PDR

UNITED STATES NUCLEAR REGULATORY COMMISSION

POLICY ISSUE
(Notation Vote)

July 15, 1988 SECY-88-202

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: STANDARDIZATION OF ADVANCED REACTOR DESIGNS

Purpose: To inform the Commission of the Department of Energy's (DOE) proposed plans for standardization of three advanced reactor designs and to obtain Commission endorsement of the criteria that the staff proposes to use in the review of DOE's plans for advanced reactor standardization. This paper responds to the Chairman's memorandum dated July 9, 1987.

Summary: This paper presents a set of criteria that the staff developed for use in the review of DOE's plans for standardization of three advanced reactor concepts. The criteria address the following standardization issues:

A. scope and level of detail of design to be standardized
B. plant options (number of reactor modules) to be standardized
C. prototype testing

The staff's proposed criteria for resolution of these issues were developed to be consistent with the intent of the Commission's policies on standardization and advanced reactors. The criteria are consistent with the staff's proposed rulemaking on Standard Design Certifications (10 CFR 52) and the staff's proposed criteria in the "Key Licensing Issues" SECY paper. Specifically, Section C of this paper and Section II.B.3 of the Key Licensing Issues paper both discuss the need for prototype testing in regard to certifying a design without a containment.

CONTACT: Jerry N. Wilson, RES
49-23729

8807290170 XA

Background: In SECY-86-368, "NRC Activities Related to the Commission's Policy on the Regulation of Advanced Nuclear Power Plants," dated December 10, 1986, the Commission was informed of the staff's plan for the review of three advanced reactor concepts sponsored by DOE. A summary of the reactor designs is provided in Enclosure 1. The reactors that are under review are:

1. A 350 MWt Modular High Temperature Gas-Cooled Reactor (MHTGR),

2. A 425 MWt liquid sodium cooled Power Reactor Inherently Safe Module (PRISM), and

3. A 900 MWt liquid sodium cooled Sodium Advanced Fast Reactor (SAFR).

One of the activities described in SECY-86-368 was the preparation of a Commission paper on the subject of standardization as it relates to the above advanced concepts. In view of the novel designs being proposed, this paper was prepared to (1) inform the Commission of the plans for standardization of the DOE sponsored advanced concepts and (2) to seek Commission endorsement on the approach and criteria to be applied by the staff in the review of the advanced concepts with respect to the revised Standardization Policy Statement (SECY-87-193 and 52FR34884).

Discussion: The three DOE sponsored advanced reactor programs all have as their objective the development of a standardized plant design which would be submitted to the NRC for Design Certification. A summary of the proposed plans for standardization of the advanced concepts is presented in Enclosure 1. These plans have associated with them certain issues which raise questions regarding Commission policy on standardization. These questions can be stated as follows:

A. Is the scope and level of detail of the plant design to be submitted for Design Certification consistent with the Commission's policy?

B. Can a single Design Certification cover the options desired for design, construction and operation of multi-module reactors on a site?

C. What R&D, prototype testing, initial startup testing or operating experience should be required for Design Certification of a new reactor design?

Each of these issues is discussed below along with a staff recommendation regarding the criteria which should be applied in the review of DOE's plans for advanced reactor standardization. Also, a preliminary staff assessment regarding the potential of the DOE proposed plans to meet these criteria is discussed. It should be emphasized that these proposed criteria are not intended to replace the Commission's Standardization Policy Statement, but rather are intended to supplement the guidance therein regarding the application of standardization to advanced reactors, including several unique issues which result from this application.

A. Is the scope and level of detail of the plant design to be submitted for Design Certification consistent with the Commission's policy?

In the revised Standardization Policy Statement, the Commission strongly encouraged the use of certified designs for plants, which are essentially complete in both scope and level of detail, in all future license applications. The Commission also recognized that review, approval, and certification of major portions of complete plants may be useful in the interim. However, applications for essentially complete designs are preferred and will be given priority in allocation of resources to support review and approval.

The PRISM and SAFR applicants plan to request Design Certification for most of the plant. In the MHTGR design, the extent of the plant to be submitted for Design Certification is limited to basically those structures, systems, and components (SSC) associated with the Nuclear Island such as the reactor vessel, primary system, and some key supporting systems. Four reasons were given by the reactor designers for limiting the certified portion of the designs:

1. They contend that all of the plant's safety systems will be contained within the certified envelope (with no systems interactions between safety and non-safety portions of the plant capable of affecting performance of the plant's safety functions). This, it is proposed, eliminates the need for NRC to approve anything other than interface requirements for the remainder of the design.

2. They are concerned that if the non-safety portion of the design were certified, NRC would get involved in design and construction verification to a greater extent than is necessary.

3. They argue that not certifying the entire plant will allow greater flexibility to incorporate design improvements or improvements in technology without having to go through the process of amending the Design Certification.

4. They state that in order to allow utilities the flexibility of procuring the balance of plant in a competitive fashion with design differences to suit their needs, Design Certification of the entire plant is not desired.

The three DOE advanced concepts also vary in the level of design detail proposed for Design Certification, as shown in Enclosure 2. In general, what is being proposed by PRISM and SAFR is certification of final design information for most primary system components, containment and the passive decay heat removal system. Beyond these, the level of design detail proposed for certification decreases and for those portions of the design not considered important to safety (such as the balance of plant) only general configuration and interface criteria will be proposed for Design Certification. The rationale for not supplying final design information on all portions of the plant to be certified is to allow for competitive procurement. For the MHTGR, Enclosure 2 describes the level of design detail for the FDA stage of licensing. The applicant will make recommendations regarding the level of design detail to be certified at a later date based upon its relative importance to radionuclide control.

With regard to DOE's position on scope and level of detail, it should be noted that the major contributors to non-standardized plants today are the differences from plant to plant external to the Nuclear Steam Supply System (NSSS). Problems external to the NSSS have been the initiator of many plant shutdowns, the focus of many Generic Safety Issues and have impacted plant safety. However, transients initiated in the non-safety related portions of the advanced designs should have less likelihood of leading to severe accidents. This is because the passive reactor shutdown and decay heat removal systems have the potential for high reliability since they are less vulnerable to failure modes involving active equipment, electric power, or human error. Therefore, even though failures or transients in the balance of plant could challenge safety systems, the overall risk from these challenges should be less than for LWRs. However, since the design and operation of the remainder of the plant is key to ensuring that the interface criteria with safety systems are met, that assumptions regarding accident initiators are maintained, and that operating experience gained on one plant is readily transferable to other plants, submittal of the entire plant for Design Certification is still preferred. This would eliminate the possibility of each plant varying substantially from the others, would make the preparation of a PRA and safety analysis more straight-forward and would minimize the time and staff resources required to review individual license applications to assess compliance with interface criteria. In addition, approval of a complete plant design at the Design Certification stage will afford a greater opportunity for wide public participation, as well as reducing the time and resources expended in repeatedly litigating the acceptability of a design at individual hearings. In short, the benefits to the Commission from standardization are maximized when the entire plant is certified.

For these reasons, the staff preference is to standardize and certify the entire plant. However, from the standpoint of performing a technical review, the staff could consider Design Certification of less than the complete plant provided that the certified portion of the plant contains all of the safety systems and the following criteria are met for the non-certified portion:

1. The interface requirements established for the non-certified portions of the design are sufficiently detailed to allow completion of a final safety analysis and a PRA for the plant.

2. Compliance with the interface requirements established for the non-certified portions of the design is verifiable through inspection, testing (separately or in the plant), previous experience or analysis. Compliance with interface requirements dealing with reliability of components or systems shall be verifiable through previous experience or testing.

3. A representative design for the non-certified portions of the plant is submitted along with the application for Design Certification as an illustration of how the interface requirements can be met and as an aid in the review of the PRA and safety analysis.

The above criteria would require certification of all the safety related portions of the plant and sufficient information on the other portions to determine overall safety. The staff would also require that the level of design detail submitted for the certified portion be final design information, equivalent to that provided in order to obtain an FDA. These criteria would ensure that the plant will be built and operated consistent with its safety analysis and PRA. Since the advanced designs are proposing balance of plant systems that are not safety related, the design flexibility desired by the designers would be retained for a large portion of the plant. The acceptability of the three DOE sponsored advanced reactor concepts with regard to scope and level of detail will be addressed in the respective SERs.

B. Can a single Design Certification cover the options desired for design, construction and operation of multi-module reactors on a site ?

Each of the three DOE sponsored advanced reactor designs is based upon a modular reactor concept. Accordingly, the reactor designers desire, as a fundamental feature of their approach, the flexibility to construct and operate various numbers of modules, including a staggered construction and operating schedule for the reactor modules on a given site. This flexibility may result in various design options with respect to the sharing of common systems (electrical power, control room, security, etc.), interface requirements and system interactions. As such, the staff considers it essential that in the application for Design Certification the following criterion be met:

> Various options regarding plant/site configuration should be described in the application for Design Certification, including variations in or sharing of common systems, interface requirements and system interactions. The safety analysis and PRA should also account for differences among the options, as necessary, including any restrictions during the construction and startup phase necessary to ensure the safe operation of those modules already on-line.

The above criterion requires that the application for Design Certification specifically address the allowable design options so as to prevent unreviewed plant or site arrangements from being constructed. It is also envisioned that each reactor module would require a separate operating license. Each of the DOE sponsored advanced reactor programs appears to be in agreement and consistent with this approach.

C. What R&D, prototype testing, initial startup testing or operating experience should be required for Design Certification of a new reactor design?

The Commission's response to question #6 in the Policy Statement on Advanced Reactors (51FR24646) addressed the issue of whether or not a prototypic demonstration of an advanced reactor concept was required prior to licensing a commercial facility utilizing an advanced reactor design. In summary, it was stated that proof of performance would have to be demonstrated and that the need for a prototype demonstration plant can be determined only by review of a specific design. In addition, the Standardization Policy Statement (52FR34885) states that, "When an advanced design concept is sufficiently mature, e.g., through comprehensive, prototypical testing, an application for design certification could be made." To implement this policy guidance, the staff proposes to apply the following criteria regarding full-size prototype testing of an advanced reactor design:

1. For advanced reactor designs which do not include a conventional containment building, demonstration of the plant's safety via full size, comprehensive prototype reactor testing will be required prior to Design Certification. Such testing should be done at an isolated site (such as the National Reactor Testing Station) and the prototype plant should conform to the same regulations and standards as the design to be certified.

2. For advanced reactor designs with a conventional containment building, a full size prototype test or demonstration will be required prior to Design Certification unless the following can be demonstrated:

a. the performance of each safety feature of the design has been demonstrated either via previous experience or via full scale testing,

b. interdependent effects among the safety features of the plant have been found acceptable by analysis, testing, or previous experience, and

c. sufficient performance data exist on the safety features of the plant to assess analytical tools used for safety analyses over a full range of operating and accident conditions, including equilibrium core conditions.

The above criteria are intended to ensure that prior to granting a Design Certification to any design significantly different from one that has been built and operated before, high confidence in the performance of the safety features of that design is demonstrated. In particular, the staff recognizes that a design without a conventional containment building represents a significant departure from past practice on LWRs and that under certain situations, LWR containment buildings have proven to be an effective component of the defense in depth approach. Therefore, sufficient confidence needs to be obtained in designs which deviate from such practice to demonstrate that uncertainties in design and performance are properly accounted for. Requiring prototype testing is considered necessary to provide such a demonstration and to compensate for removal of the traditional (and testable) containment building. Such prototype testing will help ensure that plants of that design have adequate fission product retention prior to Design Certification. These criteria will allow designs with variations in containment design to be certified. It should be noted that a Final Design Approval (FDA) is required prior to an application for Design Certification. Therefore, the FDA will identify the specific testing (initial startup, demonstration, or prototype) required to support the licensing of an individual plant or class of plants referencing that FDA.

It should be noted that, besides testing a design without a conventional containment at an isolated site, an additional option was considered. This option would allow the prototype reactor to be built and tested at a standard site (any site consistent with the requirements of R.G. 4.7) provided that a containment building and traditional emergency planning were provided for the prototype. Although this option is technically viable and is intended to provide the industry with more flexibility in developing and locating a prototype, its application has several practical problems which make it more difficult, such as:

1. R.G. 4.7 was intended to apply to sites for reactors which have been demonstrated. Specifically, it states that it is limited to LWRs and HTGRs. Therefore, this option could lead to a license application for a reactor in a potentially moderate population area whose performance is sufficiently uncertain that special tests and a containment are needed to assure its safety. This could result in much opposition and protracted licensing hearings.

2. Questions regarding the containment's affect on the performance of the passive decay heat removal system would need to be resolved such that the testing would in fact confirm the design to be certified.

3. The basis for the containment design would have to be established. In addition, it is likely that the containment design would have some unique features, due to the fact that its design would have to allow demonstration of the passive decay heat removal system. These unique features would require testing and qualification prior to their acceptance.

4. The characteristics of the site may limit the extent of and type of testing which could be performed.

If this option is of real interest to a potential applicant and if the Commission wishes to have such an option available, the staff would need to develop additional guidance addressing factors important to its implementation, such as the containment design basis, containment isolation and qualification requirements and testing program objectives and strategy.

All of the DOE sponsored advanced reactor designs have as part of their development plan, the construction, testing and operation of at least one reactor module (and its associated supporting systems) prior to receiving a Design Certification. However, the reasons given for including such a prerequisite in their plans vary among the concepts. General Electric (PRISM) has proposed a full size prototype reactor located at an isolated site. Rockwell International (SAFR) has proposed a full size prototype reactor at a utility site. For these designs, prototype safety demonstration tests are planned. Their reasons for proposing these tests are (1) on any new reactor design, lessons will be learned from its operation which can then be factored into the design prior to its certification, (2) the safety characteristics of these designs are sufficiently innovative that they need to be demonstrated via a full scale test prior to certification and (3) it is hoped that the demonstration of the safety of these concepts will be used in the licensing process to simplify the review. For the MHTGR, a standard plant demonstration is planned at a utility site; however, no special safety tests are proposed as they are considered by DOE to be unnecessary to obtain Design Certification. DOE claims that the safety features of the MHTGR are sufficiently simple (as compared to LWRs or LMRs) that full-scale safety testing is not required.

As part of the General Electric proposal for prototype reactor testing, discussions were held regarding the potential benefit of adding to the regulations an option which would allow the use of prototype demonstration of reactor performance as a means of achieving Design Certification. This option is discussed further in Enclosure 3. It is the staff's view that the forthcoming rulemaking (10 CFR 52) which covers Design Certification, including prototype testing, provides adequate definition in this regard. The staff's proposed criteria on prototype testing (as stated above) are consistent with the proposed 10 CFR 52. Specific testing required on the prototype reactors for each DOE concept will be addressed in the staff's SERs on the conceptual designs.

Conclusion: The standardization of advanced reactors poses several unique issues not faced in the standardization of current generation LWR designs. Criteria for resolution of these issues are proposed, as described above. These criteria were developed to be consistent with the intent of the Commission's policies on standardization and advanced reactors. They have been coordinated and are consistent with the rulemaking (10 CFR 52) being developed on Early Site Permits, Standard Design Certifications, and Combined Licenses for Nuclear Power Reactors.

Coordination: The Office of the General Counsel has reviewed the contents of this paper and has no legal objection.

Recommendation: The Commission endorse the staff's proposed criteria for assessing the adequacy of DOE's plans for advanced reactor standardization.

Victor Stello, Jr.
Executive Director for Operations

Enclosures: As stated

Commissioners' comments or consent should be provided directly to the Office of the Secretary by c.o.b. August 31, 1988.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT August 5, 1988, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

DISTRIBUTION:
Commissioners
OGC
OI
OIA
GPA
Regional Offices
EDO
ACRS
ASLBP
ASLAP
SECY

ENCLOSURE 1

PROPOSED DESIGNS SUBMITTED FOR REVIEW

Modular High Temperature Gas-Cooled Reactor (MHTGR)

The MHTGR concept is a helium cooled, graphite moderated reactor. A design objective of this reactor is to meet Protective Action Guideline dose limits for sheltering at the Exclusion Area Boundary with minimal reliance on active systems and without operator actions. It is a 350 MWt standard reactor module designed by GA Technologies in association with Combustion Engineering and Bechtel. The MHTGR plant design using the standard reactor module is being developed by Stone & Webster and General Electric. The Utility/User requirements for the overall plant design were established by Gas-Cooled Reactor Associates.

The reactor module consists of a steel reactor vessel connected by a crossduct vessel to an adjacent steel vessel housing the steam generator and the main helium circulator. All three vessels are located below grade in a concrete reactor cavity. The MHTGR plant design does not include a conventional containment building.

The fuel design is similar to Fort St. Vrain, except that the MHTGR uses low enriched uranium. The fuel, through its high temperature performance capability, provides the major containment function in the plant for design basis and beyond design basis events (e.g. ATWS plus loss of coolant, catastrophic vessel failure) and fuel and vessel integrity is maintained for an extended period (several days to weeks) for a total loss of all decay heat removal systems.

Key features of the design are passive reactor shutdown characteristics (i.e. large negative temperature coefficient) and a passive decay heat removal system that utilizes natural draft air flow from the atmosphere to panels located in the reactor cavity to cool the reactor vessel through radiation and convection and thereby remove decay heat. In addition, even if this system is lost, adequate decay heat removal can be achieved by radiation and conduction to the reactor cavity concrete and surrounding earth. Due to the large thermal capacity of the graphite core, the heatup of the core is slow, when compared to LWRs, even for cases where no decay heat removal system is assumed operating.

The basic plant design consists of four reactor modules connected to two steam turbines. Associated with this arrangement are the other supporting facilities (control room, fuel handling equipment, maintenance building, etc.). Within the plant the major items that the applicant considers safety related are the reactor vessel and its internals (including fuel), portions of the reactivity control systems, the associated primary pressure boundary and the passive decay heat removal system. It is currently their intent to request Design Certification for that portion of the Nuclear Island relied upon for the control of radionuclides (which includes all safety related SSC) and some key interface requirements. The remaining SSC (which includes the balance of plant and its supporting systems) would not be included in the application for Design

Power Reactor Inherently Safe Module (PRISM)

The PRISM plant design uses a 425 MWt standard reactor module designed by General Electric (GE). The PRISM plant design using the standard reactor module is being developed by a GE led team with Bechtel as the supporting architect engineer.

Sodium is the primary coolant with a pool type primary system (all primary system components are located within the reactor vessel). The reactor module is located below grade in a concrete silo. The fuel is U/Pu/Zr metal similar to that used in DOE's EBR-II reactor.

The containment is a hermetically sealed low pressure, low volume, leak-tight barrier that employs a factory fabricated containment vessel around the reactor vessel, reactor head closure assembly and intermediate heat exchanger (IHX) tubes as the low leakage barrier against postulated release of the core fission products. In addition, the containment vessel is located close to the reactor vessel to assure that primary coolant leaks from the reactor vessel do not result in loss of core cooling (i.e., the containment vessel functions as a coolant retaining guard vessel).

Key features of this design include passive reactor shutdown characteristics (via large negative temperature coefficients) and a passive decay heat removal system with natural draft air flow from the atmosphere for cooling the reactor vessel. The plant is designed to withstand large disturbances (e.g., Transient Overpower w/o Scram and Loss of Flow) with these passive systems. Due to the large thermal capacity of the sodium pool, the heatup of the core is slow, when compared to LWRs, even for cases where no decay heat removal system is assumed operating. Another key feature of this design is seismic isolation of the complete reactor module to improve resistance to severe earthquakes.

The basic plant design consists of nine reactor modules arranged with three modules supplying steam to one turbine. Associated with this arrangement are the other supporting facilities (control room, fuel handling equipment, maintenance building, etc.). Within the plant the major items that GE considers safety related are the reactor vessel and its internals (including fuel), the primary system and containment boundaries, the passive decay heat removal system, the reactor protection system, the reactivity control and shutdown system, the seismic isolators, and the reactor refueling system.

The entire plant will be submitted for Design Certification, but the level of design detail will vary. It is GE's intent to show that the reactor system is highly independent of the balance of plant, and therefore, the evaluation needs to focus mainly on the safety related systems. A complete design will be submitted for the safety related systems and for the intermediate heat transport system. The remaining SSC, which includes such items as the control room, steam and feedwater system, and balance of plant systems would then be defined by interface requirements in the application for Design Certification. A representative design for the remainder of the plant will be provided for review as an example of how the standard interface requirements can be met.

SODIUM ADVANCED FAST REACTOR (SAFR)

The SAFR design utilizes a 900 MWt standard reactor module designed by Rockwell International (RI) with Bechtel National, Inc., Combustion Engineering, Inc., and Duke Engineering Services as subcontractors. Sodium is the primary coolant, with a pool type primary system (all primary system components are located within the reactor vessel). The reactor vessel is located above grade. The fuel is U/Pu/Zr metal similar to that being used and tested in DOE's EBR-II reactor.

The containment is a low pressure, low volume barrier which utilizes a containment vessel, which surrounds the reactor vessel, and reactor vessel closure head. The containment vessel also functions to assure that primary coolant leaks from the reactor vessel do not result in uncovery of inlets to the invessel heat exchangers and consequent loss of core cooling. A secondary containment building also functions as a barrier to the release of fission products through mechanical seals in the reactor vessel closure head.

As for the MHTGR and PRISM plants, key features of the SAFR plant include passive reactor shutdown characteristics (via large negative temperature coefficients) and a passive decay heat removal system with natural draft air flow from the atmosphere for cooling the reactor vessel. The plant is designed to withstand large disturbances (e.g., Transient Overpower w/o Scram and Loss of Flow) with these passive systems. Due to the large thermal capacity of the sodium pool, the heatup of the core is slow, when compared to LWRs, even for cases where no decay heat removal system is assumed operating.

The basic plant design consists of a single reactor supplying steam to a single turbine. Associated with this arrangement are the other supporting facilities (control room, fuel handling equipment, maintenance building, etc.). Within the plant the major items that RI considers safety related are the reactor vessel and its internals (including fuel), the primary system and containment boundaries and the passive decay heat removal system. It is currently RI's intent to request Design Certification an all SSC (this includes all safety related as well as non-safety related systems such as the balance of plant auxiliary buildings and systems) except for site specific items.

While the entire plant is to be submitted for Design Certification, the level of design detail submitted will vary. The primary RI guideline is to request Design Certification of the design down to a level that assures that allowed variations do not invalidate the safety analysis or PRA. In general, Design Certification on the nuclear island will be requested down to the detailed physical design and arrangements and on the balance of plant down to the performance and procurement specifications and preliminary design and arrangements. This approach is intended to permit limited competitive bidding without impacting plant safety.

The application for Design Certification would contain four options - a one module, two module, three module, and four module plant. Differences among these options (including the sharing of common facilities) would be described. It is RI's desire that the Design Certification granted by the Commission allow a utility the flexibility to order a one, two, three or four module plant with

ENCLOSURE 2

PRELIMINARY PLANS FOR THE EXTENT OF THE ADVANCED REACTOR DESIGNS SUBMITTED FOR CERTIFICATION

Structures, Systems, and Components		Level of Design Detail to be Certified		
		MHTGR (FDA)	PRISM	SAFR
1)	Containment (Vessel & Closure)	N/A	5	5
2)	Reactor:	4	5	5
	° RV, Fuel, other internals			
3)	Heat Transport System:			
	° Primary	4	5	5
	° Secondary	N/A	5	5
	- Steam Generator	3	5	5
4)	Shutdown Heat Removal System:			
	° Safety grade reactor cavity air system	4	5	4
	° Non-safety grade auxiliary system	2/3	4	4
5)	Plant Control & Protection Systems			
	° Control systems	3	4	3/4/5
	° Protection systems (including control rods & drives)	4	5	4/5
	° Control room	1	4	4
6)	Steam & Feedwater System			
	° Steam system	2	2	2
	° Feedwater system	2	2	2
	° Turbine/Condenser	2/1	2	2
	° Cooling Towers	1	2	1
7)	Electrical Systems			
	° Safety grade	4	5	4
	° Non-safety grade	2	2	2/3
	° Switchyard	1	2	1
8)	Instrumentation:			
	° Safety grade	4	5	4
	° Non-safety grade	2	4	4
9)	Radioactive Waste Mgt. System	2	5	3/4
10)	Auxiliary Systems:			
	° FH & Storage	3	5	5
	° Water/H&V/Fire Protection	2/1	2	2
	° Communications	1	2	2
	° Na or He auxiliary	2	5	4
11)	Buildings/Structures:			
	° Reactor	4	5	4
	° Refueling	4	5	4
	° Steam Generator	4	5	4
	° Turbine/BOP	2/1	2	2
	° Maintenance	2	2	2
12)	Plant Security	2	3	2/3

Explanation of Certification Levels

1. Open design; not standardized, variable from plant to plant.

2. Interface and/or functional requirements specified for components or between systems.

3. General specification of type, process, material, location, and arrangement for structures, major structural elements located, and equipment size envelops specified.

4. Design requirements specified to the component level including detailed performance, interface, size, and arrangement characteristics. The information so provided is sufficient to permit a final design to be accomplished or to be chosen from available sources.

5. Complete design; information includes detailed fabrication drawings, specifications, acceptance criteria, and production specifications. May include "name plate information" in some cases.

Each level from 2 to 5, includes the requirements of all of the preceding levels. Additional design detail and standardization requirements are planned by the designers as user requirements to address those portions of the design not certified.

ENCLOSURE 3

DESIGN CERTIFICATION BY TESTING

During our review of the PRISM conceptual design, the General Electric Company (GE) requested discussions with the staff regarding the potential benefit of adding a new rule to 10 CFR 50 to provide another option for achieving Design Certification. This new rule (referred to as Appendix S by GE) would be similar to Appendix O to 10 CFR 50, but it would emphasize the use of demonstrated reactor performance as a means of achieving Design Certification, as in the Federal Aviation Administration's (FAA) approach to aircraft design certification (14 CFR 21). While this approach was developed by GE to facilitate certification of their PRISM design, it could be applied to any advanced reactor that has minimal prototypic operating experience.

GE's motivation for exploring the usefulness of their proposed Appendix S to 10 CFR 50 was based upon their FDA experience with the GESSAR II design and their plans for certification of the ABWR design. GE believes that the use of Appendix O for Design Certification will be restricted to proven designs with performance demonstrated through operating experience. Their view is supported by the Commission's recent policy statement on Nuclear Power Plant Standardization which encourages certification of reference system designs. The policy statement (52 FR 34885) states that, "The reference system designs, at least initially, are expected to be evolutions of existing proven LWR designs. ... When an advanced design concept is sufficiently mature, e.g., through comprehensive, prototypical testing, an application for design certification could be made." By creating a rule that clearly applies to first of a kind designs using prototypic testing to demonstrate safety characteristics, GE believes that the uncertainties associated with applying Appendix O to advanced reactors would be eliminated. Then advanced reactors can also achieve the benefits of Design Certification.

The staff has reviewed the merits of GE's approach to Design Certification as part of our overall review of DOE's advanced reactor concepts. GE's approach would include a review of a complete Safety Analysis Report, a Probabilistic Risk Assessment (PRA), and a parallel testing program which includes component testing, system testing, and testing/operation of a full-scale prototype reactor facility. The testing program would be mutually established by NRC and the applicants early in the review process and would include: the testing of normal operating conditions (startup, shutdown, power increases/decreases, fueling, and refueling); the testing of all off-normal conditions; and all reasonably testable accident conditions. This testing would verify the performance of the inherent (passive) safety features, the safety systems, and would be of sufficient scope to verify performance and releases over a wide range of accident conditions postulated to occur during the plant lifetime.

For comparison with the GE approach, the staff also reviewed FAA's approach to Design Certification. Our review of the FAA process has included numerous meetings with FAA senior review personnel. The FAA process has many similarities to NRC's review objectives and has the potential to be applied to the review of manufactured reactor modules after extensive review and testing on a full-scale prototype module had been successfully completed. The similarities in the NRC and FAA processes include rulemaking, inspection powers, enforcement powers, public meetings, use of PRA techniques, use of safety reliability goals, defined general design criteria (such as single failure tolerance, QA, record keeping, and fire protection), technical specifications (which include limiting conditions for operation), and operational follow-up via advisory bulletins and design change directives.

The FAA process initially focuses on rather extensive component, system, and structural testing for all new aspects of the prototype design which have not been reviewed and accepted by FAA. Upon successful completion of the prototype testing, FAA addresses requirements for the production manufacturing process, which includes individual acceptance testing of each product. Finally, FAA maintains an operation oversight activity throughout the product life.

GE believes that the FAA method of reviewing new designs may be well suited to reviewing shop-fabricated reactor modules. GE is proposing to include full-scale prototype testing in their certification process and is seeking to certify the PRISM design based on the NRC review of the application and the prototype test results. They will then standardize the design based on this review process and will manufacture identical modules for use in generating electricity throughout the U.S. GE projects that their design envelope will enable location of the modules in more than 90% of possible U.S. candidate sites. Based upon our review to date, it appears that portions of the FAA process could be incorporated into the NRC's approach to certification of new, advanced, shop fabricated reactors.

The NRC staff is currently developing proposed regulations (10 CFR 52) on Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Reactors to implement the Commission's Standardization Policy and as much of its legislative proposal as can be accomplished under existing statutory authority. Since the GE proposal is consistent with the Advanced Reactor Policy Statement (51 FR 24643) and the Standardization Policy Statement (52 FR 34884), the staff has included the design certification by test approach for advanced reactors into the proposed 10 CFR 52. We recommend that the Commission issue 10 CFR 52 for public comment with this approach included.

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

OFFICE OF THE
SECRETARY

October 10, 1989

NOTE FOR THE RECORD

FROM: Jack Guttmann, SECY

SUBJECT: SECY-89-202 - STANDARDIZATION OF ADVANCED REACTOR DESIGNS

The subject paper was withdrawn by the staff prior to Commission action. Consequently, the Commission did not act on this paper.

PDR

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

OFFICE OF THE SECRETARY

October 11, 1989

MEMORANDUM FOR: Teresa Neville, Acting Chief
Public Document Room

THRU: Sandy Showman, Chief
Correspondence and Records Branch

FROM: J. Guttmann for Andrew Bates, Chief
Operations Branch

SUBJECT: RELEASE OF DOCUMENT TO PDR

Attached for placement in the PDR is a copy of:

SECY-89-202 - Standardization of Advanced Reactor Designs

This document, witn a "Note for the Record" attached to the paper, is being placed in the PDR at the request of the staff and with concurrence of the Commission offices.

Attachment:
As stated

cc: DCS - P1-124

DF02
1/1