that is, the reactor system will shut itself down to a safe condition without any operator action. In summary, commenters generally were opposed to any regulation of simplicity in design, but believed that the regulatory policy should encourage it. They further believed that once the top-level safety criteria had been achieved, it is the responsibility of the designer to trade off or balance design simplicity and increased safety margin with economics of the plant operation.

Question 4 - Design Criteria

"Should the NRC develop general design criteria for advanced reactors by modifying the existing regulations, which were developed for the current generation of light water reactors, or by developing a new set of general design criteria applicable to specific concepts which are brought before the Commission?"

All but two commenters (18, 19) believed that a new set of design criteria should be developed. Westinghouse (18) believed that the current General Design Criteria are nonprescriptive and have proven to be "remarkably durable", and that a new set of criteria would not be consistent with stability and certainty in the licensing process. On the other hand, GPU Nuclear (19) felt that the existing General Design Criteria did need to be modified to be "less prescriptive and more criteria-oriented." EPRI (3) believed that the current criteria should be employed for evolutionary reactors, unless they could be shown to be excessively conservative, and that new criteria may need to be developed for advanced reactors based on radical design changes. The remainder of the commenters (except for four who did not comment on this question), felt that a new set of General Design Criteria should be developed. Two commenters (1, 11) felt that a unified set of criteria was necessary, with specific implementation being reactor type specific. Four commenters (1, 8, 9, 17) specifically stated that these should be developed and traceable to a safety goal based on acceptable risk to the public health and safety. Eight (4, 5, 6, 7, 9, 13, 15, 20) stated that they believed the criteria should be reactor type specific. Four (4, 6, 12, 13) felt that the industry and NRC should develop the criteria cooperatively. DOE (5) believed the criteria should be developed as part of the interactions between the NRC staff and each of the Department's advanced reactor programs during the development of the individual concepts.

Question 5 - Encouragement of Simplified and High Reliability Systems

"Should the NRC favor advanced reactor designs that concentrate the primary safety functions in very few large systems (rather than in multiple subsystems), thereby minimizing the need for complex benefit and cost balancing in the engineering of safe reactors?"

The 18 commenters that responded to the question supported the concept of design simplification. Fourteen commenters (1, 2, 3, 4, 5, 6, 8, 9, 11, 12, 15, 18, 19, 20) stated that they were opposed to the NRC favoring any particular design. Generally, they believed that it was up to industry to balance among concepts to arrive at a final design without the NRC being prescriptive in defining design requirements. One commenter (14) felt that the NRC should change emphasis from "defense-in-depth" to "simplifying reactor design, placing the core at least 10 feet underground, and doubling the thickness of the containment since the concept of 'defense in depth,' with multiple safety systems, simply adds to the number of buttons, levers and blinking lights." The remainder did not address this latter point.

Question 6 - Degree of Proof

"What degree of proof would be sufficient for the NRC to find that a new design is based on technology which is either proven or can be demonstrated by a satisfactory technology development program? For example, is it necessary or advisable to require a prototypical demonstration of an advanced reactor concept prior to final licensing of a commerical facility?"

Of the 20 commenters, 19 responded to this question. Nine of these (3, 4, 6, 11, 13, 15, 16, 18, 20) commented that whether or not a prototype of a facility would be required would be a function of the degree of departure from existing proven technology, the degree of uncertainty in the technology and any specific concerns with the technology. They stated that these factors would determine the need for prototype testing of either the facility or subsystems. Six (5, 6, 8, 9, 10, 15) believed that prototype testing should not be a requirement but an acceptable alternative to traditional methods for demonstrating compliance with the NRC's regulations. Four commenters (2, 7, 14, 19) felt that prototype testing for advanced reactors should be required. ORNL (7) cautioned that prototype testing would not be able to simulate such events as natural disasters, fire, sabotage, or aircraft impact. San Diego Gas and Electric (12) felt that the term "proof" was "totally inappropriate." Professional Analysis, Inc. (1) believed that a prototype facility is not sufficient to prove a concept due to the low probability of accidents of safety concern and that a concept could only be demonstrated through component prototype testing combined with risk analysis.

4 FORMULATION OF FINAL POLICY

4.1 Changes From Proposed Statement

Changes in the proposed Policy Statement that were incorporated in the final Policy Statement reflect review and consideration of the public comments and input provided by the staff to the Commissioners on August 21, 1985 (SECY-85-279) "Revised Advanced Reactor Policy Statement". In many cases the changes are for the purposes of clarification. The changes judged significant are described below in the order that they appear in the final Policy Statement:

(1) For clarification, an explicit list of three primary objectives has been added.

(2) For clarification, the definition for an advanced reactor has been added to differentiate between reactors of innovative design and reactors that represent evolutionary improvement over current generation light water reactors. This definition is discussed further in Section 5.1.

(3) The final policy statement explicitly deals with the question of enhanced margins of safety and safety goals with the added statement:

"Regarding advanced reactors, the Commission expects, as a minimum, at least the same degree of protection of the public and the environment that is required for current generation LWRs. Furthermore, the Commission expects that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions. The Commission also expects that advanced reactor designs will comply with the Commission's forthcoming Safety Goal Policy Statement."

This was added to make it clear that the Commission expects but does not require enhanced safety margins other than those that may be required by the safety goal policy.

(4) The listed desirable attributes that could assist in establishing the acceptability or licenseability of a proposed advanced reactor design has been increased from five to nine. These attributes are essentially the same as stated in the proposed Policy Statement except that they have been expanded for clarity. A proposed paragraph and attribute relating to increased standardization and shop fabrication was not carried over to the final Policy Statement since this is not unique to advanced reactors.

(5) A paragraph requesting early identification of plans for the use of proven technology and/or technology development programs was added in order to provide for early identification of issues which could impact standard plant approval and certification.

(6) The charter of the Advanced Reactors Group was expanded to "maintain knowledge of advanced reactor designs, developments and operating experience in other countries" and to "provide guidance regarding the timing and format of submittals for review." The implication that the NRC would review applications directly from foreign designers was removed.

4.2 Responses to Questions

The Commission's response to the six questions contained in the proposed Policy Statement are included in the final Policy Statement. These responses were developed considering the public comments received and the staff input provided in SECY-85-279. The questions and the Commission's response to each are contained on pages 14 through 19 in the Appendix. The questions and responses address the following topics: (1) Regulatory Approach, (2) Inherent Safety, (3) Simplified Designs, (4) Design Criteria, (5) Encouragement of Simplified and High Reliability Systems, (6) Degree of Proof.

5 GUIDELINES FOR UTILIZATION

The purpose of this section is to discuss the staff's plans for utilization and implementation of the guidance contained in the Advanced Reactor Policy Statement, including staff information needs and the approach to be used in the review of advanced reactor concepts. These plans are based both on the provisions of the Policy Statement and on certain related policies and regulations. It is not the purpose of this section to impose technical design requirements on advanced designs.

The following paragraphs reflect the staff's plans at this time which may be subject to evolutionary changes based on progress in the reviews of advanced reactor concepts and further developments in the LWR licensing structure. These plans are described here in order to provide guidance on the staff's information needs and the staff's approach to be used in the review of advanced reactor concepts. The staff reviews performed under the charter of the Policy Statement would occur before any formal application for review of either a one-of-a-kind plant or a standard plant, including design certification. In that sense they are the first of a multi-step process, leading toward construction and operation of an advanced nuclear power plant. However, this first step is not mandatory but reactor designers are encouraged to take advantage of it to obtain feedback early in the design process on licensing requirements. The review principles and results of the review discussed in this document would be expected to be used in subsequent reviews of that design, if and when a formal application for either a specific plant or a standard plant, including design certification, is filed.

5.1 Definition of Advanced Reactors

Advanced reactors are defined broadly in the Policy Statement as "those reactors that are significantly different from current generation light water reactors under construction or in operation and to include reactors that provide enhanced margins of safety or utilize simplified inherent or other innovative means to accomplish their safety functions." The staff considers that in this frame work the term "current generation reactors" refers also to the most recent evolutionary LWR designs (such as the General Electric-Advanced Boiling Water Reactor and the Westinghouse and Combustion Engineering Advanced Pressurized Water Reactors) which have improved safety features. The attributes listed in the Policy Statement for advanced reactor designs provide further definition. Also, in general, reactor designs that utilize inherent or passive safety features (features that perform their function without dependence on or influence by electric power, actuation of mechanical devices, or operator action) to perform their safety functions will be considered advanced reactors in the context of the Policy Statement. For each design submitted to the Commission for review, a determination will be made case by case about whether it should be classified as an advanced reactor and treated under the Policy Statement. In addition to the above, reactor designs that are classified as "advanced" and are reviewed as part of the staff's activities under the Advanced Reactor

Policy Statement, should have licensing requirements significantly different than those contained in the LWR Standard Review Plan (SRP), NUREG-0800. Accordingly, their review as an advanced reactor is intended to help ensure that appropriate regulatory requirements addressing the unique characteristics of these designs are developed in a timely fashion. At the present time certain high temperature gas-cooled reactor (HTGR) designs, liquid metal reactor (LMR) designs and innovative LWR designs* qualify as advanced reactor designs.

5.2 Advanced Reactors Group-Contacts and Information Needs

The Policy Statement sets out a charter for an Advanced Reactors Group (ARG) as follows:

"This group will be the focal point for NRC interaction with the Department of Energy, reactor designers and potential applicants, and will coordinate the development of regulatory criteria and guidance for proposed advanced reactors. In addition, the group will maintain knowledge of and expertise on advanced reactor designs, knowledge of developments and operating experience in other countries, and will provide guidance on an NRC-funded advanced reactor safety research program to ensure that it supports, and is consistent with, the Commission's advanced reactor policy. The Advanced Reactors Group will also provide guidance regarding the timing and format of submittals for review."

At the present time, the ARG functions as part of the Advanced Reactors and Generic Issues Branch, Division of Regulatory Applications, RES. The main function of the ARG is to serve as the focal point for NRC review of advanced reactors at the conceptual design stage. In general, the staff will implement the Policy Statement by reviewing designs at the conceptual stage (before any formal application), developing guidance on the licensing criteria applicable to that design and making a preliminary assessment of the potential of that design to meet those criteria. This review will be done primarily by the staff (under the coordination and direction of the ARG) and will include the involvement of the ACRS. Commission review will also be requested on those matters considered to have policy or other major implications.

Once a design has reached the point at which a formal application for review is submitted (either a plant specific license application or an application for standard plant review), its review will use and build on the initial reviews done by the ARG at the conceptual design stage.

Reactor designers proposing to initiate interactions with NRC on the review of an advanced reactor conceptual design should contact the Director, Office of Nuclear Regulatory Research, USNRC, Washington, DC 20555 prior to submitting design information for review.

Because of resource limitations, the NRC staff will have to determine case by case a priority for review of the proposed advanced concept considering such factors as:

*Those LWR designs that are consistent with the EPRI Advanced Light Water Reactor Design Requirements and/or contain significant safety advances beyond current licensing requirements may be reviewed under the guidelines of the Advanced Reactor Policy Statement.

(1) the potential of the design to result in an improvement in safety;

(2) level of support behind the design (industrial involvement, utility involvement);

(3) congressional or executive branch mandate; and

(4) utility interest.

In general, it is desired that the scope of review of an advanced concept include review of the entire plant (see Section 5.3.4 for further description). To enable the staff to perform a meaningful review, the following information is desired:

- Description of the plant design and its proposed design, safety and licensing criteria, including analysis of major accident scenarios demonstrating acceptable plant response.
- Probabilistic risk analysis (see Section 5.3.3 for further description).
- Description of those applicant sponsored R&D programs considered necessary to support development and licensing of the design.

The results of the staff review of this information would then be documented in a Safety Evaluation Report. This Safety Evaluation Report will identify the key safety issues associated with the design, provide guidance on the licensing criteria applicable to that design, provide an assessment of the adequacy of the applicant sponsored research and development programs proposed in support of the design and, in consideration of the above, assess whether any obvious impediments exist to licensing the advanced reactor design.

The following sections provide additional information regarding the staff review and information needs.

5.3 Review Approach and Related Policies, Practices and Regulations

As stated in the Advanced Reactor Policy Statement an advanced reactor must, as a minimum, have the same degree of protection of the public and environment as is required for current generation LWRs. However, enhanced margins of safety over current generation LWRs are expected. The degree of the enhanced margin of safety will be based on a judgment of the designs involving:

- the extent to which the designs incorporate those attributes listed as desirable in the Policy Statement,
- the uncertainties associated with the safety analysis and supporting base technology for the designs,
- the extent to which margins and defense-in-depth are employed to account for these uncertainties,

- the capability and margin included in the design to prevent and mitigate severe accidents, including compliance with the Commission's severe accident and safety goal policies,

- the previous operating experience, existing technology and proposed R&D supporting the design.

In consideration of the above, the staff will consider giving credit for enhanced safety characteristics incorporated into the design. This credit may be in the form of changed design criteria or administrative requirements. This section provides additional description of the key factors to be considered in the staff's review of an advanced design.

The existing regulatory structure for advanced reactors, of which the Policy Statement is now a part, ranges from top-level nonprescriptive criteria, such as the safety goal policy, to very detailed industry codes and standards. In reviewing an advanced reactor design at the conceptual design stage use will be made of the following NRC policies, practices, and regulations: (1) defense-in-depth philosophy, (2) safety goal policy, (3) severe accident policy (4) standardization policy, (5) existing LWR regulations and guidelines, where applicable, and (6) industry codes and standards.

How each of these items will be utilized by the staff in the review of advanced reactors is discussed below.

5.3.1 Defense-in-Depth Philosophy

There has been much discussion over the past several years about using less prescriptive or performance based licensing criteria and, it is noted, that novel design approaches could reduce the need for some types of safety equipment traditionally required on LWRs. Alternatives ranging from probabilistic based criteria to descriptive goal based criteria have been suggested. The use of such criteria is being explored and will be considered for advanced reactors (see Section 5.5). It is the staff's opinion that such criteria should be consistent with or the defense-in-depth philosophy. This is especially true when considering reactor types for which there is significantly less design, construction and operating experience as compared to LWRs. Accordingly, the staff believes that it is still essential and intends to employ engineering judgment and the defense-in-depth philosophy in the review of advanced reactors to account for uncertainties in the design. Such uncertainties may be in the areas of component/system performance, reliability, analytical tools or supporting technology. The application of defense-in-depth may take various forms, such as:

- requirements to prevent accidents, such as high reliability, redundancy and/or diversity in systems, structures and components,

- requirements to mitigate accidents, such as long response times, multiple barriers, or safety systems,

- requirements to contain radioactive materials.

The exact nature and extent of defense in depth to be required on an advanced design will be determined case by case on the merits of the design under review considering factors such as:

- reliability of safety systems
- supporting technology
- uncertainties in analytical tools, reliability, supporting data base
- margin in design for accidents beyond the design basis

5.3.2 Safety Goal Policy

On August 4, 1986, the Commission published a policy statement on "Safety Goals for the Operation of Nuclear Power Plants" (51 FR 28044). This policy statement focused on the radiological risks to the public from nuclear power plant operation and established goals that broadly define an acceptable level of such risks. Specific guidelines are being developed to establish a consistent level of safety between licensing criteria for advanced reactors and the safety goal policy. For advanced reactors these guidelines will be used, wherever appropriate.

5.3.3 Severe Accident and Source Term Policies

The Commission's "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" was issued on August 8, 1985 (50 FR 32138). Advanced reactors are expected to comply with the provisions of this policy that pertain to new plant applications. The staff is currently developing more detailed guidance regarding implementation of this policy statement. In addition, the regulatory procedures and criteria are being developed that will use the improved information from extensive research on radioactive material releases (i.e., source terms) under severe accident conditions. While some of the details of these severe accident and source term regulatory provisions may not be applicable to specific types of advanced reactors, advanced reactors are, in general, expected to conform to the relevant guidance they provide. Thus advanced reactor designers, when considering severe accidents and source terms at the conceptual design stage, are expected to show that the applicable portions of their designs meet "the intent of" or "the objectives of" the following:

(1) Demonstration of or commitment to compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island requirements for new plants as reflected in the construction permit rule, 10 CFR 50.34(f)

(2) Demonstration of or commitment to technical resolution of all applicable unresolved safety issues and the medium-priority and high-priority generic safety issues, including a special focus on ensuring the reliability of decay heat removal systems and the reliability of both ac and dc electrical supply systems;

(3) Completion of a probabilistic risk assessment (PRA) at the conceptual design stage and consideration of the severe accident vulnerabilities that the PRA exposes, along with the insights that it may add to the assurance that there is no undue risk to public health and safety.

Advanced reactor designers when addressing the above criteria are expected to take notice that the Policy Statement lists among the desirable attributes for proposed advanced reactor designs "designs that minimize the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity and independence in safety systems." Potentially, an advanced reactor could be proposed that would meet these preventative requirements with such sufficiency that relief could be justified in the type of source terms and severe accident mitigative features from that traditionally employed on LWRs. However, advanced designs are expected to consider a balance between prevention and mitigation consistent with the uncertainty associated with their analysis and to provide sufficient information to justify their design choices.

PRAs performed for the advanced reactor concepts should cover the whole plant, should address internal and external events as well as various plant operating states (full power, low power, refueling, etc.) and should confirm the bases for component and system selections, confirm the adequacy of overall plant design, be used to identify and correct any areas of high risk, and confirm the adequacy of plant response to severe accidents and mitigation measures. In addition, the PRA should be used to improve knowledge of component and structural reliability requirements and inservice inspection and testing needs. Any PRA must also estimate and factor in the uncertainties associated with it. These uncertainties must be factored into decisions which utilize PRA results.

In addition, analysis should be presented at the conceptual design stage to show the margin available in the design to accommodate events of low probability and to maintain protection of the public and environment.

5.3.4 Standardization Policy

On September 15, 1987, the Commission published a policy statement on "Nuclear Power Plant Standardization" (52 FR 34884). The development of advanced concepts should be consistent with the Commission's standardization goals and policy from the project's inception. Attention to the principles of standardization on advanced designs is not intended to discourage innovation but, rather, is intended to ensure that the end product is amenable to being standardized. Therefore, it is expected that advanced reactor designers should have as an ultimate goal the development of a standard plant design. Specific items regarding standardization which should be considered on advanced designs at the conceptual design stage are:

(1) The use of standardized practices in design, manufacture, construction, operation, and maintenance, to the extent possible;

(2) The use of standard components, structures, systems, and human engineering practices;

(3) The use of proven state-of-the-art technology, to the maximum extent possible, in the conception, design, and construction of any advanced reactor. Where the design deviates from state-of-the-art technology, a comprehensive research, development, and testing program will be necessary to demonstrate that the component or design feature being proposed performs with known characteristics and sufficient reliability to warrant standardization. To this end, the Commission stated in its Advanced Reactor

Policy Statement that it "favors the use of prototypical demonstration facilities as an acceptable way of resolving many safety related issues" (Section 5.4.4 provides additional information on prototype testing).

(4) As a minimum, at the conceptual design stage, the designer should present an essentially complete nuclear plant design for review rather than just the nuclear island or the safety-related components. Although the formal application for design approval[1] and design certification[2] may request design approval and certification of only interface criteria for certain systems, structures and components, a representative design for the complete plant should be presented at the conceptual design stage to allow the staff to assess the adequacy of the interface criteria and to aid in the review.

To ensure that each of the above considerations is adequately addressed, designers should provide more information at the conceptual design stage than a simple commitment to meet standardization goals. Information should be provided that describes their plans for achieving standardization.

5.3.5 Existing Regulations and Guidelines

The Standard Review Plan (SRP) Rule (10 CFR 50.34(g)) requires that applications for light-water-cooled nuclear power plant construction permits, operating licenses, preliminary design approvals and final design approvals docketed after May 17, 1982, include an evaluation of the facility against the SRP in effect on May 17, 1982, or the SRP in effect 6 months before the docket date of the application, whichever is later. The staff believes that advanced reactor designers should also review the SRP for applicability to their designs at the conceptual design stage. For those SRP sections identified as applicable, the advanced reactor design should be consistent with those requirements. Where advanced designs are different, designers should propose alternatives to the SRP requirements to account for the unique characteristics of their design.

In general, the staff will develop licensing criteria for advanced reactors by utilizing LWR criteria, where applicable, and by modifying existing criteria or developing new criteria to account for the unique characteristics of the design. The use of less or nonprescriptive criteria will be considered as discussed in Section 5.5.

[1]Design approval is addressed in 10 CFR 50, Appendix O, whereby a standard reactor design, or a major portion thereof, is reviewed and approved by the NRC staff and ACRS. The approved design would then be relied upon by the staff and ACRS in their review of individual license applications that reference the design. Design approval is a prerequisite to design certification.

[2]Certification through rulemaking is addressed in 10 CFR 50, Appendix O, whereby a standard reactor design, or a major portion thereof, is reviewed and approved by the NRC staff and then certified by the Commission for use through a formal rulemaking process. That portion of the design approved in a rulemaking proceeding would not be subject to review by the staff or challenge in individual license applications that reference the certified design.

5.3.6 Industry Codes and Standards

The use of industry codes and standards for the technical details of reactor and support systems designs has been a fundamental part of reactor licensing for many years. Over the years a large body of such codes and standards has been developed by experts in conjunction with the NRC and provide in most cases the essential details of how higher level criteria, policies, guides, rules, and regulations may be met. Like the use of appropriate operational experience, the use of these existing codes and standards, wherever practicable, is encouraged in advanced designs rather than proposing specialized unique approaches.

One of the reasons for the successful use of industry codes and standards in licensing LWRs is that the standards committees consist of a combination of members representing different interests and experiences such as reactor vendors, utilities, equipment manufacturers, and government and sometimes foreign representatives. The output of these committees represents a consensus on the important characteristics to be controlled in the areas covered by the standards. The staff encourages that committees such as the American Nuclear Society's ANS-53, "HTGR Management Committee" and ANS-54, "Committee on LMFBR Standards" be continued and used by advanced reactor designers.

5.3.7 Treatment of Sabotage

As indicated by the quote below from the Commission's Policy Statement on Severe Accidents, the importance of sabotage as a contributor to severe accident risk is recognized:

> "The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized as special considerations in the design and in the operating procedures developed for new plants."

In addition, Generic Issue A-29, "Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage," is one of the medium-priority Generic Safety Issues for which that policy expects new designs to demonstrate technical resolution.

The Advanced Reactor Policy Statement, in response to question number 1, indicated that in the area of sabotage the Commission intends to make use of existing and future regulations in reviewing advanced reactors. As such, the vulnerability of advanced reactors to sabotage is an important consideration and advanced reactors will be required to meet the same regulations regarding physical protection as LWRs. It is expected that, in many cases, advanced reactors, due to their inherent safety characteristics and simplified safety systems, will be less reliant upon physical security systems and procedures for protection against sabotage than current generation plants. Accordingly, at the conceptual design stage, advanced reactor designers should submit a short description of the advantages and disadvantages their design provides in protection from insider and outsider sabotage as compared to a current generation LWR.

5.4 Supporting Technology

The Policy Statement addresses the role of supporting technology several times as quoted below:

> "The Commission expects that these designs [for advanced reactors] will reflect the benefits of significant research and development work and include experience gained in operating the many power and development reactors both in the United States and throughout the world."

> "Among the attributes ...which, therefore, should be considered in advanced design are:... Design features that can be proven by citation of existing technology or which can be satisfactorily established by commitment to a suitable technology development program."

> "During the initial phase of advanced reactor development, the Commission particularly encourages design innovations which enhance safety and reliability... and which are either proven or can be demonstrated by a straight-forward technology development program."

In the subsections below are brief discussions on the use of supporting technology in the areas of operating experience, technology development, foreign information and data and use of prototype testing. Advanced reactor designers are expected to provide information on the application of each of these areas to their designs.

5.4.1 Operating Experience

The staff believes that the use of technology proven through operating experience is the most direct, least expensive and preferred means for the demonstration of licensability of reactor concepts. The available sources of operating experience should be used wherever possible. It is emphasized that sources of useful operating experience are not limited to reactors. For example, other industries provide valuable experience with water systems, testing and inspection procedures, control systems, and electrical and mechanical systems and components.

5.4.2 Technology Documentation and Development

Each submittal for review of an advanced design at the conceptual design stage should include a "technology development plan" or equivalent documentation. The technology development plan should document the scientific and engineering data that will be developed to support the design and safety analysis of the advanced reactor concept. This scientific and engineering data could include laboratory research, component development and testing, verifications during plant preoperational testing or startup, periodic testing and/or inspection during plant operation, and the use of a reactor prototype test. At the conceptual design stage the staff review will provide a preliminary assessment of the adequacy of the technology development plan for the design, utilizing engineering judgment, experience and insights gained from its review of the design.

5.4.3 Foreign Information and Data

Foreign programs can provide valuable design information, operating experience and basic data about advanced reactors. Regardless of the reliance to be placed on the information from foreign sources, each advanced reactor applicant submitting its design to the NRC for review should provide a summary of any applicable foreign reactor experience. This should include a discussion of major design differences and similarities, performance related experience and applicable research and development. How this information was factored into the advanced design should also be discussed. This is considered important because, in general, the experience base associated with advanced concepts is less than that for LWRs and the consideration of other experience is essential. The use of foreign data to support a U.S. advanced reactor design is acceptable provided the staff has sufficient access to the design, analysis and experimental data being used.

5.4.4 Use of Prototype Test

The Advanced Reactor Policy Statement does not require *a priori* that a prototype test reactor be constructed and operated; however, it does state that "The Commission favors the use of prototypical demonstration facilities as an acceptable way of resolving many safety related issues." The staff will, however, have to be satisfied for the design being reviewed that there is a basis for each claim regarding system and equipment performance and reliability. For reactor designs that depart significantly from proven technology, the staff favors the use of a prototype full-scale test facility to demonstrate those features of the design which are fundamental to its safety performance. This alternative has the potential for reducing or removing uncertainties because it will represent an integrated test of all plant systems under prototypical conditions, including the effects of construction, maintenance and operation. As part of the review of the conceptual design, the staff will make a case-by-case judgment about the need for a prototype test to resolve safety issues considering such factors as:

(1) Departure from proven technology,
(2) Uncertainties in performance and how they could be reduced,
(3) Degree of defense-in-depth, and
(4) Other R&D programs planned in support of the design.

It must be kept in mind that prototype tests cannot impact many of the uncertainties associated with certain types of events such as earthquakes, sabotage, and degraded core accidents. Risks from these types of events must be evaluated using engineering judgment and where applicable, probabilistic methods.

Regarding the need for a protypical demonstration facility to support design certification, the Commission stated in its Policy Statement on Nuclear Power Plant Standardization that "When an advanced design concept is sufficiently mature, e.g., through comprehensive, prototypical testing, an application for design certification could be made." Accordingly, advanced reactor designers should, at the conceptual design stage, describe their plans for the construction, testing and operation of a prototype plant to support design certification.

5.5 The Use of Less or Nonprescriptive Design Criteria

The Commission's guidance on and encouragement of the use of less prescriptive or nonprescriptive criteria in the regulatory process is given in its responses to two of the six questions contained in the proposed Advanced Reactor Policy Statement. These responses are included in the final Policy Statement, attached as an appendix to this document and are excerpted below:

Response to Question 1 (Regulatory Approach)

> "In developing additional criteria and guidance to address those characteristics which differ from LWRs less prescriptive criteria will be considered. The use of less prescriptive criteria will depend upon the design in question and the ability to verify compliance with the criteria. Advanced reactor designers are encouraged as part of their design submittals to propose specific review criteria or novel regulatory approaches which NRC might apply to their designs."

Response to Question 4 (Design Criteria)

> "In following this approach, it is the Commission's intent to establish, for each design reviewed, the licensing criteria that apply to that design. As stated in the response to Question No. 1, these criteria will be a combination of applicable LWR criteria and criteria developed to address the unique characteristics of that design. Reactor designers are encouraged to propose specific criteria and novel regulatory approaches which might apply to their design."

The Policy Statement does not include a definition for nonprescriptive criteria but does observe that "Many of the Commission's existing regulations, criteria, and guidelines are of a nonprescriptive nature..." and cites the safety goal policy as an example. The development of less prescriptive regulatory requirements is also a goal in "NRC Policy and Planning Guidance," NUREG-0885, Issue 5, 1986.

The role of and the justification for the use of less or nonprescriptive licensing criteria in those areas where existing LWR criteria do not apply is an area which will receive considerable emphasis in the review of advanced reactors. While the use of less or nonprescriptive criteria may be desirable in many cases, certain information and study is needed to assure that, in the event they are used, an acceptable level of safety is attained. To illustrate the information and considerations which need to be addressed in this area, a list of items follows that designers should be prepared to address during the course of an advanced reactor review if they propose to use less or nonprescriptive criteria for their designs. This list serves to illustrate the way newly proposed criteria will be examined by the staff. The fact that the staff will carefully evaluate any proposed new criteria is not intended to discourage their development. On the contrary, the staff encourages the development of improved regulatory approaches and will give high priority to reviews of new criteria to support the development of advanced reactors. In general, the staff expects advanced reactor designers to propose those criteria which, in their judgment, apply to their design, including any less or nonprescriptive criteria. Where such criteria depart from the traditional level of specificity employed on LWRs regarding design configuration and plant performance, the following information should be provided to justify and clarify the use of the less or nonprescriptive criteria and to assist the NRC in making the requisite assessment:

(1) A description of why such criteria are being proposed and what changes in the scope or type of NRC regulation are desired or implied by the use of the new criteria. For example, if probabilistic based criteria are proposed, will NRC be required to regulate data bases, reliability assurance programs or maintenance programs to help ensure reliability goals are met?

(2) A description of the way the proposed criteria will lead to a safer plant design and not detract from safety. For example, would the use of the proposed criteria lead to the use of components, systems or structures of superior reliability than would be required by the traditional regulatory structure?

(3) A description of the extent to which less or nonprescriptive criteria are to be employed in the regulation of the proposed design, including the proper mix between nonprescriptive and deterministic criteria, and considering the need to preserve the defense-in-depth philosophy to account for uncertainties and unknowns.

(4) Standardization of design has long been encouraged by the Commission. It is possible that the adoption of less or nonprescriptive regulations could work against standardization. Although a less or nonprescriptive approach may seem attractive for new and innovative designs it should be noted that in the past this flexibility has produced instead a multiplicity of designs with no clear advantage among them. Therefore, a description would be useful of the compatibility of the proposed regulatory approach with the Commission's standardization goals, along with a description of how the nonprescriptive regulation should be implemented to ensure there is no detrimental effect on the Commission's standardization efforts.

(5) The scope of the analyses to be used to justify and implement the proposed criteria should be discussed. This should include discussion of the way analyses are to be maintained over the life of the plant. For example, to implement reliability based criteria, should the reliability analysis be updated over the life of the plant to reflect both plant specific and industry wide operating experience?

6 REFERENCES

Code of Federal Regulations, Title 10, "Energy," U.S. Government Printing Office, Washington, D.C., January 1, 1987.

U.S. Nuclear Regulatory Commission, "Regulation of Advanced Nuclear Power Plants; Statement of Policy," 51 FR 24643, July 8, 1986.

U.S. Nuclear Regulatory Commission, "Proposed Policy for Regulation of Advanced Nuclear Power Plants," 50 FR 11884, March 26, 1985.

U.S. Nuclear Regulatory Commission, SECY-85-279, "Revised Advanced Reactor Policy Statement, August 21, 1985.

U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," July 1981.

U.S. Nuclear Regulatory Commission, "Safety Goals for the Operation of Nuclear Power Plants," 51 FR 28044, August 4, 1986.

U.S. Nuclear Regulatory Commission, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," (50 FR 32138) August 8, 1985.

U.S. Nuclear Regulatory Commission, "Nuclear Power Plant Standardization, Policy Statement," 52 FR 34884, September 15, 1987.

U.S. Nuclear Regulatory Commission, NUREG-0885, "NRC Policy and Planning." Issued February 5, 1986.

APPENDIX

NUCLEAR REGULATORY COMMISSION

10 CFR PART 50

REGULATION OF ADVANCED NUCLEAR POWER PLANTS;
STATEMENT OF POLICY

AGENCY: Nuclear Regulatory Commission.

ACTION: Final Policy Statement.

SUMMARY: The Nuclear Regulatory Commission intends to improve the licensing environment for advanced nuclear power reactors to minimize complexity and uncertainty in the regulatory process. This statement gives the Commission's policy regarding the review of, and desired characteristics associated with, advanced reactors. This policy statement is a revision of the "Proposed Policy for Regulation of Advanced Nuclear Power Plants" that was published for comment on March 26, 1985 (50 FR 11884).

EFFECTIVE DATE:

FOR FURTHER INFORMATION CONTACT: Ken Herring and Dennis Rathbun, Office of Policy Evaluation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone: 202-634-3295.

SUPPLEMENTARY INFORMATION:

BACKGROUND

The Commission's primary objectives in issuing an advanced reactor policy statement are threefold:

- First, to encourage the earliest possible interaction of applicant, vendors, and government agencies, with the NRC;

- Second, to provide all interested parties, including the public, with the Commission's views concerning the desired characteristics of advanced reactor designs; and

- Third, to express the Commission's intent to issue timely comment on the implications of such designs for safety and the regulatory process.

Such interaction and guidance early in the design process should enhance stability and predictability in the licensing and regulation of advanced reactors.

Advanced reactors are considered here to be those reactors that are significantly different from current generation light water reactors under construction or in operation.

The Commission expects that these designs will reflect the benefits of significant research and development work, and include the experience gained in operating the many power and development reactors both in the United States and throughout the world. The Commission expects that advanced reactors would provide more margin prior to exceeding safety limits and/or utilize simplified, inherent, passive, or other innovative means to reliably accomplish their safety functions. The Commission expects, as a minimum, at least the same degree of protection of the public and the environment that is required for current generation LWRs. For the longer term, the Commission expects designs to provide enhanced margins of safety. To provide regulatory guidance during the development phase of advanced reactor design, the Commission wishes to encourage the earliest possible interaction between the NRC and other government agencies, reactor designers, and potential licensees.

This advanced reactor policy statement sets forth the general characteristics of advanced reactor design, which the Commission believes advanced reactors should exhibit, to increase assurance of safety, to improve public understanding, and to promote more effective regulation. As the agency responsible for assuring the protection of the public from the potential hazards of nuclear power plants, the Commission will keep the public informed of its judgment on the safety aspects of advanced reactor designs as such designs come before the Commission.

A report which discusses the revisions to the Policy Statement will be published shortly as NUREG-XXX "TITLE." A copy of NUREG-XXX will be available for inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C.

REGULATORY POLICY FOR ADVANCED REACTORS

The Commission intends to improve the licensing environment for advanced nuclear power reactors and to minimize complexity and uncertainty in the regulatory process. This is a statement of the Commission's policy regarding the review of, and desired characteristics associated with, advanced reactors. This policy statement is a revision of the "Proposed Policy for Regulation of Advanced Nuclear Power Plants" that was published for comment on March 26, 1985 (50 FR 11884).

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- First, to encourage the earliest possible interaction of applicant, vendors, and government agencies, with the NRC;

- Second, to provide all interested parties, including the public, with the Commission's views concerning the desired characteristics of advanced reactor designs; and

- Third, to express the Commission's intent to issue timely comment on the implications of such designs for safety and the regulatory process.

Such interaction and guidance early in the design process should enhance stability and predictability in the licensing and regulation of advanced reactors.

The Commission considers the term "Advanced" to apply to reactors that are significantly different from current generation light water reactors (LWRs) now under construction, or in operation and to include reactors that provide enhanced margins of safety or utilize simplified inherent or other innovative means to accomplish their safety functions.

Currently, certain high temperature gas-cooled reactors (HTGRs), liquid metal reactors (LMRs), and light-water reactors (LWRs) of innovative design are considered advanced designs.

LEGISLATIVE BACKGROUND

The Commission's policy with respect to regulation of advanced reactors is guided by the legislative background. The Energy Reorganization Act of 1974, which established the Nuclear Regulatory Commission, specifically delegated to NRC "licensing and related regulatory authority" for demonstration nuclear reactors other than those already in existence "...when operated as part of the power generation facilities of an electric utility system, or when operating in any other manner for the purpose of demonstrating the suitability for commercial application of such a reactor..." The Energy Research and Development Administration (now the Department of Energy) was charged with "...encouraging and conducting research and development, including demonstration of commercial feasibility and practical applications of the extraction, conversion, storage, transmission, and utilization phases related to the development and use of energy from...nuclear...sources."

Under Section 205 of the Energy Reorganization Act, the NRC must provide a "Long-term plan for projects for the development of new or improved safety systems for nuclear power plants." The NRC is precluded from designing, or doing research on, complete new designs for the purpose of establishing or developing their commercial potential. 1/

PREVIOUS EXPERIENCE

The Commission has had experience in the regulation of HTGRs and LMRs as well as in the regulation of LWRs. The NRC has reviewed several applications for HTGR construction permits, and a conceptual design for a gas-cooled breeder reactor, and has granted an operating license to Peach Bottom-1 and to Fort St. Vrain. The NRC also expended substantial effort from 1975 to 1979 in reviewing General Atomic's Standard high-temperature, gas-cooled nuclear reactor steam supply system (GASSAR). In addition, the NRC has supported a modest program of safety research on gas-cooled reactors every year since the agency's inception.

The Commission has also had experience in the review and licensing of LMRs. In the past the FERMI-1 and SEFOR reactors were reviewed and licensed. DOE's Fast Flux Test Facility (FFTF) was reviewed and approved but not licensed, and a formal construction permit licensing proceeding was

1/ The general principal defining the scope of NRC's research can be described as avoiding a conflict of interest-- "[NRC] should never be placed in a position to generate , and then have to defend, basic design data of its own" as expressed in the Conference Report to the Energy Reorganization Act of 1974.

conducted for the Clinch River Breeder Reactor (CRBR). The CRBR was subject to the same regulatory process as any current commerical nuclear power project.

Finally, the Commission notes that the precedent for the broad policy approach to advanced reactor regulation, as proposed here, is firmly established in the 1979 Nonproliferation Alternative Systems Assessment Program (NASAP), wherein the NRC considered the safety and licensability of a variety of advanced reactor concepts within the context of nonproliferation objectives. The concepts considered and reported on by the NRC in the 1979 study ranged from preliminary conceptual designs to variations of existing (LWR) power plants designs.

COMMISSION POLICY

Consistent with its legislative mandate, the Commission's policy with respect to regulating nuclear power reactors is to assure adequate protection of the public health and safety and the environment. Regarding advanced reactors, the Commission expects, as a minimum, at least the same degree of protection of the public and the environment that is required for current generation LWRs. Furthermore, the Commission expects that advanced reactors will provide enhanced margins of safety and/or utilize simplified, inherent, passive, or other innovative means to accomplish their safety functions. The Commission also expects that advanced reactor designs will comply with the Commission's forthcoming safety goal policy statement.

Among the attributes which could assist in establishing the acceptability or licensability of a proposed advanced reactor design, and which therefore should be considered in advanced designed are:

- Highly reliable and less complex shutdown and decay heat removal systems. The use of inherent or passive means to accomplish this objective is encouraged (negative temperature coefficient, natural circulation).

- Longer time constants and sufficient instrumentation to allow for more diagnosis and management prior to reaching safety systems challenge and/or exposure of vital equipment to adverse conditions.

- Simplified safety systems which, were possible, reduce required operator actions, equipment subjected to severe environmental conditions, and components needed for maintaining safe shutdown conditions. Such simplified systems should facilitate operator comprehension, reliable system function, and more straight-forward engineering analysis.

- Designs that minimize the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity and independence in safety systems.

- Designs that provide reliable equipment in the balance of plant, (or safety-system independence from balance of plant) to reduce the number of challenges to safety systems.

- Designs that provide easily maintainable equipment and components.

- Designs that reduce potential radiation exposures to plant personnel.

- Designs that incorporate defense-in-depth philosophy by maintaining multiple barriers against radiation release, and by reducing the potential for an consequences of severe accidents.

- Design features that can be proven by citation of existing technology or which can be satisfactorily established by commitment to a suitable technology development program.

If specific advanced reactor designs with some of all of the above of the foregoing attributes are brought to the NRC for comment and/or evaluation, the Commission can develop preliminary design safety evaluation and licensing criteria for their safety related aspects. Combination of some or all of the above attributes may help obtain early licensing approval with minimum regulatory burden. Designs with some or all of these attributes are also likely to be more readily understood by the general public. Indeed, the number and nature of the regulatory requirements may depend on the extent to which an individual advanced reactor design incorporates general attributes such as listed above. However, until such time as conceptual designs are submitted, the Commission believes that regulatory guidance must be sufficiently general to avoid placing unnecessary constraints on the development of new design concepts.

To provide for more timely and effective regulation of advanced reactors, the Commission encourages the earliest possible interaction of applicants, vendors, other government agencies, and the NRC to provide for early identification of regulatory requirements for advanced reactors, and to provide all interested parties, including the public, with a timely, independent assessment of the safety characteristics of advanced reactor designs. Such licensing interaction and guidance early in the design process, will contribute toward minimizing complexity and adding stability and predictability in the licensing and regulation of advanced reactors.

While the NRC itself does not develop new designs, the Commission intends to develop the capability for timely assessment and response to innovative and advanced designs that might be presented for NRC review. Prior experience has shown that new reactor designs -- even variations of established designs -- may involve technical problems that must be solved in order to assure adequate protection of the public health and safety. The earlier such design problems are identified, the earlier satisfactory resolution can be achieved. Prospective applicants are reminded that, while the NRC will undertake to review and comment on new design concepts, the applicants are responsinle for documentation and research necessary to support any specific license application. (NRC research is conducted to provide the technical bases for rulemaking and regulatory decisions; to support licensing and inspection activities; and to increase NRC's understanding of phenomena for which analytical methods are needed in regulatory activities).

During the initial phase of advanced reactor development, the Commission particularly encourages design innovations which enhance safety and reliability (such as those described above) and which generally depend on technology which is either proven or can be demonstrated by a straight-forward technology development program. In the absence of a significant history of operating experience on an advanced concept reactor, plans for innovative use of proven technology and/or new technology development programs should be presented to the NRC for review as early as possible, so that the NRC can assess how the proposed program might influence regulatory requirements. To achieve these borad objectives, an Advanced Reactors Group has been established in the Office of Nuclear Reactor Regulation. This group will be the focal point for NRC interaction with the Department of Energy, reactor designers and potential applicants, and will coordinate the development of regulatory criteria and guidance for proposed advanced reactors. In addition, the group will maintain knowledge of advanced reactor designs, developments and operating experience in other countries, and will provide guidance on an NRC-funded advanced reactor safety research program to ensure that it supports, and is consistent with, the Commission's advanced reactor policy. The Advanced Reactors Group will also provide guidance regarding the timing and format of submittals for review. The Advisory Committee on Reactor Safeguards (ACRS) will play a significant role in reviewing proposed advanced reactor design concepts and supporting activities.

COMMISSION POSITION REGARDING POLICY STATEMENT QUESTIONS

Six questions pertaining to the proposed policy for advanced reactors were included for comment in the original policy statement. The public responses to these questions are summarized in the "Abstract of Comments" section. After careful consideration of the public comments, the Commission response to the issues raised in each question is as follows:

Question 1. Should NRC's regulatory approach be revised to reduce dependence on prescriptive regulations and, instead, establish less prescriptive design objectives, such as performance standards? If so, in what aspects of nuclear power plant design (For Example, reactor core power density, reactor core heat removal, containment, and siting) might the performance standards approach be applied most effectively? How could implementation of these performance standards be verified?

COMMISSION RESPONSE

Many of the Commission's existing regulations, criteria, and guidelines are of a nonprescriptive nature, and the extent to which the Commission's proposed safety goals, (which are also of a nonprescriptive nature) will be used in the regulation of nuclear reactors is currently being evaluated. In the review and regulation of advanced reactors the Commission intends to make use of existing and future regulations where they are applicable to advanced reactors. Many such regulations are expected to be of a nonprescriptive nature. The areas where existing regulations and guidelines would be used include: quality assurance, equipment qualification, external events, sabotage, fire protection, radiation protection, and operator training and qualification. In developing additional criteria and guidance to address those characteristics which differ from LWRs less prescriptive criteria will be considered. The use of less prescriptive criteria will depend upon the design in question and the ability to verify compliance with the criteria. Advanced reactor designers are encouraged as part of their design submittals to propose specific review criteria or novel regulatory approaches which NRC might apply to their designs.

Question 2. Should the regulations for advanced reactors require more inherent safety margin for their design? If so, should the emphasis be on providing features that permit more time for operator response to off-normal conditions, or should the emphasis be on providing systems that are capablle of functioning under conditions that exceed the design basis?

Commission Response

The Commission encourages the incorporation of enhanced margins of safety in advanced designs and will encourage the use of designs that accomplish their safety functions in as reliable and simplified a fashion as practical. The Commission considers inherent or passive safety systems to have the potential for high reliability and encourages the consideration of such means (in lieu of active systems) in advanced designs.

To encourage such action the Commission, in its review of these advanced designs, will look favorably on designs with greater safety margin and/or highly reliable safety systems. Such desirable features can be design-related or can take the form of reduced administrative requirements.

Question 3. Should licensing regulations for advanced reactors mandate simplified designs which require the fewest operator actions, and the minimum number of components needed for achieving and maintaining safe shutdown conditions, thereby facilitating operator comprehension and reliable system function for off-normal conditions?

Commission Response

The Commission will encourage designs which are simpler and more reliable in accomplishing their safety functions. While current generation nuclear power plants, in operation or under construction represent no undue risk to either the public or the environment, the Commission believes that reactors with improved safety characteristics can and will be developed. Such improved safety characteristics support the Commission's Long-range Goal of minimizing the risk to the public and the environment through the "ALARA" approach.

Question 4. Should the NRC develop general design criteria for advanced reactors by modifying the existing regulations, which were developed for the current generation of light water reactors, or by developing a new set of general design criteria applicable to specific concepts which are brought before the Commission?

Commission Response

In developing licensing criteria for advanced reactors, the Commission intends to build upon existing regulations wherever practical, as discussed in the response to Question No. 1. In following this approach, it is the Commission's intent to establish, for each design reviewed, the licensing criteria that apply to that design. As stated in the response to Question No. 1, these criteria will be a combination of applicable LWR criteria and criteria developed to address the unique characteristics of that design. Reactor designers are encouraged to propose specific criteria and novel regulatory approaches which might apply to their design.

Question 5. Should the NRC favor advanced reactor designs that concentrate the primary safety functions in very few large systems (rather than in multiple subsystems), thereby minimizing the need for complex benefit and cost balancing in the engineering of safe reactors?

Commission Response

While the NRC will not necessarily favor one design approach over another in regard to the number of safety systems, the NRC will encourage the use of simplified systems and systems of high reliability for the accomplishment of safety functions.

Question 6. What degree of proof would be sufficient for the NRC to find that a new design is based on technology which is either proven or can be demonstrated by a satisfactory technology development program? For example, is it necessary or advisable to require a prototypical demonstration of an advanced reactor concept prior to final liensing of a commercial facility?

Commission Response

The Commission requires proof of performance of certain safety-related components, systems or structures prior to issuing a license on a design. For LWR's this proof has traditionally been in the form of analysis, testing, and research development sufficient to demonstrate the performance of the item in question. Similar proof of performance for certain components, systems or structures for advanced reactors will also be required. The requisite proof will be design dependent. Therefore, the Commission's specific assessment of a safety technology development program for an advanced reactor design, or of the possible need for a prototypical demonstration of that design can be determined only by review of a specific design. However, the Commission favors the use of prototypical demonstration facilities as an acceptable way of resolving many safety related issues.

The dissenting views of Commissioner Asselstine and the additional views of Commissioner Bernthal are attached.

FOR THE NUCLEAR REGULATORY COMMISSION

Samuel J. Chilk
Secretary of the Commission

Dated at Washington, D.C.
This 1st day of July, 1986

Additional Views of Commissioner Bernthal on Advanced Reactor Policy Statement

Less than three years ago, the Commission began to consider seriously its responsibility (and the mandate of Congress) to become more deeply involved with early review and comment on new and advanced reactor design concepts. Such early design review has long been a commonplace within the Federal Aviation Administration, for example, where timely FAA review and comment on new airframe design proposals is longstanding tradition.

The Commission has since undergone considerable progressive evolution in its thinking on this subject, and in this document the Commission, for the first time, has gone on record as supporting such timely, anticipatory safety review of new design concepts. In addition, the Commission has plainly stated its expectation that next-generation reactors will exhibit enhanced and simplified safety characteristics, and has set down broad and diverse guidelines for how it believes such characteristics might be achieved.

There is little doubt that this policy statement as it stands fails to conform in some respect with each Commissioner's ideal of what such a statement should be. But I find the statement to be a major step forward; it commits the Commission to exactly the kind of "proactive" planning that Commissioner Asselstine still seems to find absent.

Many of the specific objections raised by my colleague are puzzling. His sweeping statement that "containment capabilities are minimized to reduce costs" and "core power densities have been driven to the limits of materials capabilities and our understanding of decay heat removal phenomena" are scientifically insupportable and inconsistent with the facts as generally understood. The fact is that containment capabilities were in general designed to cope with well-known accident scenarios, and core power limits were conservatively derived.

Nor <u>should</u> the Commission insist on "specific requirements" for advanced reactor designs -- indeed, such insistence would go far beyond our mandate (and our capability). Such specificity was never the intent of this policy statement. Detailed specification of systems such as containment, for example, was never contemplated as an objective of the "advanced reactor" policy; indeed, one can imagine advanced reactor designs that might demand <u>less</u> containment capability than current generation LWR plants.

In sum, it was never intended that this statement promulgate "a set of safety requirements". As the statement notes, broad safety requirements are to be addressed in the Commission's forthcoming Safety Goal Policy Statement (to the extent they are not already addressed in the Severe Accident Policy Statement and elsewhere). Furthermore, The Commission's response to Question 6 makes clear its encouragement of plant designs firmly grounded in prototypical plants -- just as Commissioner Asselstine desires.

Nor does this policy "accept the next generation of U.S. power plants if [they] provide a level of safety equivalent to that achieved in the U.S. designs that were completed 10 years ago." There is necessarily room for interpretation in the Commission's pronouncement, but whether or not the Commission might ever issue (or be asked to issue) new construction permits replicating "current generation plants, plants whose designs were largely frozen more than 10 years ago" is not the question. It is amply clear from this policy statement that "the Commission expects that advanced [emphasis added] reactors will provide enhanced margins of safety...", and the Commission has broadly defined "advanced" to include reactors that lie beyond current generation designs.

Finally, Commissioner Asselstine's comment that the "next generation of plants should be more reliable, more forgiving, simpler, easier to construct, easier to operate, and easier to maintain than the current generation" is a nice synopsis of the broad guidelines clearly set forth in this policy statement. I am pleased that he concurs in the desirability of those traits.

Dissenting Views of Commissioner Asselstine

I do rot believe that this advanced reactor policy statement provides the sound regulatory basis needed to support a new generation of nuclear power plants in this country. This policy statement encourages, but does not require, safety improvements in advanced reactor design, and expresses a willingness on NRC's part to conduct safety reviews of advanced reactor design concepts so that NRC will be in a position to act on any future plant or design license application. The primary decision made in developing this policy is the commitment to maintain a small advanced reactor group within the Agency that would serve as the focal point for interaction with reactor design groups. However it appears that even this commitment may be in jeopardy given current budgetary constraints.

I believe that more is needed to articulate an effective regulatory policy and to ensure a successful program for future nuclear power plants in this country, whether those plants are of a type similar to current light water reactors or whether they are of more fundamentally different design. Such a policy should reconsider the Commission's regulatory practices of the past thirty years. Those past practices can be characterized as primarily a reactive regulatory regime to what the designers propose. It leaves resolution of issues to what one industry executive has called the rough, tough, surly competitive elements. Safety systems are limited because of cost considerations. Containment capabilities are minimized to reduce costs. 1/ Core power densities have been driven to the limits of materials capabilities and our understanding of decay heat removal phenomena. 2/ And the balance of plant is designed to lower standards than the reactor systems to minimize costs. These competitive forces are what led to the level of safety achieved in the current generation of nuclear power plants and are in part responsible for the poor performance of some of our plants.

The NRC and AEC before it have often avoided developing stringent specifications or design requirements because of a fear that if the Commission were to be too specific in its requirements, the emerging industry might be slowed in its growth and innovation might be discouraged. That argument might have had some validity in the 1960's and 1970's when the current generation of reactors was being designed without the benefit of signifi-

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1/ For example, to keep the containment size down, crucial pumps, heat exchangers, and emergency water supplies have been located outside the containment, which results in flow paths for highly contaminated water that effectively bypass the containment. In addition, containment volumes and design pressures have been traded-off for pressure suppression schemes that substantially complicate safety analyses and that add additional vulnerabilities to the public health and safety. Initially containments were intended to be an independent barrier to substantial releases given a core meltdown. Some of that defense-in-depth was given up for the sake of costs, when large power reactors came on the scene in the mid-1960's and it became known that the decay heat and the core meltdown phenomena could fail the containment.

2/ For example, in the event of a loss of coolant accident, external water supplies must be rapidly injected into the core to keep it from melting. While some relatively small-scale integral experiments on loss of coolant phenomena have been completed, there are still multinational supported research programs underway to further examine thermal hydraulic phenomena during accidents. Further, we are just beginning expensive, integral effects tests on thermal hydraulic phenomena associated with a class of pressurized water reactors.

cant operating experience or data. However, now that we have considerable worldwide experience with a large variety of nuclear reactor designs, I believe it is time for NRC to become more proactive in what it will require of future generations of reactors.

Following the TMI-2 accident, the notion of a demarcation between the current generation of plants and a future generation of plants was raised, with the distinction that the latter would be designed based on a reformulation of the Siting Criteria and General Design Criteria to reflect all that had been learned over the years, including the broader lessons of TMI-2. Thus, the TMI Action Plan was developed with the current generation of plants in mind, leaving open the question of possible broader changes for a future generation of plants. One such broad change could be to go beyond the so-called single failure criterion which experience shows may not be serving us well. The June 9, 1985 accident at Davis-Besse is a case in point where 14 separate failures occurred.

Many foreign countries are requiring four independent trains of safety systems whereas NRC requires only two. When NRC reviews advanced designs such as the one being jointly developed by a U.S. vendor and a foreign country, the NRC staff does not require as prudent additional safety features being required by the foreign country. Rather, Commission practices and procedures require a cost-benefit analysis to justify any additional safety feature. This analysis is typically incomplete and often crude. Furthermore, the Commission gives little consideration to the enormous uncertainties in reactor risks in its decisionmaking process. This approach to reactor safety needs improvement.

There has been insufficient thought and effort in developing a map for the future. The Advanced Reactor Policy Statement provides no guidance on what containment capabilities will be required; on whether the single failure criterion is adequate for the future; on acceptable core power densities (an issue which has significant bearing on the core meltdown risks to the public); and on the root causes of the core meltdown risks that might be addressed by design improvements in a future generation of reactors. Nor is there guidance on what standards the balance of plant must meet. Nothing is said about the fuel cycle and the process for licensing the fuel cycle associated with some of the advanced designs currently being examined. For example, one problem area presented by some designs is the proliferation potential of the reactor's fuel cycle. This fuel cycle could present the need for the Commission to reopen the aborted proceeding on plutonium recycle. And, finally the Commission gives essentially no guidance on whether a prototypical plant will be required before allowing widespread use of that design. This policy statement encourages much, just like the Commission encourages excellence in operations. However, the Commission too often accepts far less. I would have expected that NRC would approach a future generation of nuclear power plants with an attitude of correcting past weaknesses. Unfortunately, the Advanced Reactor Policy Statement does not reflect that kind of attitude.

Other countries with extensive nuclear power programs appear to be designing, constructing, operating and maintaining better nuclear power plants than those of this country. Foreign countries are demanding more safety and reliability in their current generation of plants than the NRC is requiring of the U.S. plants. Yet, this Advanced Reactor Policy Statement accepts the next generation of U.S. power plants if such a design provides a level of safety equivalent to that achieved in the U.S. designs that were completed over 10 years ago. I do not think such a policy serves the country well. My concern is not merely that we should keep up with others. Rather, my concern is that the current generation of plants is still surprising us in their performance. As the Commission has recently acknowledged to the Congress, the current generation of nuclear power plants in this country can best be characterized as a complex technology that is not fully mature. There remain great uncertainties in the level of risk they pose to the public. In such circumstances, I believe prudent decisionmaking should come down on the side of improved safety, not only for the current generation of plants but for the next generation as well.

If there is to be a future generation of nuclear power plants and if the nuclear option is to be an important element of the nation's future energy mix, then the NRC, the vendors, the utilities, and the Congress must ensure that the next generation of power plants is substantially better than the current generation. The next generation of plants should be more reliable, more forgiving, simpler, easier to construct, easier to operate, and easier to maintain than the current generation. Any design that does not accomplish this is not acceptable in my view. I say this for a straightforward reason. We cannot afford to will to the future reactor designs that have a fifty percent chance of a core meltdown every ten to twenty years in a population of 100 reactors. We should not will to the future the great uncertainties in safety levels that exist today. Nor should we will to the future consumer reactor designs that have a 50 to 60 percent capacity factor.

We must step back and examine the strengths and weaknesses of past and current designs and the approaches taken in getting where we are today. Only then, in my view, can we intelligently map a course for the future. I am encouraged that there is a segment within the industry that is undertaking a fresh look at the nuclear technology. The forward-looking members of the industry are attempting to generate a set of requirements that, from the standpoint of the utilities, must be met before utilities will consider placing new orders. I find it disappointing that the NRC is unwilling to generate a set of safety requirements for the next generation of power plants.

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May 10, 2001

Mr. Thomas L King
Office of Nuclear Regulatory Research
U.S Nuclear Regulatory Commission
Washington, D.C 20555

DOCKET: Project 713

RE: Regulatory Issues Related to the Pebble Bed Modular Reactor (PBMR)

Dear Mr. King

As you know, Exelon Generation (Exelon) is currently participating in a detailed feasibility study of the PBMR If the results of this study are favorable, Exelon intends to seek the regulatory approval required to construct and operate a PBMR as a merchant power plant in the United States. As discussed in the meeting with NRC on April 30, 2001, Exelon has identified a number of NRC regulations that could pose an undue and unintended burden when applied to a gas-cooled modular reactor facility or merchant plants This letter provides the basis for that discussion.

In general, NRC regulations governing nuclear power plants were developed for large light water reactors (LWRs) owned and operated by electric utilities. For the most part, these regulations were not designed for and do not contemplate gas-cooled modular reactor facilities being operated as merchant plants The regulations creating potential burdens include the following.

- License requirements in 10 CFR § 50.10
- Financial protection requirements in 10 CFR Part 140
- Decommission funding requirements in 10 CFR § 50 75(e)
- Requirements for an antitrust review under 10 CFR § 50 33a
- Requirements on annual fees in 10 CFR Part 171
- Operator staffing requirements in 10 CFR § 50.54(m)
- Minimum decommissioning costs in 10 CFR § 50.75(c)
- Fuel cycle impacts in 10 CFR Part 51
- Financial qualifications in 10 CFR § 50.33(f)

The enclosed position papers summarize the additional burden that each of these requirements could impose on the PBMR, Exelon's proposals concerning actions the NRC could take to eliminate or mitigate those burdens, and the reasons why those actions are in accordance with NRC's legal authority under the Atomic Energy Act of 1954, as amended We have not included a paper on emergency planning as indicated in our meeting because existing NRC emergency planning regulations provide latitude for NRC to address gas-cooled reactors on a case-by-case basis.

The attached table summarizes Exelon's proposals with respect to these regulations. For the first PBMR facility, Exelon will include within its license application a request for an exemption from most of these regulations, and in other cases will provide information to resolve the matters addressed by the

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per Amy Cubbage

regulations. To avoid the need for such actions for subsequent PBMR applications (and other new nuclear plants), Exelon believes that NRC should initiate rulemaking to revise the regulations to accommodate gas-cooled reactors, modular reactors, and merchant plants In general, these rulemakings should occur as part of a single, integrated rulemaking structured to address licensing of new plants. We believe that such rulemaking would be in accordance with COMJSM-00-003 dated February 13, 2001, in which the Commission stated that the staff should critically assess the existing regulatory infrastructure related to licensing of new plants and identify possible enhancements.

For present purposes, Exelon requests the NRC to review the attached position papers and to meet with Exelon to discuss NRC's preliminary views regarding Exelon's proposals Exelon will need this information in connection with the preparation of the PBMR application if the results of the detailed feasibility study are favorable More importantly, NRC's views related to issues such as financial protection, decommissioning funding, and annual fees may affect the results of the detailed feasibility study itself and Exelon's decision to proceed with the PBMR. Therefore, it is especially important for Exelon to obtain early NRC views on these issues by June Additionally, since NRC's regulations ordinarily require submission of antitrust information nine months prior to an application for a construction permit or combined license, and since collection of the required antitrust information involves substantial effort, Exelon also needs to obtain NRC's views related to the conceptual acceptability of Exelon's proposal on this matter by June.

In particular, in our follow-on discussions in June, Exelon would like to explore NRC's views on the following questions:

- If Exelon provides the information and justifications discussed in the attached position papers, is Exelon's proposal conceptually acceptable to the NRC?
- In addition to the information and justifications discussed in the papers, is there any other information or justifications that NRC would need to accept Exelon's proposal?

Thank you for your consideration and assistance in connection with PBMR matters We look forward to working with you to address and resolve these important regulatory issues related to the PBMR.

Sincerely,

James A. Muntz

James A Muntz
Vice President, Nuclear Projects

cc: William Travers, EDO
Samuel Collins, Director NRR
Ashok Thadani, Director RES
William Borchardt, Associated Director NRR
Richard Barrett, NRR
Janice Moore, OGC
Stuart Rubin, RES
Amy Cubbage, NRR
Diane Jackson, NRR

SUMMARY OF EXELON'S PROPOSALS

Regulation	**Exelon's Proposal**	***Exemption Request* for First Application?**	**Rulemaking Recommended For Subsequent Applications?**
Requirement that nuclear reactors have a license under 10 CFR § 50 10	Issue a single license for a facility with multiple modules	No, unless NRC believes that it cannot issue a single license for multiple modules	Yes
Financial protection requirements in 10 CFR Part 140	Treat multiple modules at a site as a single nuclear facility for purposes of financial protection	Yes, unless NRC considers a modular facility to be a single nuclear reactor	Yes, to clarify that a modular facility may be treated as a single nuclear reactor.
Decommissioning funding requirements in 10 CFR § 50 75(e)	Allow use of partial pre-payment with a 20-year external sinking fund for decommissioning [1]	No, unless NRC finds that partial prepayment with an external sinking fund does not satisfy 10 CFR § 50 75(e)(vi)	Yes, to identify additional acceptable funding methods, such as partial prepayment with an external sinking fund
Requirements for an antitrust review under 10 CFR § 50 33a	Create a class of merchant plants exempt from antitrust review	No, unless NRC does not create a class of merchant plants excepted from antitrust review under Section 105(c)(7) of the Atomic Energy Act by the end of 2001	Yes, to confirm that a merchant plant meeting certain criteria is not required to submit antitrust information or undergo an antitrust review
Requirements to pay annual fees in 10 CFR Part 171	Treat multiple modules at a site as a single facility for purposes of annual fees	No, given the lead time until operation of the first PBMR, it should be possible to resolve this issue by rulemaking prior to operation	Yes
Operator staffing requirements in 10 CFR § 50 54(m)	Establish operator staffing requirements specifically for the PBMR	Yes	Yes, at least as part of the design certification rule for the PBMR
Decommissioning costs under 10 CFR § 50 75(c)	Provide an estimate of the decommissioning costs for a PBMR module	No	No

1 This is one alternative that Exelon is currently evaluating There may be other alternatives that are also acceptable or even preferable

Regulation	Exelon's Proposal	*Exemption Request for First Application?*	**Rulemaking Recommended For Subsequent Applications?**
Environmental impacts of fuel cycle under 10 CFR Part 51	Identify the specific impacts of the fuel cycle and transportation attributable to the PBMR in a manner analogous to 10 CFR §§ 51 51 and 51 52 for LWRs	No Currently, these sections are only applicable to light water reactors	Yes, at least as part of design certification rulemaking for the PBMR
	10 CFR § 51.23 is applicable to and resolves the Waste Confidence issue for the PBMR	No	No
Financial qualifications under 10 CFR § 50 33(f)	Provide information on financial qualifications	No	Yes, to identify criteria that, if satisfied, would establish the financial qualifications of an applicant for a merchant plant.

Note This table only addresses resolution of issues through exemptions and rulemaking It does not discuss possible statutory changes that could resolve some of these issues

NUMBER OF LICENSES AS APPLICABLE TO A PEBBLE BED MODULAR REACTOR (PBMR) FACILITY

I. ISSUE:

The Atomic Energy Act (AEA) contains a number of provisions related to issuance of licenses for reactors:

- Section 101 of the AEA and 10 CFR § 50.10(a) prohibit a person from possessing or using a "utilization facility" except as authorized by a license issued by the Commission. The Commission's regulations in 10 CFR § 50.2 define "utilization facility" as a nuclear reactor. If each PBMR module is treated as a separate nuclear reactor, each individual module could require a separate license.
- Section 161(h) of the AEA and 10 CFR § 50.52 grant the Commission the authority to "combine in a single license" activities that would typically be licensed separately. This paper discusses how these various regulations should be reconciled for a PBMR facility consisting of multiple modules.

II. EXELON'S PROPOSAL:

1) In the first PBMR license application, Exelon will apply for a single license for multiple PBMR modules.

2) Independently of the PBMR licensing proceeding, the NRC should initiate rulemaking to clarify that a set of modules may be treated as a single nuclear facility for licensing and other purposes.

III. ANALYSIS:

Section 101 of the AEA requires a person to obtain a license to possess or use a "utilization facility." Section 11(cc) of the AEA defines the term "utilization facility" as any equipment or device capable of making use of special nuclear material or peculiarly adapted for making use of atomic energy in such quantity as to be of significance to the common defense and security or health and safety of the public. This definition is broad, and could be interpreted as including a set of integrated modules.

10 CFR § 50.2 is more specific, and defines "utilization facility" as "any nuclear reactor." A "nuclear reactor" is defined by 10 CFR § 50.2 as "an apparatus, other than an atomic weapon, designed or used to sustain nuclear fission in a self-supporting chain reaction." Under this section, each module could be classified as a "nuclear reactor."

Neither Section 101 of the Atomic Energy Act nor the corresponding provisions in 10 CFR § 50.10(a) requires that each utilization facility have a separate license -- instead, both the Act and the regulation make it unlawful for a person to possess or use a utilization facility except as authorized by a license issued by the Commission. Therefore, the Commission could, consistently with the language of both Section 101 of the Act and Section 50.10 of the regulations, issue a single license for multiple modules.[1]

Furthermore, Section 161(h) of the AEA states that the Commission may consider in a single application one or more activities for which a license is required. Additionally, 10 CFR § 50.31 states that an applicant may combine several applications for different licenses into one application. This provision has often been used to submit a single application for construction permits or operating licenses for multiple reactors at a single site. Therefore, existing regulations permit Exelon to file a single application for multiple modules at a site.

Additionally, Section 161(h) of the AEA and 10 CFR § 50.52 state that the Commission may combine in a single license the activities of an application which would otherwise be licensed separately. These provisions are typically used to combine licenses for radioactive materials issued under 10 CFR Parts 30, 40, and 70 with an operating license for a single reactor issued under Part 50. However, nothing in the language or legislative history of the AEA or the Commission's regulations would preclude the Commission from combining two or more Part 50 licenses for multiple modules into a single license.

Exelon believes that issuing a single license for multiple PBMR modules would have several beneficial effects. First, issuance of a single license for multiple modules would enable the modules to be treated legally, as well as practically, as a single nuclear facility. As discussed in Exelon's position paper on "Financial Protection Requirements Under the Price-Anderson Act and 10 CFR Part 140 as Applicable to a Pebble Bed Modular Facility," the requirements imposed by Part 140 would be prohibitively burdensome, if applied to each module rather than to a PBMR facility as a whole. Additionally, as discussed in other papers,

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1 We could find nothing in the legislative history of the Atomic Energy Act that directly discusses whether a single license may be issued for more than one reactor, or whether more than one

requirements on annual fees in Part 171 and operator staffing in 10 CFR § 50.54(m) would be unduly burdensome if applied to each module. These problems would be ameliorated if multiple modules were subject to a single license. Furthermore, issuance of a single license for a facility consisting of multiple modules would have other benefits, such as administrative efficiency and promotion of standardization among the modules.

It is important to note that 10 CFR Part 52 appears to contemplate issuance of a single license for multiple modules. In particular, 10 CFR § 52.103(g) states:

> Prior to operation of the facility, the Commission shall find that the acceptance criteria in the combined license [COL] are met. *If the combined license is for a modular design*, each reactor module may require a separate finding as construction proceeds. (Emphasis added)

Under this provision, a single COL could be issued for multiple modules prior to commencement of construction, and the Commission would make a separate pre-operational finding for each module or set of modules as its construction is completed.[2]

Therefore, Exelon believes that NRC may issue a single license for multiple modules given the existing language in the Atomic Energy Act and the Commission's regulations. To avoid uncertainty for future license applications for modular reactor facilities, NRC should initiate rulemaking to expand the

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reactor may be treated as a single utilization facility.

2 The licensing of a modular facility under Part 50 could be more complex due to the two- step licensing process. The Commission could issue a single construction permit for multiple modules. Upon completion of the construction of the first module or first set of modules (and any requisite hearings), the Commission could issue an operating license (OL) for all of the modules; however, pending completion of construction of the other modules, the OL would only authorize operation of the first module or first set of modules. As construction of each additional module or set of modules is completed, the NRC would provide an opportunity for hearing, make the requisite finding under 10 CFR § 50.57(a)(1), and amend the OL to authorize operation of the module or set of modules in question.

definitions of utilization facility and nuclear reactor in 10 CFR § 50.2 to include multiple modular reactors at a site. For the purpose of the definitions, Exelon suggests that the total size of a modular reactor facility be limited to no more than 1500 MWe (which would bound the size of a PBMR facility, which is expected to consist of up to 10 modules each with a rated capacity of between 100 and 150 MWe).

FINANCIAL PROTECTION REQUIREMENTS UNDER THE PRICE-ANDERSON ACT AND 10 CFR PART 140 AS APPLICABLE TO A PEBBLE BED MODULAR REACTOR (PBMR) FACILITY

I. ISSUE:

The Price-Anderson Act imposes certain financial protection requirements on each licensee of a nuclear "facility," which includes a maximum retrospective premium of almost $90 million in the event of a nuclear incident involving a nuclear plant in the United States. NRC's implementing regulations impose these requirements on each "nuclear reactor," so that a licensee would be liable for a maximum retrospective premium of nearly $90 million *per reactor*. 10 CFR § 140.11. If NRC were to impose this requirement on each module, a 10-module PBMR nuclear facility would have a potential liability of almost $900 million. This amount is greatly disproportionate to the potential liability for other reactor facilities of similar size, and runs counter to the intent of the Act in spreading the risk of liability across the industry.

II. EXELON'S PROPOSAL:

1) For the first PBMR application, Exelon will request an exemption from the requirements of 10 CFR § 140.11. Exelon will request that NRC treat a 10-module PBMR facility as one nuclear "facility" within the meaning of the Price-Anderson Act.

2) Independently of the licensing of the PBMR, the NRC should initiate rulemaking to provide that a multiple module facility is a single "facility" under the Price-Anderson financial protection requirements.

III. ANALYSIS:

A. Potential Liability of a PBMR under 10 CFR Part 140

The Price-Anderson Act is included in Section 170 of the Atomic Energy Act (AEA), 42 U.S.C. § 2210. It contains a comprehensive statutory scheme intended to: (1) protect the public against losses from personal injury or property damage arising out of nuclear incidents involving the design, construction, operation or maintenance of nuclear facilities, or the handling or use of nuclear materials; and (2) encourage the development of the nuclear industry by limiting the total liability arising out of any nuclear incident and protecting and indemnifying any person, or entity, who might otherwise be liable, against personal liability in this area by spreading the risk of liability about the industry.

Under Section 170(b) of the Act, the amount of primary financial protection required for facilities designed for producing substantial amounts of electricity and having a rated capacity of 100,000 electric kilowatts [100 MWe] or more must be equal to the maximum amount of commercially available nuclear liability insurance. 42 U.S.C. § 2210(b). This amount is currently $200 million. In addition to this primary financial protection, Section 170(b) requires licensees of such facilities to participate in an industry retrospective rating plan, or secondary layer of protection. This secondary protection provides for the assessment of additional deferred premiums in the event that the public liability from a nuclear incident exceeds the primary financial protection required of the licensee involved in the incident. *Id*

At the present time, the total amount of financial protection available under the Act from both the primary and secondary layers is about $9.7 billion, as follows: (1) the primary layer of $200 million; and (2) a secondary layer of approximately $9.5 billion, based upon a maximum retrospective premium of $88.095 million per nuclear incident per nuclear facility. Under Section 170(b) of the AEA, the maximum amount of the standard deferred premium that may be charged per year to a licensee is $10 million for each facility for which [the] licensee is required to maintain the maximum amount of primary financial protection.

10 CFR § 140.11 requires that financial protection be provided for each *nuclear reactor*. This requirement has significant implications for modular facilities such as the PBMR. If a multiple module PBMR facility is not treated as a single licensed nuclear "facility" for purposes of Price-Anderson, Exelon's potential liability in the event of a nuclear incident at another plant would be multiplied by the number of modules at a site. For example, if the maximum retrospective premium charge of $88.095 million were applied on a per module basis, a ten-module facility would be subject to additional retrospective assessments of more than $880 million for each PBMR facility, for each nuclear incident at another plant. Neither Exelon nor its lenders would find this acceptable. Without relief, 10 ten-module facilities would assume secondary financial liability roughly equal to the entire financial protection that is available under Price-Anderson today. This result would be contrary to the intent of the Price-Anderson Act in spreading the risk of liability across the industry.

B. Legal Authority of the Commission to Treat Multiple Modules as a Single Facility for Purposes of the Price-Anderson Act

The imposition of such disproportionate liability on a PBMR facility is not required by the Price-Anderson Act. Under the Act, the NRC has the authority to treat multiple modules at a site as a single nuclear facility.

Although 10 CFR § 140.11 imposes financial protection requirements on each "nuclear reactor," the Price-Anderson Act is not so restrictive. Section 170(a) of the AEA requires each "license" to have a condition requiring the "licensee" to maintain financial protection. Section 170(b) of the AEA requires each "licensee" to have primary financial protection for "facilities" and to have a secondary layer of financial protection "for facilities designed for producing substantial amounts of electricity and having a rated capacity of 100,000 electrical kilowatts or more."

Thus, Section 170 of the AEA and 10 CFR § 140.11(a)(4) contain similar provisions, except that the Act pertains to "licenses," "licensees," and "facilities," while the Commission's regulations pertain to "nuclear reactors." As discussed below, the rulemaking history of 10 CFR § 140.11 and the legislative history of the Price-Anderson Act do not suggest that each nuclear reactor must be treated as a single licensed nuclear "facility" under the Price-Anderson Act.

1. Rulemaking History

Nowhere in the rulemaking history of 10 CFR Part 140 is there any suggestion that each nuclear reactor must be treated as a single licensed nuclear facility under the Price-Anderson Act. *See generally* Financial Protection Requirements and Indemnity Agreements, 26 Fed. Reg. 2944 (to be codified at 10 CFR Part 140) (Apr. 7, 1960); 24 Fed. Reg. 3508 (proposed May 1, 1959); 25 Fed. Reg. 6681 (proposed Aug. 28, 1958); 24 Fed. Reg. 7223 (proposed Sept. 11, 1957).[1]

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[1] Both 10 CFR § 50.2 and § 140.2 define "nuclear reactor" narrowly as any apparatus used to sustain nuclear fission in a self-supporting chain reaction. If the Commission had intended the term "nuclear reactor" (with such a narrow defini tion) to represent the only interpretation of such a broad term as "facility" as used in the Act, the Commission would presumably have discussed this matter in these Federal Register notices. Because the Commission did not do so, its use of the term "nuclear reactor" in the regulations presumably represents an exercise of the Commission's rulemaking discretion rather than a statutory interpretation of the term "facility."

To the contrary, the Commission has treated an entire site (rather than each reactor on the site) as a single facility for some purposes under the Price-Anderson Act. For example, 10 CFR § 140.11(b) states that primary financial protection [i]n any case where a person is authorized pursuant to part 50 of this chapter to operate two or more nuclear reactors at the same location must only be in the amount of the highest amount which would otherwise be required for any of those reactors: *Provided*, That such primary financial protection covers all reactors at the location. The Commission originally adopted this provision requiring only one primary policy for each site because the insurance syndicates have advised that the nuclear energy liability policies which they are planning to issue will cover nuclear hazards arising out of the possession, disposal, or use of special nuclear material at a described location. 24 Fed. Reg. at 3510.

Thus, the rulemaking history of the NRC regulations implementing the Act suggests that a PBMR with multiple modules on a single site could be treated as a single nuclear facility under the Price-Anderson Act.

2. Legislative History

The legislative history of the Act supports the conclusion that the Commission is free to interpret multiple modules as a single nuclear "facility" under the Price-Anderson Act. The term "facility" as used in Section 170 is not defined. Therefore, the Commission has discretion in providing its own definition, consistent with the intent of the Act.

Furthermore, even if the term "facility" were interpreted as meaning "utilization facility," the definition of "utilization facility" in the AEA is sufficiently broad to allow the

Commission to treat multiple modules as a single "utilization facility." Section 11(cc) of the AEA defines that term as follows:

> any equipment or device except an atomic weapon, determined by rule by the Commission to be capable of making use of special nuclear material in such quantity as to be of significance to the common defense and security, or in such manner as to affect the health and safety of the public, or peculiarly adapted for making use of atomic energy in such quantity as to be of significance to the common defense and security, or in such manner as to affect the health and safety of the public

42 U.S.C. 11(cc). There is nothing in this language that would prevent the Commission from treating multiple modules as a single utilization facility. Furthermore, there is nothing in the legislative history that would prevent the Commission from treating multiple modules or reactors as a single utilization facility. [2]

* * *

In conclusion, a careful reading of the legislative and rulemaking history in this area demonstrates that there is no legal or statutory barrier to the NRC amending or clarifying Part 140 to treat multiple PMBR modules as a single PBMR nuclear "facility" for purposes of the Price-Anderson Act.

C. Appropriate Treatment of the PBMR under the Price-Anderson Act

For the first PBMR application, NRC should grant an exemption from 10 CFR § 140.11, so that the PBMR facility is treated similarly to an equivalent sized light water reactor (LWR). In particular, Exelon's potential liability for retrospective premiums *in the*

[2] During the drafting and debates concerning the Price -Anderson Act and the subsequent amendments to the Act that created the secondary layer of protection, the words "reactor" and "facility" were sometimes used interchangeably. *See, e.g.*, 103 Cong. Rec. 10711 (daily ed. Jul. 1, 1957 (statement of Rep. Price); *Hearings Before the Joint Committee on Atomic Energy*, 84th Cong. 109 (1956) (statement of Charles H. Weaver, Vice -President of Westinghouse Electric Corp); S. Rep. No. 85-296 (1957), *reprinted in* 1957 U.S.C.C.A.N. 1803; H.R. Rep. No. 85-435, at 20 (1957); S. Rep. No. 94-454 (1975), p. 9, *reprinted in* 1975 U.S.C.C.A.A.N. 2251, 2259. However, since a reactor is undoubtedly a utilization facility, and since the concept of modular

event of an accident at another plant should not be substantially higher than the liability of an equivalent sized LWR, merely because Exelon is using a modular design rather than a LWR design. As Exelon will show in its license application, the risks of a severe accident at a 10-module PBMR facility are less than the risks of a severe accident at a LWR (and therefore the risk that another nuclear plant will incur retrospective liability under the Price-Anderson Act *as a result of an accident at the PBMR facility* is less than the risk of such liability from an accident at a LWR). Exelon's application for the first PBMR application will provide additional support for such an exemption, including providing a technical justification for the exemption based upon a comparison of the risks of a PBMR facility and an LWR.

Given the flexibility provided by the Price-Anderson Act and the AEA in general, Exelon believes that NRC has the authority to grant an exemption from 10 CFR § 140.11 for the first PBMR application, and to treat multiple modules at a site as a single nuclear facility with a single license for purposes of the Price-Anderson Act (or otherwise limit the potential liability of the PBMR).

As a long term solution to this matter, NRC should initiate rulemaking to amend Section 140.11(a)(4) to state explicitly that the financial protection requirements apply to each licensee for a nuclear "facility," and that a nuclear facility may include multiple modules at a site. The definitions of utilization facility and nuclear reactor in 10 CFR § 50.2 should also be amended to include multiple reactor modules at a site. Exelon is working with the Nuclear Energy Institute to provide supporting information and justification for such rulemaking.

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reactors had not yet been developed, the interchangeable use of these terms is not particularly surprising and does not preclude multiple reactors from being treated as a single facility.

In proposing such a change in the regulations, Exelon realizes that it may be appropriate to limit the number and size of modules that may be treated as a single nuclear facility. Exelon suggests that the total size of each modular nuclear reactor facility subject to the Price-Anderson financial protection requirements be limited to no more than 1500 MWe (which would bound a 10-module PBMR facility). Such a limit provides a reasonable basis for rulemaking, by placing a modular nuclear facility on an equivalent footing with a current LWR, for purposes of the Price-Anderson Act.

DECOMMISSIONING FUNDING FOR A PEBBLE BED MODULAR REACTOR (PBMR) FACILITY

I. ISSUE:

10 CFR § 50.75 requires licensees to establish financial assurance for decommissioning. Section 50.75(e)(1) provides six methods for providing financial assurance. These methods include prepayment, an external sinking fund, surety, insurance, or other "equivalent" method. However, Section 50.75(e)(1) essentially restricts use of external sinking funds to licensees that recover decommissioning funds through rates or a non-bypassable charge. Most other licensees have used the prepayment method (e.g., licensees in license transfer proceedings).

This paper evaluates the implications of these requirements for the PBMR.

II. EXELON'S PROPOSAL:

1) The first PBMR license application will propose a decommissioning funding method for the PBMR. Exelon has not yet selected a decommissioning funding method. However, Exelon is evaluating the possibility of seeking NRC approval for an alternative decommissioning funding mechanism that provides for partial prepayment of the total decommissioning cost estimate and annual contributions for the remainder spread over 20 years. Exelon believes that such a mechanism would be permissible under Section 50.75(e)(1)(vi) as an "equivalent" method (or, at the very least, would qualify for an exemption under 10 CFR § 50.12).

2) NRC should initiate rulemaking to modify Section 50.75(e)(1) to explicitly authorize the use of this alternative funding mechanism for new plants. This rulemaking should be initiated independently of the licensing proceeding for the PBMR, and should

also address other alternative decommissioning funding methods being developed by the industry.

III. <u>ANALYSIS</u>:

10 CFR § 50.75(e)(1) states that financial assurance for decommissioning is to be provided by one or more of the following methods: (i) prepayment in the form of a trust, escrow account, government fund, certificate of deposit, or other payment acceptable to the NRC,

(ii) external sinking fund for a licensee that recovers the estimated cost of decommissioning through "cost of service" rates or non-bypassable charge for decommissioning costs, (iii) surety method, insurance, or other guarantee method, (iv) a statement of intent (for a federal licensee), (v) contractual obligations, and (vi) any other mechanism, or combination of mechanisms, that provides (as determined by the NRC) an assurance mechanism equivalent to the other methods in this section. Since a new PBMR modular facility would likely not recover decommissioning costs through rates or a non-bypassable charge, it would not be allowed to use the external sinking fund method under 10 CFR § 50.75(e)(1)(ii) for the PBMR.

Most license transfers to date involving sales of reactors to unaffiliated third parties have satisfied NRC's decommissioning funding assurance requirements by fully prepaying and conveying those funds to the new licensee at closing. According to the NRC, while prepayment places a significant up-front burden on licensees, prepayment provides assurance that a licensee will be able to meet its decommissioning obligations. However, if NRC were to require 100% prepayment of the decommissioning cost estimate for new plants, such prepayment might jeopardize the economic viability of any the economic viability of any new plant that is to be operated on a merchant basis because of the higher present worth of a prepayment relative to other funding mechanisms which contemplate payment(s) at a later time.

Exelon is giving further consideration to whether some of the other funding arrangements authorized under 10 CFR § 50.75(e) may be feasible for a PBMR operated as a merchant plant by Exelon. For example, Exelon is considering the insurance option pursuant to 10 CFR § 50.75(e)(1)(iii), and long term power sales contracts that provide for the funding of decommissioning costs pursuant to 10 CFR § 50.75(e)(1)(v). Exelon is also considering some funding mechanisms being developed by the industry.

Additionally, Exelon is evaluating the economic feasibility of requiring a new PBMR to accumulate decommissioning funding on an accelerated basis during the first 20 years of operation. Use of such a funding mechanism, in which Exelon would make partial prepayment (5%, for example) of the total decommissioning cost estimate and annual contributions for the remainder spread over 20 years, would substantially reduce the initial costs associated with the PBMR while still providing assurance of funds for decommissioning at the time a module is likely to be decommissioned.

Exelon believes that such a prepayment funding mechanism would provide adequate assurance of decommissioning funding for a new plant. By definition, it will guarantee that sufficient funds are available if a plant operates for its licensed lifetime. Furthermore, partial prepayment, coupled with accelerated funding over the first 20 years of operation, is reasonable in light of the small risk of premature shutdown during that period.

In particular, according to NUREG-1350, NRC has issued more than 120 full power operating licenses for power reactors with a capacity of 100 MWe or greater. Of these, all but nine operated for approximately 20 years or longer (or are currently operating). Of these nine, five operated for more than 12.5 years; two operated for about nine years; one (Pathfinder) operated for about three years; and one (TMI-2) was closed due to an accident. This history indicates that more than 90% of power reactors have operated for approximately 20 years or longer (or are currently operating) and that all but two of the remaining plants have operated for about 9 years or longer. This history provides adequate assurance that the alternative funding method will cover the decommissioning costs at the time of termination of operation.[1]

Exelon believes that this alternative approach satisfies 10 CFR § 50.75(e)(1)(vi) which allows a licensee to provide financial assurance via "[a]ny other mechanism, or combination of mechanisms, that provides, as determined by the NRC upon its evaluation of the specific circumstances of each licensee submittal, assurance of decommissioning funding equivalent to that provided by the [enumerated] mechanisms." If NRC disagrees, however, Exelon believes that NRC could grant an exemption from Section 50.75(e)(1) to permit this alternative funding approach (or select another option).

If Exelon decides to use an alternative funding mechanism, its application for the PBMR will provide more details and a justification for the mechanism. However, if NRC

[1] Exelon recognizes that the NRC considered and rejected an accelerated funding mechanism when it revised the decommissioning funding rule in 1998. However, NRC rejected such an approach *for existing operating reactors*, many which have operated for well over twenty years. As NRC noted, an accelerated funding mechanism *for existing operating reactors* might not as sure adequate decommissioning at the end of the licensed lifetime, let alone in the event of premature shutdown. Obviously, this rationale is not applicable to newly licensed plants. The NRC did not

is conceptually opposed to use of partial prepayment with accelerated funding over twenty years (either under Section 50.75(e)(1)(vi) or as an exemption), Exelon needs to know as soon as possible so that this can be factored into Exelon's evaluation of the economic feasibility of the PBMR. Additionally, if NRC believes that there may be other acceptable funding mechanisms that can accomplish the same purpose, Exelon is willing to consider the economic feasibility of those methods. To avoid duplicative efforts for future merchant nuclear power plants, the NRC should initiate rulemaking to revise 10 CFR § 50.75(e)(1) and explicitly allow alternative approaches for new plants. Exelon is working with the Nuclear Energy Institute and other nuclear generation companies to identify a number of possible alternative funding methods and develop supporting information for use in rulemaking. This rulemaking should be initiated independently of licensing of the PBMR.

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consider whether use of an accelerated funding approach would be adequate for newly licensed reactors. (63 Fed. Reg. 50461, 50469-70, September 22, 1998).

NRC ANTITRUST REVIEW AUTHORITY AS APPLICABLE TO A PEBBLE BED MODULAR REACTOR (PBMR) FACILITY

I. ISSUE:

Section 105 of the Atomic Energy Act (AEA) requires that the NRC conduct an antitrust review, seek the advice of the Attorney General, and if necessary conduct a hearing on antitrust matters in connection with applications for a construction permit (CP) or combined operating license (COL) for a nuclear power reactor. NRC's implementing regulations in 10 CFR § 50.33a provide that applicants for such licenses are required to submit to the NRC detailed transmission, distribution, and business planning information that will allow the Attorney General of the United States and NRC staff to conduct an antitrust review of the proposed project.

Pursuant to Section 105(c)(7) of the AEA, NRC has the authority, with the approval of the Attorney General, to determine that issuance of certain classes of licenses would not significantly affect the licensees' activities under the antitrust laws, and therefore except such applicants from NRC antitrust review under Section 105. Recognizing the current status of competition in the electric utility industry and the fundamental competitive realities surrounding the operation of any new merchant nuclear project, the NRC should make a determination under Section 105(c)(7) that applicants that will operate their plants as merchant plants are excepted from NRC antitrust review.

II. EXELON'S PROPOSAL:

1) The NRC should initiate a proceeding, and seek the approval of the Attorney General, to determine that the issuance of licenses to merchant plant applicants will not significantly affect such applicants' activities under the antitrust laws. NRC should make a determination pursuant to Section 105(c)(7) that merchant plant applicants are excepted from antitrust review. Any such determination should also provide appropriate criteria for determining whether an applicant qualifies as a merchant plant operator.

2) The NRC should also initiate a rulemaking to clarify that its rules do not require that a merchant plant applicant submit the antitrust information identified in 10 CFR § 50.33a. The rule should state that an applicant need only provide information sufficient for the NRC to make a determination as to whether the applicant qualifies as a member of the excepted class. This model is consistent with the approach pursued by NRC when it made its determination that it would not conduct antitrust reviews in connection with license transfers. [1]

III. ANALYSIS:

Section 105, the "Antitrust Provisions" of the AEA, requires NRC to conduct an antitrust review in consultation with the Attorney General, prior to issuing a license under Section 103 for a nuclear generating facility. In particular, Section 105 of the AEA requires the NRC to determine whether activities under the license would create or maintain a situation "inconsistent with the antitrust laws." NRC has traditionally exercised this authority by conducting antitrust reviews and, if necessary, hearings. In some instances, these reviews and hearings have resulted in NRC imposing various antitrust conditions in the license. These conditions have often involved access to transmission.

The regulations implementing Section 105 are contained in 10 CFR Part 50. Section 50.33a, "Information requested by the Attorney General for antitrust review," states that nine
months prior to submitting its application, an applicant for a construction permit for a nuclear power reactor shall submit the information requested by the Attorney General as described in Appendix L, if the applicant has more than 200 MWe of generating capacity. Appendix L, Section II, "Required Information," lists 20 separate issues that must be addressed by the applicant in the antitrust submittal.

The antitrust review provisions of Section 105 have limited applicability to the modern electric industry, and they serve no useful purpose with respect to proposed operation of a nuclear reactor on a merchant plant basis. Changes in the electric industry – including the emergence of a competitive wholesale electric market and mandated open access to the transmission system – reduce, if not eliminate, the incremental protection of competition that the NRC provides through its antitrust review for license applications for merchant plants.

Section 105(c)(7) empowers NRC to except a class of licenses from antitrust review "as the Commission may determine would not significantly affect the applicant's

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1 *See Kansas Gas & Electric Co.* (Wolf Creek Generating Station, Unit 1), CLI- 99-19, 49 NRC 441 (1999); Final Rule, "Antitrust Review Authority: Clarification," 65 Fed. Reg. 44,649 (July 19, 2000).

activities under the antitrust laws."[2] NRC should use its existing authority under Section 105(c)(7) to provide an exception from antitrust review for merchant plant applicants that meet certain criteria, *e.g.*, Exempt Wholesale Generators (EWGs) or generators authorized to sell power at wholesale at market based rates. By definition, such merchant plants operate in a competitive environment. Additionally, EWGs do not control transmission systems. Furthermore, Federal Energy Regulatory Commission (FERC) Order 888 obligates transmission providers to file open access transmission tariffs. Additionally, there are a large number of different generating companies owning and operating merchant plants and competing in the generation market, and the construction of new generation (increasing supply) is pro-competitive. Therefore, the licensing of a merchant plant will not create any situation inconsistent with the antitrust laws.

NRC could take action to create an excepted class of licenses by order, policy statement, or rulemaking. Exelon suggests that NRC follow an approach akin to the one it took in *Wolf Creek*, wherein NRC would issue a *Federal Register* notice and solicit public comments regarding whether it should determine that the issuance of licenses to applicants who qualify as merchant plant operators would not significantly affect such applicants' activities under the antitrust laws, and therefore except such applicants from NRC antitrust review under Section 105. Upon issuance of such a

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2 Section 106 of the Atomic Energy Act states the Commission may group facility licenses into classes "upon the basis of similarity of operating and technical characteristics of the facilities," and may define the various activities to be carried out at each class of facility and the amounts of special nuclear material available for use by each facility. There does not appear to be a connection between the term "class of facilities" as used in Section 106 and the term "class or types of licenses" as used in Section 105(c)(7). These sections were enacted more than 15 years apart, and neither Section 105(c)(7) nor its legislative history refers to Section 106. Furthermore, the language in Section 106 (which refers to "operating and technical characteristics") is simply inapposite to the type of antitrust issues addressed in Section 105.

determination, NRC could then initiate a rulemaking to clarify that its rules do not require that a merchant plant applicant submit the antitrust information identified in 10 CFR § 50.33a. The rule should state that such an applicant need only provide information sufficient for the NRC to make a determination as to whether the applicant qualifies as a member of the excepted class.

Exelon has been working with the Nuclear Energy Institute (NEI) to support the creation of the excepted class for merchant plants. We urge NRC to make such a determination prior to the end of this year, pursuant to the authority granted in Section 105(c)(7) of the AEA. If NRC does not reach a decision by the end of this year, Exelon will need to provide the required antitrust information or request an exemption from § 50.33a which will permit Exelon to defer filing of antitrust information until after NRC makes a decision on whether it will except merchant plant operators from antitrust review.

ANNUAL FEES UNDER 10 CFR § 171.15 AS APPLICABLE TO A PEBBLE BED MODULAR REACTOR (PBMR) FACILITY

I. ISSUE:

10 CFR § 171.15(a) states that each person licensed to operate a power reactor shall pay an annual fee "for each unit for each license" held at any time during the Federal fiscal year in which the fee is due. If each PBMR module is treated as a reactor, Section 171.15 could be construed so as to impose a separate fee for each module. Therefore, the annual fee for a 10-module PBMR would be greatly disproportionate to the annual fee for an equivalent sized boiling water reactor (BWR) or pressurized water reactor (PWR). This could place a modular reactor design at a competitive disadvantage with other designs and act as a disadvantage to the development of modular reactors.

II. EXELON'S PROPOSAL:

For the purposes of assessing annual fees, it is not reasonable to treat multiple PBMR modules at a site the same as multiple PWRs or BWRs at a site. NRC should initiate rulemaking to change Section 171.15 to specify that only one annual fee will be required for each facility or set of modular reactors at a given site. This rulemaking on Section 171.15 should be completed prior to issuance of the license for the first PBMR.

III. ANALYSIS:

The requirements set forth in 10 CFR Part 171 originally were promulgated in 1986. *See* 51 Fed. Reg. 24078 (July 1, 1986). The NRC enacted Part 171 to comply with the requirements of the Consolidated Omnibus Budget Reconciliation Act of 1985 which required the NRC to "assess and collect annual charges from persons licensed by the Commission pursuant to the Atomic Energy Act of 1954" in order to recover the Commission's estimated budget costs. *Id.* The NRC consequently promulgated the requirement that "[e]ach person licensed to operate a power reactor shall pay an annual fee for each power reactor unit for which the person holds an operating license" to recoup a portion of its costs. Annual Fee for Nuclear Power Reactor Operating Licenses and Conforming Amendments, 51 Fed. Reg. 33224, 33230 (Sept. 18, 1986).

When discussing the fee schedules, the NRC stated that "[t]he annual charge should be assessed under the principle that licensees who require the greatest expenditures of the agency's resources should pay the greatest annual charges." Revision of Fee Schedules, 56 Fed. Reg. 14870, 14871 (Apr. 12, 1991). *See also* 136 Cong. Rec. H 10107 (Oct. 16, 1990). Although the NRC never stated in the Federal Register why "reactors" were used as the basis for assigning fees, instead of sites or facilities, the NRC commented that "[a]fter examining and analyzing the historical data available, the Commission has determined that the bulk of its licensee-related activities have and will continue to be directly related to the regulation of *large power reactors*." 51 Fed. Reg. at 24084 (emphasis added). Presumably, this statement provides the link between the decision to require fees for each reactor instead of the entire site or facility. In 1986, when this rule was originally considered, the NRC and the industry had no reason to foresee any need to word the rule differently. Almost all commercial nuclear power facilities in existence were large reactors, and a multiple modular facility had not yet been developed or approved.

10 CFR § 171.15(a) states that each person licensed to operate a power reactor shall pay an annual fee "for each unit for each license" held at any time during the Federal fiscal year in which the fee is due. In turn, Section 171.15(b) states that the 2000 Fiscal Year annual fee for "each operating power reactor" is $2,815,000. If each PBMR module is treated as a reactor, Section 171.15 would impose a separate fee for each module. Therefore, the annual fee for a 10-module PBMR would be almost $30,000,000. In contrast, the annual fee for an equivalent sized BWR or PWR would be less than $3,000,000. There is no basis for providing such disparate treatment to a PBMR facility.

For several reasons, NRC resources for regulating a 10-module PBMR facility will be similar to or lower than NRC resources for regulating a large BWR or PWR, and therefore NRC's annual fees for each should be similar. First, the PBMR modules at a site will have a single licensing basis. Second, the PBMR design will be simpler and safer than the design of a PWR or BWR. Finally, a PBMR facility will have a smaller workforce than existing reactors, thereby simplifying NRC's oversight responsibilities.

Furthermore, NRC assesses annual fees to recover its costs that cannot be assigned to any particular facility. *See* 51 Fed. Reg. at 24078. For this purpose, it would be unfair to assess higher fees for multiple modules that have a combined power level equivalent to a single large PWR or BWR. Higher fees would, in essence, penalize Exelon for selecting a modular design rather than a LWR design, and would serve to discourage development of a newer and safer technology.

For all of these reasons, it is reasonable and appropriate to treat multiple PBMR modules at a site as a single facility for purposes of assessing annual fees, and NRC should initiate rulemaking to accomplish this goal. In order to implement this rulemaking, NRC should define the term "modular facility." Exelon suggests that the total size of a modular reactor facility be limited to no more than 1500 MWe (which would bound a PBMR facility, which is currently expected to consist of as many as ten modules each with a rating of between 100 and 150 MWe). Exelon believes that this provides a reasonable basis for defining a modular reactor facility in light of the current state of modular design technology and the size of current large scale PWRs and BWRs.

The previous paragraph provides a conceptual basis for rulemaking to modify Section 171.15. Exelon is working with the Nuclear Energy Institute to provide NRC with more detailed information to support rulemaking on this issue.

Resolution of this issue is not necessary for licensing or design certifications of the PBMR. However, this issue does have a significant impact on the economic feasibility of the PBMR. Therefore, Exelon requests NRC to indicate whether it is conceptually willing to initiate such rulemaking or other alternatives for accomplishing the same object (such as granting an exemption to the PBMR, or creating special annual fee provisions for modular reactors).

OPERATOR STAFFING REQUIREMENTS UNDER 10 CFR § 50.54(m) AS APPLICABLE TO A PEBBLE BED MODULAR REACTOR (PBMR) FACILITY

I. ISSUE:

10 CFR § 50.54(m) specifies minimum licensed operator staffing requirements. However, it does not identify staffing requirements for sites with more than two units with a common control room. Moreover, Section 50.54(m) contains requirements on the location of operators; i.e., it requires that one senior reactor operator (SRO) be in the control room of a unit during operation, that one reactor operator (RO) be at the controls for each unit during operation, and that a SRO be present during fuel handling. If NRC were to treat each PBMR module as a separate unit, the staffing requirements in Section 50.54(m) would be excessive and unnecessary. This paper discusses a process for specifying more reasonable operator staffing requirements.

II. EXELON'S PROPOSAL:

1) The first PBMR license application and the PBMR design certification application will propose and justify licensed operator staffing requirements for three or more PBMR modules at a site with a common control room. Because Section 50.54(m) currently does not contain any requirements for such configurations, approval of such staffing requirements will not require an exemption.

2) For operation involving the first two PBMR modules, the minimum staffing requirements in Section 50.54(m) are probably excessive. Additionally, the requirements in Section 50.54(m) on the location of SROs and ROs would be excessive if applied to a PBMR facility. Therefore, as part of its application, Exelon will request and justify an exemption from these requirements for the PBMR.

3) To avoid duplicative reviews for subsequent PBMR applications, the application for design certification of the PBMR under Part 52 will also specify licensed operator staffing requirements and request an exemption from Section 50.54(m).

III. <u>ANALYSIS</u>:

10 CFR § 50.54(m) identifies minimum staffing requirements for SROs and ROs for various plant modes. These staffing requirements vary, depending upon the number of "units" at a site and whether the units have a common control room. In general, for each shift with all units operating, the number of required ROs is 2U and the number of required SROs is U+1, where U is the number of units (with a decrease of one RO and SRO if there is a common control room). However, Section 50.54(m) does not specify staffing requirements for more than two units at a site with a common control room.

In addition to these requirements, Section 50.54(m) also specifies the following staffing requirements:

- Each licensee shall have at its site a person holding a senior operator license for all fueled units at the site who is assigned responsibility for overall plant operation at all times there is fuel in any unit. If a single senior operator does not hold a senior operator license on all fueled units at the site, then the licensee must have at the site two or more senior operators, who in combination are licensed as senior operators on all fueled units.

- When a nuclear power unit is in an operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times.

- Each licensee shall have present, during alteration of the core of a nuclear power unit (including fuel loading or transfer), a person holding a senior operator license or a senior operator license limited to fuel handling to directly supervise the activity and, during this time, the licensee shall not assign other duties to this person.

In general, the formula used to develop the staffing levels in Section 50.54(m), and the requirements on the location of operators in Section 50.54(m), are excessive for PBMRs. These staffing requirements were developed when all operating nuclear power plants relied on active safety systems to mitigate accidents. Since the PBMR is a passive plant that does not require early operator intervention to mitigate accidents, staffing levels less than those indicated in Section 50.54(m) are appropriate for the PBMR.

As the Commission recognized when it promulgated Section 50.54(m) in the aftermath of the Three Mile Island incident, an exemption from the staffing requirements may be warranted to provide for "reduced staffing levels based on plant size, lack of complexity, or other unique factors." 48 Fed. Reg. 31611 (July 11, 1983). The first PBMR license application and design certification application will justify a reduced staffing level.

Section 50.54(m) currently does not contain any staffing requirements for more than two units at a site with a common control room. Therefore, no exemption will be needed to specify minimum staffing requirements for operation of three or more modules with a common control room. In contrast, Section 50.54(m) provides minimum staffing requirements applicable to two units with a common control room and contains requirements regarding the location of ROs and SROs. If a module is treated as a "unit," an exemption from these requirements will be needed to provide for lower staffing. Such an exemption will be requested as part of the application for the license for the first PBMR facility and the design certification rule for the PBMR.

5/10/01

DECOMMISSIONING COST ESTIMATE FOR A PEBBLE BED MODULAR REACTOR (PBMR) FACILITY

I. ISSUE:

10 CFR § 50.75(c) specifies a minimum amount for the decommissioning fund for boiling water reactors (BWRs) and pressurized water reactors (PWRs). However, this section does not specify a minimum amount for the required decommissioning fund for a gas cooled reactor.

II. EXELON'S PROPOSAL:

The first PBMR license application will include a cost estimate for decommissioning a PBMR module.

III. ANALYSIS:

10 CFR § 50.75(c) specifies a minimum amount for the decommissioning fund for BWRs and PWRs but not for a gas cooled reactor. Because the design of the PBMR is significantly different than the design of a BWR or PWR, neither of the cost estimates currently in Section 50.75(c) is appropriate for a PBMR module.

Therefore, the license application for the PBMR will include a decommissioning cost estimate. Because construction of the PBMR modules at a site will most likely be staggered, and because the PBMR modules might be decommissioned at different times, the cost estimate will apply to decommissioning of a single PBMR module.

CONSIDERATION OF THE ENVIRONMENTAL IMPACTS OF THE FUEL CYCLE AND TRANSPORTATION AS APPLICABLE TO A PEBBLE BED MODULAR REACTOR (PBMR) FACILITY

I. ISSUE:

10 CFR §§ 51.51 and 51.52 (Tables S-3 and S-4) specify the environmental impacts attributable to the fuel cycle and transportation for light water reactors (LWRs) but not for other types of reactors. As a result, this issue is unresolved for the PBMR.

Additionally, 10 CFR § 51.23 resolves issues related to the environmental impacts of storage of spent fuel following cessation of reactor operation until a mined geologic repository is available to dispose of the spent fuel (the "waste confidence" rule). This paper addresses whether such resolution applies to spent fuel generated by a PBMR.[1]

II. EXELON'S PROPOSAL:

1) In the first PBMR application, Exelon will identify the environmental impacts attributable to the fuel cycle and transportation for a PBMR facility.

2) Based upon the resolution of these issues for the first PBMR application, NRC should initiate rulemaking to create tables for the PBMR that are similar to Tables S-3 and S-4.

3) Long term onsite storage of spent fuel beyond the licensed lifetime of the PBMR is not a concern under the NRC's Waste Confidence Rule in 10 CFR § 51.23.

1 It is expected that the PBMR will use 8% enriched Uranium-235 fuel, which is classified as low enriched uranium (LEU) fuel under 10 CFR § 50.2. The only regulation that imposes more restrictive requirements on 8% enriched fuel than on the 4% enriched fuel typically used in

LWRs is 10 CFR § 50.68(b), which requires a criticality monitoring system for use, handling, and storage of fuel assemblies with an enrichment greater than 5%.

III. ANALYSIS:

A. Tables S-3 and S-4

Since Tables S-3 and S-4 in 10 CFR §§ 51.51 and 51.52 are limited to LWRs, issues related to the environmental impacts attributable to the fuel cycle and transportation have not been resolved by rulemaking for other types of reactors.

As a result, as part of the first PBMR application, Exelon will provide information on the environmental impacts of the fuel cycle and transportation attributable to a PBMR facility.

Once this issue has been resolved for the first PBMR application, NRC should initiate rulemaking to eliminate the need for duplicative reviews of this same information for subsequent PBMR applications. Since these impacts are generic for all PBMR facilities (and any comparable facilities), the results of the evaluation of these impacts for the first PBMR application should serve as the basis for the rulemaking. This rulemaking could entail the addition of tables to Part 51 similar to the existing Tables S-3 and S-4, or this issue could be resolved as part of the design certification rulemaking for the PBMR.

B. Waste Confidence Rule

In the early 1980s, the NRC conducted a generic rulemaking to assess the degree of assurance that radioactive wastes could be disposed of safely, to determine whether disposal or offsite storage would be available, and to determine whether the waste could be stored safely at reactor sites beyond the expiration of existing facility licenses until offsite disposal or storage is available.

The rulemaking came to be known as the "Waste Confidence" proceeding. On August 31, 1984, the NRC published five findings, accompanied by a final rule (codified at 10 CFR § 51.23) that incorporated the findings as the basis for excluding case-by-case consideration of environmental effects of extended onsite storage of spent fuel in reactor and spent fuel storage facility licensing proceedings. *See* 49 *Fed. Reg.* 34658, 34688. The NRC's Waste Confidence Rule, as revised,[2] states that:

> The Commission has made a generic determination that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of that reactor at its spent fuel storage basin or at either onsite or offsite independent spent fuel storage installations. Further, the Commission believes there is reasonable assurance that at least one mined geologic repository will be available within the first quarter of the twenty-first century, and sufficient repository capacity will be available within 30 years beyond the licensed life for operation of any reactor to dispose of the commercial high-level waste and spent fuel originating in such reactor and generated up to that time.

10 CFR § 51.23(a) (emphasis added). This provision does not distinguish between types of spent fuel.[3] Additionally, in making its findings in support of the Waste Confidence Rule, the Commission explicitly considered non-LWR fuel, including fuel from gas cooled reactors. *See, e.g.,* 49 Fed. Reg. at 34663 and 34683. Accordingly, the Waste Confidence Rule is broad enough to cover fuel irradiated in a gas-cooled reactor like the PBMR.

Furthermore, as a practical matter, there should be a repository available long before the end of the licensed lifetime of the PBMR. The Waste Confidence Rule states

2 The NRC recently reaffirmed its decision in the Waste Confidence Rulemaking, finding that there have been "no major shifts in national policy, no major unexpected institutional developments, [and] no unexpected technical information . . . that would cast doubt on the Commission's Waste Confidence findings . . . ." 64 *Fed. Reg.* 68005, 68007 (Dec. 6. 1999).

3 Part 51 does not define "spent fuel." The closest definition is "spent nuclear fuel" in 10 CFR Part 2, Subpart K, governing hearing procedures for expansion of spent nuclear fuel storage capacities. *See* 10 CFR 2.1105. That definition states that spent nuclear fuel means "fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing." *Id.* This definition also does not distinguish between the

that there is reasonable assurance that a repository will be available by the first quarter of the twenty-first century (i.e., by 2025). In contrast, Exelon does not expect that the first PBMR will begin operation in the United States until 2006. Given a 40-year licensed lifetime for the PBMR, the license for the first PBMR would not expire until 2046 at the earliest - - long after the repository is expected to be available.

Under the Nuclear Waste Policy Act (NWPA), 42 U.S.C. §§ 10101 *et seq.*, the Department of Energy (DOE) will be required to accept irradiated PBMR fuel. The NWPA makes the federal government responsible for permanent disposal of spent nuclear fuel. 42 U.S.C. § 10131(4). To carry out this responsibility, the NWPA authorizes the Secretary of the DOE to enter into contracts with any person who generates, among other things, "spent nuclear fuel." 42 U.S.C. § 10222(a)(1). For civilian nuclear power plants, these contracts provide payment of fees in exchange for DOE's "acceptance of title, subsequent transportation, and disposal of . . . spent fuel." *Id.*

Nothing in the NWPA excludes irradiated PBMR fuel. The federal government's obligation applies to "spent nuclear fuel," which is defined as "fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing." *See* 42 U.S.C. §§ 10102(23), 10222. Also, the contract mechanism which applies to civilian nuclear power reactors would include any "power plant required to be licensed as a utilization facility under section 103 or 104(b) of the Atomic Energy Act of 1954." *See* 42 U.S.C. § 10102(6). Since the irradiated PBMR fuel meets the definition of "spent nuclear fuel," and the PBMR itself

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type of fuel generated in the reactor. Accordingly, under NRC regulations, the type of fuel generated – whether LWR or PBMR fuel – should not affect the Waste Confidence Rule.

will be licensed pursuant to Section 103 of the Act, DOE would be required to enter into a contract with Exelon for the ultimate disposal of the irradiated PBMR fuel.

The DOE's regulations implementing the contract requirement explicitly support this position. DOE regulations in 10 CFR § 961.1 clarify that DOE "will make available nuclear waste services to the owners and generators of spent nuclear fuel," and that "DOE will take title to, transport, and dispose of spent nuclear fuel . . . delivered to DOE by those owners of generators who execute the contract" set forth in 10 CFR § 961.11. This contract explicitly states that:

> the DOE has the responsibility following commencement of operation of a repository to take title to the spent nuclear fuel [SNF] or high-level radioactive waste [HLW] involved as expeditiously as practicable upon the request of the generator or owner of such waste or spent nuclear fuel.

Furthermore, Article I.18 of the contract states that the contract "applies to the delivery by Purchaser to DOE of SNF and/or HLW of domestic origin from civilian nuclear power reactors." Finally, Appendix E.4 of the contract explicitly states that such fuel includes "non-LWR fuel" (which is classified as nonstandard fuel under the contract). Thus, the standard DOE contract explicitly encompasses non-LWR fuel such as PBMR fuel, and DOE is required to accept such fuel from licensees who execute DOE's standard contract.

In summary, the Nuclear Waste Policy Act requires DOE to take title to and dispose of spent PBMR fuel. Since NRC expects the DOE repository to be in operation by the time the license for the first PBMR facility expires, long term storage of spent fuel from a PBMR does not represent a concern under the NRC's Waste Confidence Rule.

FINANCIAL QUALIFICATIONS FOR A LICENSE FOR A PEBBLE BED MODULAR REACTOR (PBMR) FACILITY

I. ISSUE:

10 CFR § 50.33(f) requires an applicant for a license to provide information on its financial qualifications, and Appendix C to Part 50 identifies the type of financial qualifications information that should be submitted. "Electric utilities" are excepted from the requirement to submit financial qualifications information. Exelon Generation is not an electric utility as defined in 10 CFR § 50.2 and therefore will be subject to the requirement to submit detailed financial qualifications information under Section 50.33(f). This requirement is burdensome and is unwarranted for applicants that have assets or parental guarantees.

II. EXELON'S POSITION:

1) For the first PBMR application, Exelon will submit the information required by Appendix C to Part 50. Exelon will submit estimates for total construction costs and total annual operating costs for each of the first five years of operation of the entire PBMR facility and the source of funds to cover such operating costs.

2) To avoid duplicative reviews for subsequent applications, the NRC should initiate rulemaking to revise its financial qualifications regulations to enable certain categories of merchant generating companies to have the same status as utilities. In particular, Section 50.33(f) should be revised to state that an applicant is financially qualified if it satisfies certain criteria.

III. ANALYSIS:

Section 182(a) of the AEA requires license applications to include such information on the financial qualifications of the applicant as the Commission may specify by regulation. The NRC's regulations governing financial qualification reviews for licenses to construct or operate nuclear power plants are contained in 10 CFR § 50.33(f).

For a non-electric utility to establish its financial qualifications, Appendix C to 10 CFR Part 50 requires the applicant for a construction permit provide at least three types of information: (1) an estimate of construction costs, (2) source of construction funds, and (3) the latest published annual financial reports, together with any current interim financial statements. An applicant for an operating license must submit information "that demonstrates the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated operation costs for the period of the license." *See* 10 CFR 50.33(f)(2). While the explicit terms of this regulation address "costs for the period of the license," in practice, this means that applicants must submit estimates for total construction costs and total annual operating costs for each of the first five years of operation of the facility and the source of funds to cover such operating costs. This could include projections of the market price of power, long-term contracts, and corporate revenues from other sources that may be used at the nuclear plants.

If the applicant is a newly-formed entity, Appendix C requires that additional financial information be submitted including: (1) the legal and financial relationships with stockholders, corporate affiliates, and others upon which the applicant is relying for financial assistance, (2) information to support the financial capability of parent companies and corporate affiliates to meet their financial commitments, and (3) the applicant's statements of assets, liabilities, and capital structure as of the date of the application.

Exelon will supply the required information for the first PBMR application. However, the NRC should initiate rulemaking to establish specific criteria that would enable non-utilities to demonstrate financial qualifications without providing the detailed information currently required by NRC regulations and guidance. Exelon will work with the Nuclear Energy Institute to develop such criteria and provide more detailed information to support this rulemaking. This rulemaking should proceed independently of the licensing of the first PBMR.