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Abstract

This paper employs the criteria for evaluations from the Nuclear Power Option Viability Study to examine the bases for proposed innovations in light water reactor safety technology. These bases for innovation fall into four broad categories as follows: (1) virtually exclusive reliance on passive safety features to preclude core damage in all situations, (2) design simplification using some passive safety features to reduce the frequency of core damage to less than about 10^{-6} per reactor-year, (3) passive containment to preclude releases from any accident, and (4) designing to limit licensing attention to one or at least a few systems. Of these, only the first two, and perhaps only the second, hold significant promise for providing for the viability of advanced light water reactors.

Introduction

During fiscal years 1984 and 1985, a group at the Oak Ridge National Laboratory performed a program of work entitled the Nuclear Power Options Viability Study (NPOVS).[1,2] One of the several objectives of the NPOVS was to characterize the research and development needs essential for the successful deployment of advanced reactor concepts early in the next century, by which time the need for new baseload generating capacity is likely to be evident. To limit the study to a tractable scope, the NPOVS examined only those concepts claimed by their proponents to emphasize passive safety features in the design.[3] For the purpose of the NPOVS, "passive safety" was defined as the reliance on natural physical laws and properties of materials to effect shutdown and radioactive decay heat removal without fuel damage and without relying on mechanically or electrically actuated and driven devices such as those employed in active (engineered) systems. The passive safety features, which are characteristic of most of the advanced reactor concepts, are listed in Table 1. To assess and evaluate each concept's viability with respect to marketability and licensability, the NPOVS derived and employed seven quantifiable criteria (Table 2) and a number of less easily quantified essential and desirable characteristics (Table 3).

Although the NPOVS was purposely limited to advanced reactor concepts which emphasized passive safety, it was fully understood that passive safety is not an end in itself but rather an approach that some proponents believe can be used to salvage the nuclear power option. Currently, there appear to be at least four distinct bases upon which

Table 1

Passive Safety Features Being Employed in the Design of Advanced Thermal Reactors (Ref. 3)

- A reduced core power density, sometimes by as much as half of that in traditional designs.

- An increased passive heat sink either available within the primary system or unobstructively available to the primary system by the action of the physical laws of nature.

- A passive or inherent assured shutdown mechanism which responds as in most reactors to overpower/loss of flow transients but always prior to core damage.

- The elimination or minimization of leak paths from the primary system coolant piping by physical design modifications.

- An enhanced robustness of fuel elements.

- The incorporation of the latest advances in coolant chemistry and materials technology to reduce the primary coolant radioactivity levels and to minimize corrosion induced leak paths.

Table 2

The NPOVS Criteria for the Evaluation of Advanced Reactor Concepts (Ref. 1)

- The calculated risk to the public due to accidents is less than or equal to the calculated risk associated with the best modern light water reactors (LWRs).
- The probability of events leading to loss of investment is less than or equal to 10^{-4} per year (based on plant costs).
- The economic performance of the nuclear plant is at least equivalent to that for coal-fired plants. (Financial goals for the utility are met, and busbar costs are acceptable to the public utility commissions.)
- The design of each plant is complete enough for analysis to show that the probability of significant cost/schedule overruns is acceptably low.
- Official approval of a plant design must be given by the U.S. Nuclear Regulatory Commission (NRC) to assure the investor and the public of a high probability that the plant will be licensed on a timely basis if constructed in accordance with the approved design.
- For a new concept to become attractive in the marketplace, demonstration of its readiness to be designed, built, and licensed and to begin operations on time and at projected cost is necessary.
- The design should include only those nuclear technologies for which the prospective owner/operator has demonstrated competence or can acquire competent managers and operators.

Table 3

The NPOVS Essential and Desirable Characteristics of Advanced Reactor Concepts (Ref. 1)

Essential Characteristics:

- Acceptable front-end costs and risks
  - Construction economics
    - Low and controllable capital costs (utilizing, for example, shop fabrication, a minimum of nuclear grade components, and standardization)
    - Designed for long lifetime
  - Investment economics, including risk
    - Low costs associated with accidents
    - Low costs associated with construction delays
    - Low costs associated with delayed or unanticipated actions by regulatory bodies
    - Low costs associated with delayed or unanticipated actions for environmental protection
    - Unit sizes to match load growth
    - Uncertainties in technology and experience not likely to negate investment economics
- Minimum cost for reliable and safe operation
  - High availability
  - Minimum requirements for operating and security staffs
  - Designed for ease of access to facilitate maintenance
  - Simple and effective modern control system
  - Low fuel cycle costs
  - Adequate seismic design

Table 3 (Cont.)

- Practical ability to construct
  - Availability of financing
  - Availability of qualified vendors
  - Availability of needed technology
  - Adequately developed licensing regulations applicable to the concept
  - Ease of construction enhanced by design
- Public acceptance
  - Operational safety of power plants
  - Safe transportation and disposal of nuclear waste
  - Low radioactive effluent
  - Low effect on rates of construction and operation
  - Adequate management controls on construction and operation
  - Utility and regulatory credibility.

Desirable Characteristics:

practical research, development, and demonstration requirements
ease of siting
load-following capability
resistance to sabotage
ease of waste handling and disposal
good fuel utilization
ease of fuel recycle
technology applicable to breeder reactors
high thermal efficiency
low radiation exposure to workers
high versatility relative to applications
resistance to nuclear fuel diversion and proliferation
on-line refueling
low visual profile

different proponents approach innovation in reactor safety technologies. These bases for innovation can be characterized as follows:

1. The first approach, which has been espoused by Weinberg,[4] Crane,[5] Hannerz,[6] and others, is that a major, if not always radical or revolutionary, change is needed in reactor design. This change involves almost exclusive reliance on passive safety features in order to preclude or significantly delay core damage. Without core damage, the public health and safety is always assured. With significant increases in the time available before core damage, public health and safety is more easily and convincingly accommodated. "Licensing by test" is often espoused by the proponents of this approach as a means to convince the public and the regulators of inherent plant safety. In general, this approach is reflected in the ASEA-ATOM Process Inherent Ultimate Safety (PIUS) pressurized water reactor (PWR) and to a lesser extent in the General Electric (GE) small boiling water reactor (BWR).[8] The Small BWR is prototypical of the type of reactor which the Electric Power Research Institute (EPRI) is addressing as a part of its Advanced Light Water Reactor (ALWR) Program.[9] These concepts sacrifice the economics of very high power ratings for the ability to prevent or delay core damage and for the potential cost savings from reduced requirements for engineered safety devices as well as from shorter construction times.

2. The second approach is that adopted by the leading major U.S. reactor vendors in an alliance with those of Japan.[10,11] This approach to designing ALWRs seeks to focus innovation within the context of accumulated experience. The objectives are to simplify the design and demonstrate safety probabilistically by carefully redesigning the reactor to reduce the total core melt frequency to about 10^{-6} per reactor-year. Elements of passive safety features are included but are limited since design evolution rather than radical change is believed to be most cost effective. This approach also seeks to maintain the perceived economy of scale of high power ratings.

3. The third approach is one based on the contention that current reactors are adequately designed for both economy and safety but that the public concern about perceived safety hazards can be overcome best by innovative design of the containment structures to achieve an ultrasafe configuration.[12] Ultrasafety is to be achieved by the use of core retention devices below ground level and of a combination of passive or active containment coolers and filters. Proponents have proposed making existing plants ultrasafe with backfitted enhancements to the containment.

4. Finally, there is an aspect of innovation that is common to some extent to all of the above approaches to enhanced reactor safety. This approach seeks to focus the attention of licensing to one or at most only a few systems and components which assure the public health and safety. The idea is to demonstrate that the safety of the design is not contingent upon multiple, independent and

redundant defense-in-depth features but rather depends only upon one or a few features which can be relied upon exclusively or with a very high probabilistic confidence. In particular, this approach seeks to simplify licensing by separating out the safety features from those which are argued to represent "investment protection."

This paper employs the NPOVS criteria (Table 2) to examine briefly the current bases for proposed innovations in light water reactor (LWR) safety technology. The bases for such innovations are also reviewed against the licensing history of the Fort St. Vrain HTGR, which was apparently the first reactor to attempt the use of "inherent safety" arguments in licensing documentation.[13] Special attention is given to the potential for problems which may be incurred in "licensing by test" and in separating out "investment protection" features from the traditional defense-in-depth approach to licensing.

Innovation through Passive Safety

The LWRs receiving the most attention, because of their respective proponents' emphasis on the use of passive safety features, are ASEA-ATOM's PIUS-PWR and the GE Small BWR. Since its introduction, the PIUS-PWR has gone through at least three design iterations and has also spawned a set of derivative concepts such as the PIUS-BWR[14] and the smaller, barge-mounted Intrinsically Safe and Economical Reactor (ISER).[15] Since none of the latter has a strong industrial proponent nor differs significantly in terms of the fundamental principles motivating the PIUS approach, the PIUS derivative concepts will not be addressed further here. However, the Small BWR has apparently set a standard against which the PWR vendors are attempting to compete within EPRI's Small (<600 MWe) ALWR Program. The information about the small advanced PWR concepts is sketchy and based primarily on the preliminary information presented recently in the *EPRI Journal*[9] plus another recent paper on the Babcock & Wilcox concept.[16].

There are several important and fundamental differences between the PIUS approach to innovation and that of the concepts being studied in the EPRI Small ALWR Program. These differences are highlighted as follows.

PIUS attempts to make the leakage of water from the reactor vessel an incredible event by the use of a thick (~8 m) prestressed concrete pressure vessel with redundant metallic membranes and with no vessel penetrations below the height of a minimum seven day water supply for accommodating the decay heating. PIUS incorporates suction breakers on all vertical penetrations into the lower pool. The primary system housing and piping are immersed in the cool, highly borated water pool contained within the large pressure vessel. The primary coolant and the borated pool water are in direct contact at thermal interfaces provided by density locks located at the top and bottom of the primary system's insulated housing. The pressure differential across the density lock is maintained in equilibrium (no flow) by the primary system recirculation pump located on the hot leg. The core is placed at the bottom of the primary system, and the steam generators are at the top. The pool and primary system share a common pressurizer unit. Core reactivity control is effected by controlled boration and deboration. The concept has no control rods. Scram is effected by turning off the recirculation pump and allowing the highly borated pool water to enter the primary system through the lower density lock. Natural circulation through the density locks into the surrounding pool and the use of pool coolers effects normal residual heat removal. Any transient that could cause an approach to boiling in the reactor core creates sufficient thermal-hydraulic instability at the density locks to cause a scram.

In contrast, the small ALWRs are less intimately linked to their passive heat sinks and backup shutdown mechanisms than is PIUS. The small ALWRs also utilize control rods and a more conventional steel pressure vessel housed in a large, relatively dry containment building. Each of the small ALWR concepts utilizes one or more passive cooling features, such as an isolation condenser in the small BWR or a natural circulation heat exchanger in the Westinghouse small PWR, for effecting residual heat removal to the passive heat sink without actuating the direct injection of cooling water from the passive heat sink into the reactor. For the small ALWRs the passive heat sink and backup shutdown mechanism are essentially an elevated pool of borated water connected to the reactor by gravity-drain pipes with check valves held shut by reactor pressure. The Babcock & Wilcox concept is stated also to require some flow from diesel-driven pumps.[9,16] Emergency core cooling is intended to be initiated in the small ALWR concepts only during vessel depressurization due to either a line break or the actuation of a fail-open depressurization valve on reactor vessel overpressure. The latter could result from a total loss of normal heat sinks or from an overpower transient without scram. Apparently, in each concept, enough water is provided in the elevated pools to flood the lower containment to a level above the core and possibly above the postulated break location, although cooling through the reactor vessel wall is mentioned, so that several days of passive cooling are provided. Thus, the flooding of the lower containment causes the configuration of the small ALWRs to more resemble that of the PIUS concept during the latter stages of a worst case accident. Although not stated in available documents, it is presumed that the lower containment of each small ALWR is designed to preclude leakage.

The cartoon of the small PWR concept presented in the *EPRI Journal* article[9] shows also a steel containment shell which provides natural draft air cooling of the containment walls and thereby condenses steam rising from the flooded lower containment. This concept is also discussed briefly in Ref. 16. However, the EPRI cartoon does not illustrate the assumed status of any containment shell penetrations, and the discussion[16] is limited to transients such as station blackout. The proponents of PIUS assumed such penetrations could and therefore would have failed so that pool boiloff would be lost following the worst case accident. Of course, the small ALWR concepts are still in their early stages of development and so the actual safety benefit and cost effectiveness of passive containment cooling may not have yet been assessed fully. On the other hand, the proponents of PIUS may need to further examine the need for containment. The assumption of the PIUS designers is that they can design high integrity fuel rods which will not be challenged except by an externally induced catastrophe such as war.

The evaluation of the passively safe LWRs against the NPOVS criteria (Table 2) can yield only a preliminary assessment of potential viability because of the preliminary status of each concept's development. All of these concepts address the safety issues in such a way that the technology envisioned appears to be feasible in terms of reducing public risk below that of the best modern LWRs. The real questions to be answered relate to the potential for loss of investment and to the economic performance. Although no single event may threaten loss of investment, overall poor performance of new features, such as that which is being experienced at Fort St. Vrain, may in effect force a loss of investment upon a utility. None of the concept's proponents has yet to publish a detailed capital and operating cost estimate for external review. Without these cost estimates, concept viability cannot be adequately assessed. The small ALWRs appear to have the potential advantage of a more evolutionary design based on experience with small LWRs in the past. The proponents of PIUS have yet to settle on the final reactor configuration, and PIUS represents a greater departure from existing experience.

From the standpoint of technology development, several conclusions can be drawn about the extent and nature of development and testing needs of these reactors. For PIUS, these needs are extensive and include further demonstration of fluid interface stability, extensive study and testing of steam generator modules, thorough testing of underwater fuel and equipment handling systems, steam flow and pressurizer stability for the possible multimodular design, thermal insulation development and testing, and demonstration of the prestressed concrete pressure vessel design, particularly for the top closure. The proponents of PIUS push for a non-nuclear full-sized demonstration, but a demonstration reactor may be required to adequately test the novel features of PIUS and to firmly establish commercial viability.

For the small ALWRs, the major development requirements appear to involve demonstration of the gravity-drain emergency core cooling systems, steam injector testing for the Small BWR, demonstration testing of the specific design for the natural circulation reactor vessel coolers (isolation condensor or elevated heat exchanger) and design, development and testing of the depressurization valves and control systems. Both BWR in-vessel pumps and the PWR canned pumps are existing technologies which have only to be adapted and tested for the specific design configuration.

Since boration control and its associated chemical processing systems are apparently being eliminated from the small PWR designs, detailed failure modes and effects analysis is required to ensure that emergency boration transients do not occur often since recovery will be time consuming. But surveillance testing methods will also have to be developed to demonstrate that the emergency cooling system will work passively as designed and does not degrade over time. Since flooding of the lower containment is possible from the elevated pool, the potential for pressurized thermal shock of the pressure vessel will have to be assessed fully due to the potential for inadvertent actuation of the flooding mechanisms.

Finally, the Small BWR appears to have a certain measure of advantage over the small PWRs because of the elimination of external recirculation loops to the BWR vessels. Compared to the Small BWR, the configuration of PWR primary flow loops external to the vessel poses relatively greater risk from loss of coolant accidents due to possible piping failures. Alternatives, such as a single vessel loop configuration similar to the Consolidated Nuclear Steam Generator design or local accommodation of pipe breaks by use of a multicell containment, should be evaluated thoroughly. The multicell containment option is addressed later in this paper.

Innovation through Evolution and Simplification

Developing the ALWRs as an extension of the past quarter century of experience is basically a natural expectation. The current lull in new reactor orders provides time to evaluate and incorporate the lessons learned into standard plant designs, but unfortunately this lull also taxes both vendor profit margins for supporting such investment and the patience of nuclear engineering professionals who desire to see a future in their careers. The two leading reactor vendors have aligned themselves with Japanese industrial partners in the development of the evolutionary, large (>600 MWe) ALWRs,[10,11] but the other two PWR vendors are not remaining idle.[16,17]

As discussed in a previous paper,[3] the larger ALWRs have adapted many of the passive features listed in Table 1. The larger ALWRs still rely on active (engineered) features for emergency core cooling, but the larger ALWRs are being designed with larger, deeper pressure vessels, reduced core power density, and, on PWRs, larger volume pressurizers. These passive design features allow stretching out the time period before emergency core cooling is needed during anticipated transients and loss of coolant accidents. The active emergency core cooling systems are also being redesigned for greater reliability through improved simplicity and redundancy. Primary system configuration and components have also been modified to reduce or eliminate potential leak paths. For the large advanced PWRs, soluble borate control is being eliminated in favor of using spectral shift and grey rods.

As discussed in the NPOVS Final Report, the NPOVS team recognized the advantages which accrue to the large ALWR concepts simply because of their evolutionary nature. However, it is also recognized that, while design simplification may lead to some design enhancement, the large ALWR will still be costly to construct and a complicated device to operate. The potential demonstrations provided in the Japanese deployment of these plants may be a sufficient basis for convincing a prospective buyer to invest in another large plant when the need for additional capacity is realized. An ALWR will be more attractive when the standard design has been approved by the Nuclear Regulatory Commission (NRC) as required in the NPOVS viability criteria. The reactor vendors must pursue early and continuing interaction with the NRC during the design development. The EPRI Requirements Document,[18] which is currently being developed with partial support from the U. S. Department of Energy (DOE), should also be very useful in defining and limiting the design and construction effort to a tractable scope for any utility with prior nuclear experience. Effort expended in making the Requirements Document complete, accurate and, most of all, usable can and should have a major positive impact on the acceptability of the large ALWRs by the utilities.

Ultrasafety Through Enhanced Containment

Under NRC sponsorship, alternate containment concepts have been evaluated since the middle 1970s as a means of enhancing the safety of LWRs against the potential consequences of severe accidents. A comparative evaluation of these alternatives was one product of the Industry Degraded Core (IDCOR) Program.[19] The IDCOR comparison concluded that in general the alternate containment concepts were not cost-beneficial; however, the referenced comparison did not address issues of risk aversion or enhancing public acceptance as part of the cost-benefit evaluation. The NPOVS did not address containment as a sole or separate technology for assuring nuclear viability because at most enhancing containment only addresses one of the NPOVS criteria, namely, that of public health risk (Table 2). Enhancing containment does not appear to either enhance economic competitiveness or necessarily to assure easier licensing.

Recently, however, several papers have appeared which promote the advantages of either passive containment systems[21,22] or containment enhancements to achieve ultrasafety.[12,23] In actual fact, the passive containment system proposed by Nucledyne for PWRs is a very similar conceptual approach to that now being proposed by the participants in the EPRI Small ALWR Program discussed above. The passive containment system relies on the use of multiple cells within containment to isolate portions of the primary system such as the reactor vessel, the pumps, and the steam generators. The passive containment system includes pressure suppression systems (quench tanks) to accommodate pipe breaks or primary system component failures, an elevated pool of borated water which can flood the containment cell in which the primary system pressure boundary pool has failed, diverse heat exchangers for decay heat removal within containment or through the secondary system, and reportedly an enhanced resistance to the effects of earthquakes and missiles because of the multicell configuration. Although the multicell construction is claimed by the proponents to enable access to plant components for replacement, the multicell construction would also apparently be more complicated, and leak tightness of the individual cells would have to be assured to derive benefit over that of the single large containment envisioned for the small advanced PWRs. Such benefit would appear to derive primarily from limiting the amount of containment flooding needed during a worst case accident or possibly produced by an inadvertent containment flooding event. Thorough analysis would be required to adequately assess the cost-benefit of such a system.

The other recent proposal is that of enhancing containment to achieve "ultrasafety" which is essentially defined as totally eliminating the potential for offsite doses which could result from an LWR core melt. The ultrasafe containment is to be achieved by providing a chill-vent filter system to eliminate all possible radioactive releases from an atmospheric pressure containment structure and by providing natural circulation water-cooled core retention devices below grade under the containment. The chill-vent filter system would be constructed as a tower filled with loosely packed rock and activated carbon. The loose rock would be chilled to about -60°C by an onsite refrigeration plant (an active system). Although krypton condenses at -152°C, the large surface area of the loosely packed, chilled rock is claimed to prevent release of radioactive noble gases, and the carbon filter eliminates iodines.

The containment would be connected directly to the chill-vent filter system and maintained at atmospheric pressure. An intervening wet well between the reactor vessel and filter would serve as a pressure suppression mechanism and an initial scrubber of any radioactive releases from normal operation or accidents. During normal operation, the chill-vent would be isolated by louvers or rupture membranes, and atmospheric pressure equilibration would be handled through a smaller vent filter which would be isolated during an accident. The proponents are not concerned with filter icing and thereby plugging following an accident since they claim that this situation would be equally effective at eliminating releases.

The cost of the chill-vent filter and passive core retention devices is estimated by the proponents at about $4-5M as a backfit on existing plants. The IDCOR evaluations of similar systems were much higher starting at about $25M. In terms of the NPOVS criteria (Table 2) for advanced reactor viability, the only perceived advantage of ultrasafe containment systems would be as an add-on to the large ALWRs in order to enhance public acceptance and perhaps thereby simplify licensing by reducing intervenor challenges. Technically, the chill-vent filter would have to be demonstrated by testing to show conclusively that the performance of the chiller and the activated carbon is not readily degraded by inleakage of atmospheric moisture or other external effects such as smoke and pollution. Further, the vent filter structure would have to be rigorously qualified to withstand seismic events, tornados, and other potential disrupting forces. The cost-benefit of potential public acceptance would have to be weighed against developing the technology for the small ALWRs which are attempting to avoid fuel-damaging situations to begin with.

Related Issues

The proponents of safety enhancements often support their respective positions by arguing for some perceived advantage in licensing, economics, or public acceptance which accrues especially to their concept. The proponents of PIUS have proposed building a non-nuclear demonstration test to illustrate both the fundamental safety principles and the operational stability of their overall design. They believe that having such a demonstration, particularly of the safety principles, will support their contention that diverse, redundant, and independent safety systems, such as diesel generators, are not necessary for PIUS. The proponents of the GE Small BWR have made similar arguments but also contend that demonstration testing of individual components and systems can contend that demonstration testing of individual components and systems can be as equally convincing as the expensive approach of testing the whole plant system or larger subsets. The respective proponents of the DOE-sponsored Modular Gas-Cooled Reactor and the Modular Liquid-Metal Reactors, neither of which is being addressed in this paper, are the most ardent supporters of a full nuclear demonstration and safety test of a single module. Thus, the proponents of different concepts have different levels of investment to which they are willing to commit for a safety demonstration. However, all proponents of the passively safe reactors hope to achieve licensing simplification by focusing attention on proving the assured actuation and operation of the few passive features which will in turn assure public health and safety for the worst case accidents and design basis events.

An important advantage of passive safety is the possibility to gain relief from NRC imposition of nuclear quality assurance and other regulatory requirements, such as diversity, redundancy, and independence, for those plant systems which can be eliminated from the safety envelope by reliance on the passive features. Thus, the proponents hope to achieve less voluminous technical specifications, smaller scopes for programs of surveillances and testing, and less record-keeping and documentation in general. However, the proponents also appear to underestimate or at least not to dwell on the potentially large amount of analysis and documentation needed to support NRC acceptance of the safety test results. Safety tests may also not be sufficiently extensive to satisfy all possible NRC concerns and, being performed in advance, may not be truly representative of the system or subsystem which evolves and is ultimately deployed. The latter could potentially raise additional NRC defense-in-depth concerns. Finally, NRC has not yet shelved the requirement for defense-in-depth.[24,25] Therefore, proponents of passively safe reactors need to give appropriate attention to simplification of potential licensing problems.

In addition, prototypical licensing experience may possibly be observed by examining the history of Fort St. Vrain licensing. This history has admittedly been evolutionary because of the changes in the regulatory system that have resulted since licensing of this plant began in the middle 1960s. In the Preliminary Safety Analysis Report, the inherent safety features of Fort St. Vrain were highlighted and frequently cited.[13] Such wording was eliminated by the time of the issuance of the Final Safety Analysis Report three years later. Fort St. Vrain was shown by analysis and supporting test data to accommodate both the very conservatively postulated Design Base Depressurization Accident and the Permanent Loss of Forced Cooling with substantial margin in offsite doses compared to 10 CFR Part 100 limits. By comparison to current LWRs, the Fort St. Vrain design basis events were closer to beyond design basis events.

However, the Fort St. Vrain Technical Specifications and nuclear quality assurance criteria were also required to cover the systems needed to accommodate anticipated transients and less severe accident initiators and thereby to assure very large margins to fuel and component damage in terms of both temperatures reached and response times available. Based on the licensing experience at Fort St. Vrain, it is very hard to conceive that the NRC would or even should eliminate requirements for defense-in-depth for advanced plants, especially since the defense-in-depth philosophy remains a specific tenet cited in the NRC's policy statement (Ref. 24).

Finally, some concept proponents, not necessarily those of the ALWRs discussed here, have proposed that defense-in-depth will still be available simply because of the owner's concern about assuring investment protection by avoiding the possibility of events degenerating to the point of relying solely on passive safety features. The small ALWRs have redundant features as discussed above although the proponents have yet to define whether these are intended to be investment protection features or to fall within the safety-licensing envelope. If the NRC were to be convinced to relenquish effective oversight of such preventive features because of the assured operability of the passive backup, the state public utility commissions and potential intervenors may view this situation under the provisions of 10 CFR Part 8.4 (e) through (i) as allowing the states to exercise jurisdiction over the quality assurance and performance of these preventive features in terms of protecting the rate payer from imprudent actions by the utility. Thus, the utilities may never be relieved of regulatory requirements governing "investment protection" features either because NRC will maintain a defense-in-depth posture consistent with current policy or because the states could assume an economically-based regulatory oversight.

Conclusions

As described in this paper, innovations in reactor safety technology appear to fall into two categories, technical and institutional. This paper has briefly surveyed the technical bases for ALWR design innovation against the NPOVS criteria for viability. Because of the early stages of design conceptualization, a detailed evaluation is not possible at this time. However, the need for technical demonstrations and further analysis have been described based on available design information. Because of their more evolutionary design aspects and the opportunity for demonstration plants being constructed first in Japan in the relative near term, the large ALWRs appear to be the most promising concepts. The small, passively safe ALWRs are very attractive for their innovative enhancements of safety but require greater effort in development and proof testing. This paper has also presented appropriate cautions about the proponents' bases for proposing institutional innovations in safety analysis and licensing. These institutional considerations are as important to the pursuit of design alternatives as are the technical bases for proposed innovations.

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