UNITED STA...

NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

March 21, 1994

The Honorable Ron Klink
United States House of Representatives
Washington, D.C. 20515-3804

Dear Congressman Klink:

This replies to your March 7 communication forwarding Mr. Walston Chubb's letter to you.

As you know, the Nuclear Regulatory Commission was established, at least in part, because of a Congressional perception that there was a basic conflict of interest between the old Atomic Energy Commission's dual roles of developing and promoting the various uses of nuclear energy and, at the same time, regulating them to protect the public health and safety.

Accordingly, the Energy Reorganization Act of 1974 charged the NRC with only one mission--assuring that, if nuclear energy is used for civilian purposes in this country, there is reasonable assurance that the public health and safety and the environment are adequately protected.

The NRC does not have a regulation which says or suggests "...that nuclear power plant accidents can cause prompt deaths ten miles from the plant." Further, in a 1985 Policy Statement, the Commission made it plain that existing nuclear power plants pose no undue risk to public health and safety.

In the following year, the Commission issued a Policy Statement on "Safety Goals for the Operations of Nuclear Power Plants." The document focuses on the risks to the public from nuclear power plant operation and its objective is to establish goals that broadly define an acceptable level of radiological risk. I am enclosing copies of both Policy Statements for your information.

In addition, since the 1979 Three Mile Island accident, the Commission has sponsored an active program in research on severe nuclear power plant accidents as part of a multi-faceted approach to safety which also includes improved plant operations, human factor considerations and probabilistic risk assessment. The results of this work have been or may be used in modifying the Commission's rules or policies in areas such as siting, emergency planning and containment design.

CC52

I hope that this information will be helpful in replying to Mr. Chubb's letter and, as you requested, I am returning your correspondence.

Sincerely,

Dennis K. Rathbun

Dennis K. Rathbun, Director
Office of Congressional Affairs

Enclosures:
As stated

gives some guidance on what the Commission expects of licensee fitness for duty programs. However, I believe that the Commission should have gone further.

Instead merely issuing a policy statement, the Commission should have promulgated a rule. The rule should be a relatively simple, nonprescriptive rule which would do two things. First, it would prohibit anyone who is unfit for duty from being permitted access to vital areas of plants. Second, it would require licensees to have a program and procedures to ensure that no one who is unfit for duty gains access to vital areas. The Commission should then work with the industry to develop guidance on what are the essential elements of an adequate fitness for duty program. There are several reasons why I believe that this would be a better approach.

The most important reason for my preference for a rule and specific guidelines is that a rule is enforceable while a policy statement is not. With a rule the Commission would have a clear basis for enforcement action in all cases in which a utility fails to establish and maintain an effective finess for duty program. The NRC has broad authority under the Atomic Energy Act to take enforcement action by issuing an order should there be an immediate threat to public health and safety. The Commission would also be able to take enforcement action if it could tie a specific safety problem to a lapse in the licensee's fitness for duty program. However, the Commission is unlikely to be able to do so. For example, if a maintenance worker makes a mistake in assembling safety equipment because he is under the influence of drugs or alcohol and equipment later malfunctions, it is unlikely that the true cause of the mistake would be discovered. In fact, the problem would most likely be attributed to some defect in the worker's training. Further, waiting until a specific safety problem surfaces or an immediate threat occurs and then trying to correct the fitness for duty program after the fact is not the best way to ensure that licensees have effective fitness for duty programs. Thus, our general enforcement authority does not provide us with enough flexibility to deal with all potential fitness for duty problems in a timely manner. Absent a specific event, it would not allow us to do much of anything if a licensee simply has not developed or implemented an adequate program. This policy statement represents a continuation of the reactive approach to regulation which has so often failed in the past.

A second reason for my preference for a rule with minimum guidelines is that the policy statement is too amorphous. Even the "specific" guidance the Commission does provide is fairly vague. The policy statement provides little insight into what the Commission considers to be an adequate fitness for duty program or what standard the staff is supposed to use as it monitors the progress of the industry over the next eighteen months.

The Commission should work together with the industry to identify the essential elements of an adequate fitness for duty program. While the policy statement comments favorably upon the EEI guidelines developed by the industry, those guidelines are optional, not mandatory. The utilities can, therefore, pick and choose among the various elements and decide whether to include them in their programs. Moreover, the EEI guidelines themselves are quite general in nature, and are subject to varying interpretations. Absent further guidance on what is an acceptable fitness for duty program, the utilities can and probably will adopt widely differing approaches on such elements as chemical testing and offsite drug use. Not all approaches are likely to be acceptable. The Commission should not wait until 18 months from now, when all the utilities are supposed to have their programs in place, to let the industry know whether the Commission agrees with what they have done. The Commission and the industry ought to decide now which elements are absolutely essential to an adequate program, and then everyone will be working from a common base of understanding.

The Commission and the industry should also establish the specific criteria against which individual licensee programs will be evaluated so that the ground rules for evaluating programs and for monitoring progress will be in place before the 18 month monitoring period begins. Absent such guidelines, it is difficult to see how INPO and NRC staff reviews of these programs will provide any meaningful insights as to their adequacy.

Thus, to ensure enforceability, to set the ground rules in advance and to ensure that all utilities meet at least a minimum set of standards, I believe the Commission should issued a rule and should establish guidance, in cooperation with the industry, on just exactly what are the essential elements of a fitness for duty program.

The additional views of the Commission follow:

The Commission does not share Commissioner Asselstine's great concern about the legally non-binding character of the policy statement per se. The Commission's hands are not tied if it finds inadequate compliance with straight-forward and explicit policy guidelines. The Atomic Energy Act confers broad authority for the Commission to take prompt enforcement action should any licensee facility, in the Commission's judgment, not be operated in a manner that protects the public health and safety. A policy statement, at this juncture, offers the quickest means to achieve the end we all desire.

Dated at Washington, DC, this 30th day of July 1986.

For the Nuclear Regulatory Commission.
Lando W. Zech, Jr.,
Chairman.

51 FR 28044
Published 8/4/86

51 FR 30028
Published 8/21/86
Effective 8/4/86

10 CFR Part 50

Safety Goals for the Operations of Nuclear Power Plants; Policy Statement; Republication

[Editorial Note.—The following document was originally published at page 28044 in the issue of Monday, August 4, 1985. It is being republished in its entirety, with corrections, at the request of the agency.]

AGENCY: Nuclear Regulatory Commission.

ACTION: Policy statement.

SUMMARY: This policy statement focuses on the risks to the public from nuclear power plant operation. Its objective is to establish goals that broadly define an acceptable level of radiological risk. In developing the policy statement, the NRC sponsored two public workshops during 1981, obtained public comments and held four public meetings during 1982, conducted a 2-year evaluation during 1983 to 1985, and received the views of its Advisory Committee on Reactor Safeguards.

The Commission has established two qualitative safety goals which are supported by two quantitative objectives. These two supporting objectives are based on the principle that nuclear risks should not be a significant addition to other societal risks. The Commission wants to make clear that no death attributable to nuclear power plant operation will ever be "acceptable" in the sense that the Commission would regard it as a routine or permissible event. The Commission is discussing acceptable risks, not acceptable deaths.

• The *qualitative safety goals* are as follows:

—Individual members of the public should be provided a level of

protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

—Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

• The following *quantitative objectives* are to be used in determining achievement of the above safety goals:

—The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

—The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

EFFECTIVE DATE: August 4, 1986.

FOR FURTHER INFORMATION CONTACT: Merrill Taylor, Regional Operations and Generic Requirements Staff, Office of the Executive Director for Operations, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301/492–4356).

SUPPLEMENTARY INFORMATION: The following presents the Commission's Final Policy Statement on Safety Goals for the Operation of Nuclear Power Plants:

I. Introduction

A. Purpose and Scope

In its response to the recommendations of the President's Commission on the Accident at Three Mile Island, the Nuclear Regulatory Commission (NRC) stated that it was "prepared to move forward with an explicit policy statement on safety philosophy and the role of safety-cost tradeoffs in the NRC safety decisions." This policy statement is the result.

Current regulatory practices are believed to ensure that the basic statutory requirement, adequate protection of the public, is met. Nevertheless, current practices could be improved to provide a better means for testing the adequacy of and need for current and proposed regulatory requirements. The Commission believes that such improvement could lead to a more coherent and consistent regulation of nuclear power plants, a more predictable regulatory process, a public understanding of the regulatory criteria that the NRC applies, and public confidence in the safety of operating plants. This statement of NRC safety policy expresses the Commission's views on the level of risks to public health and safety that the industry should strive for in its nuclear power plants.

This policy statement focuses on the risks to the public from nuclear power plant operation. These are the risks from release of radioactive materials from the reactor to the environment from normal operations as well as from accidents. The Commission will refer to these risks as the risks of nuclear power plant operation. The risks from the nuclear fuel cycle are not included in the safety goals.

These fuel cycle risks have been considered in their own right and determined to be quite small. They will continue to receive careful consideration. The possible effects of sabotage or diversion of nuclear material are also not presently included in the safety goals. At present there is no basis on which to provide a measure of risk on these matters. It is the Commission's intention that everything that is needed will be done to keep these types of risks at their present very low level; and it is the Commission's expectation that efforts on this point will continue to be successful. With these exceptions, it is the Commission's intent that the risks from all the various initiating mechanisms be taken into account to the best of the capability of current evaluation techniques.

In the evaluation of nuclear power plant operation, the staff considers several types of releases. Current NRC practice addresses the risks to the public resulting from operating nuclear power plants. Before a nuclear power plant is licensed to operate, NRC prepares an environmental impact assessment which includes an evaluation of the radiological impacts of routine operation of the plant and accidents on the population in the region around the plant site. The assessment undergoes public comment and may be extensively probed in adjudicatory hearings. For all plants licensed to operate, NRC has found that there will be no measurable radiological impact on any member of the public from routine operation of the plant. (Reference: NRC staff calculations of radiological impact on humans contained in Final Environmental Statements for specific nuclear power plants: e.g., NUREG–0779, NUREG–0812, and NUREG–0854.)

The objective of the Commission's policy statement is to establish goals that broadly define an acceptable level of radiological risk that might be imposed on the public as a result of nuclear power plant operation. While this policy statement includes the risks of normal operation, as well as accidents, the Commission believes that because of compliance with Federal Radiation Council (FRC) guidance, (40 CFR Part 190), and NRC's regulations (10 CFR Part 20 and Appendix I to Part 50), the risks from routine emissions are small compared to the safety goals. Therefore, the Commission believes that these risks need not be routinely analyzed on a case-by-case basis in order to demonstrate conformance with the safety goals.

B. Development of this Statement of Safety Policy

In developing the policy statement, the Commission solicited and benefited from the information and suggestions provided by workshop discussions. NRC-sponsored workshops were held in Palo Alto, California, on April 1–3, 1981 and in Harpers Ferry, West Virginia, on July 23–24, 1981. The first workshop addressed general issues involved in developing safety goals. The second workshop focused on a discussion paper which presented proposed safety goals. Both workshops featured discussions among knowledgeable persons drawn from industry, public interest groups, universities, and elsewhere, who represented a broad range of perspectives and disciplines.

The NRC Office of Policy Evaluation submitted to the Commission for its consideration a Discussion Paper on Safety Goals for Nuclear Power Plants in November 1981 and a revised safety goal report in July 1982.

The Commission also took into consideration the comments and suggestions received from the public in response to the proposed Policy Statement on "Safety Goals for Nuclear Power Plants," published on February 17, 1982 (47 FR 7023). Following public comment, a revised Policy Statement was issued on March 14, 1983 (48 FR 10772) and a 2-year evaluation period began.

The Commission used the staff report and its recommendations that resulted from the 2-year evaluation of safety goals in developing this final Policy Statement. Additionally, the Commission had benefit of further comments from its Advisory Committee on Reactor Safeguards (ACRS) and by senior NRC management.

Based on the results of this information, the Commission has determined that the qualitative safety goals will remain unchanged from its March 1983 revised policy statement, and the Commission adopts these as its safety goals for the operation of nuclear power plants.

II. Qualitative Safety Goals

The Commission has decided to adopt qualitative safety goals that are supported by quantitative health effects objectives for use in the regulatory decisionmaking process. The Commission's first qualitative safety goal is that the risk from nuclear power plant operation should not be a significant contributor to a person's risk of accidental death or injury. The intent is to require such a level of safety that individuals living or working near nuclear power plants should be able to go about their daily lives without special concern by virtue of their proximity to these plants. Thus, the Commission's first safety goal is—

Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

Even though protection of individual members of the public inherently provides substantial societal protection, the Commission also decided that a limit should be placed on the societal risks posed by nuclear power plant operation. The Commission also believes that the risks of nuclear power plant operation should be comparable to or less than the risks from other viable means of generating the same quantity of electrical energy. Thus, the Commission's second safety goal is—

Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The broad spectrum of expert opinion on the risks posed by electrical generation by coal and the absence of authoritative data make it impractical to calibrate nuclear safety goals by comparing them with coal risks based on what we know today. However, the Commission has established the quantitative health effects objectives in such a way that nuclear risks are not a significant addition to other societal risks.

Severe core damage accidents can lead to more serious accidents with the potential for life-threatening offsite release of radiation, for evacuation of members of the public, and for contamination of public property. Apart from their health and safety consequences, severe core damage accidents can erode public confidence in the safety of nuclear power and can lead to further instability and unpredictability for the industry. In order to avoid these adverse consequences, the Commission intends to continue to pursue a regulatory program that has as its objective providing reasonable assurance, while giving appropriate consideration to the uncertainties involved, that a severe core damage accident will not occur at a U.S. nuclear power plant.

III. Quantitative Objectives Used To Gauge Achievement of The Safety Goals

A. General Considerations

The quantitative health effects objectives establish NRC guidance for public protection which nuclear plant designers and operators should strive to achieve. A key element in formulating a qualitative safety goal whose achievement is measured by quantitative health effects objectives is to understand both the strengths and limitations of the techniques by which one judges whether the qualitative safety goal has been met.

A major step forward in the development and refinement of accident risk quantification was taken in the Reactor Safety Study (WASH-1400) completed in 1975. The objective of the Study was "to try to reach some meaningful conclusions about the risk of nuclear accidents." The Study did not directly address the question of what level of risk from nuclear accidents was acceptable.

Since the completion of the Reactor Safety Study, further progress in developing probabilistic risk assessment and in accumulating relevant data has led to a recognition that it is feasible to begin to use quantitative safety objectives for limited purposes. However, because of the sizable uncertainties still present in the methods and the gaps in the data base—essential elements needed to gauge whether the objectives have been achieved—the quantitative objectives should be viewed as aiming points or numerical benchmarks of performance. In particular, because of the present limitations in the state of the art of quantitatively estimating risks, the quantitative health effects objectives are not a substitute for existing regulations.

The Commission recognizes the importance of mitigating the consequences of a core-melt accident and continues to emphasize features such as containment, siting in less populated areas, and emergency planning as integral parts of the defense-in-depth concept associated with its accident prevention and mitigation philosophy.

B. Quantitative Risk Objectives

The Commission wants to make clear at the beginning of this section that no death attributable to nuclear power plant operation will ever be "acceptable" in the sense that the Commission would regard it as a routine or permissible event. We are discussing acceptable risks, not acceptable deaths. In any fatal accident, a course of conduct posing an acceptable risk at one moment results in an unacceptable death moments later. This is true whether one speaks of driving, swimming, flying or generating electricity from coal. Each of these activities poses a calculable risk to society and to individuals. Some of those who accept the risk (or are part of a society that accepts risk) do not survive it. We intend that no such accidents will occur, but the possibility cannot be entirely eliminated. Furthermore, individual and societal risks from nuclear power plants are generally estimated to be considerably less than the risk that society is now exposed to from each of the other activities mentioned above.

C. Health Effects—Prompt and Latent Cancer Mortality Risks

The Commission has decided to adopt the following two health effects as the quantitative objectives concerning mortality risks to be used in determining achievement of the qualitative safety goals—

- *The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.*
- *The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.*

The Commission believes that this ratio of 0.1 percent appropriately reflects both of the qualitative goals—to provide that individuals and society bear no significant additional risk. However, this does not necessarily mean that an additional risk that exceeds 0.1 percent would by itself constitute a significant additional risk. The 0.1 percent ratio to other risks is low enough to support an expectation that people living or working near nuclear power plants would have no special concern due to the plant's proximity.

The average individual in the vicinity of the plant is defined as the average individual biologically (in terms of age and other risk factors) and locationally who resides within a mile from the plant site boundary. This means that the average individual is found by accumulating the estimated individual risks and dividing by the number of individuals residing in the vicinity of the plant.

In applying the objective for individual risk of prompt fatality, the Commission has defined the vicinity as the area within 1 mile of the nuclear power plant site boundary, since calculations of the consequences of major reactor accidents suggest that individuals within a mile of the plant site boundary would generally be subject to the greatest risk of prompt death attributable to radiological causes. If there are no individuals residing within a mile of the plant boundary, an individual should, for evaluation purposes, be assumed to reside 1 mile from the site boundary.

In applying the objective for cancer fatalities as a population guideline for individuals in the area near the plant, the Commission has defined the population generally considered subject to significant risk as the population within 10 miles of the plant site. The bulk of significant exposures of the population to radiation would be concentrated within this distance, and thus this is the appropriate population for comparison with cancer fatality risks from all other causes. This objective would ensure that the estimated increase in the risk of delayed cancer fatalities from all potential radiation releases at a typical plant would be no more than a small fraction of the year-to-year normal variation in the expected cancer deaths from nonnuclear causes. Moreover, the prompt fatality objective for protecting individuals generally provides even greater protection to the population as a whole. That is, if the quantitiative objective for prompt fatality is met for individuals in the immediate vicinity of the plant, the estimated risk of delayed cancer fatality to persons within 10 miles of the plant and beyond would generally be much lower than the quantitative objective for cancer fatality. Thus, compliance with the prompt fatality objective applied to individuals close to the plant would generally mean that the aggregate estimated societal risk would be a number of times lower than it would be if compliance with just the objective applied to the population as a whole were involved. The distance for averaging the cancer fatality risk was taken as 50 miles in the 1983 policy statement. The change to 10 miles could be viewed to provide additional protection to individuals in the vicinity of the plant, although analyses indicate that this objective for cancer fatality will not be the controlling one. It also provides more representative societal protection, since the risk to the people beyond 10 miles will be less than the risk to the people within 10 miles.

IV. Treatment of Uncertainties

The Commission is aware that uncertainties are not caused by use of quantitative methodology in decisionmaking but are merely highlighted through use of the quantification process. Confidence in the use of probabilistic and risk assessment techniques has steadily improved since the time these were used in the Reactor Safety Study. In fact, through use of quantitative techniques, important uncertainties have been and continue to be brought into better focus and may even be reduced compared to those that would remain with sole reliance on deterministic decisionmaking. To the extent practicable, the Commission intends to ensure that the quantitative techniques used for regulatory decisionmaking take into account the potential uncertainties that exist so that an estimate can be made on the confidence level to be ascribed to the quantitative results.

The Commission has adopted the use of mean estimates for purposes of implementing the quantitative objectives of this safety goal policy (i.e., the mortality risk objectives). Use of the mean estimates comports with the customary practices for cost-benefit analyses and it is the correct usage for purposes of the mortality risk comparisons. Use of mean estimates does not however resolve the need to quantify (to the extent reasonable) and understand those important uncertainties involved in the reactor accident risk predictions. A number of uncertainties (e.g., thermal-hydraulic assumptions and the phenomenology of core-melt progression, fission product release and transport, and containment loads and performance) arise because of a direct lack of severe accident experience or knowledge of accident phenomenology along with data related to probability distributions.

In such a situation, it is necessary that proper attention be given not only to the range of uncertainty surrounding probabilistic estimates, but also to the phenomenology that most influences the uncertainties. For this reason, sensitivity studies should be performed to determine those uncertainties most important to the probabilistic estimates. The results of sensitivity of studies should be displayed showing, for example, the range of variation together with the underlying science or engineering assumptions that dominate this variation. Depending on the decision needs, the probabilistic results should also be reasonably balanced and supported through use of deterministic arguments. In this way, judgements can be made by the decisionmaker about the degree of confidence to be given to these estimates and assumptions. This is a key part of the process of determining the degree of regulatory conservatism that may be warranted for particular decisions. This defense-in-depth approach is expected to continue to ensure the protection of public health and safety.

V. Guidelines For Regulatory Implementation

The Commission approves use of the qualitative safety goals, including use of the quantitative health effects objectives in the regulatory decisionmaking process. The Commission recognizes that the safety goal can provide a useful tool by which the adequacy of regulations or regulatory decisions regarding changes to the regulations can be judged. Likewise, the safety goals could be of benefit in the much more difficult task of assessing whether existing plants, designed, constructed and operated to comply with past and current regulations, conform adequately with the intent of the safety goal policy.

However, in order to do this, the staff will require specific guidelines to use as a basis for determining whether a level of safety ascribed to a plant is consistent with the safety goal policy. As a separate matter, the Commission intends to review and approve guidance to the staff regarding such determinations. It is currently envisioned that this guidance would address matters such as plant performance guidelines, indicators for operational performance, and guidelines for conduct of cost-benefit analyses. This guidance would be derived from additional studies conducted by the staff and resulting in recommendations to the Commission. The guidance would be based on the following general performance guideline which is proposed by the Commission for further staff examination—

Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation.

To provide adequate protection of the public health and safety, current NRC regulations require conservatism in design, construction, testing, operation and maintenance of nuclear power plants. A defense-in-depth approach has been mandated in order to prevent accidents from happening and to mitigate their consequences. Siting in less populated areas is emphasized. Furthermore, emergency response capabilities are mandated to provide additional defense-in-depth protection to the surrounding population.

These safety goals and these implementation guidelines are not meant as a substitute for NRC's regulations and do not relieve nuclear power plant permittees and licensees from complying with regulations. Nor are the safety goals and these implementation guidelines in and of themselves meant to serve as a sole basis for licensing decisions. However, if pursuant to these guidelines, information is developed that is applicable to a particular licensing decision, it may be considered as one factor in the licensing decision.

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

Dated at Washington, DC, this 30th day of July 1986.

For the Nuclear Regulatory Commission.

Lando W. Zech, Jr.,

Chairman.

Additional Views by Commissioner Asselstine on the Safety Goal Policy Statement

The commercial nuclear power industry started rather slowly and cautiously in the early 1960's. By the late 1960's and early 1970's the growth of the industry reached a feverish pace. New orders were coming in for regulatory review on almost a weekly basis. The result was the designs of the plants outpaced operational experience and the development of safety standards. As experience was gained in operational characteristics and in safety reviews, safety standards were developed or modified with a general trend toward stricter requirements. Thus, in the early 1970's, the industry demanded to know "how safe is safe enough." In this Safety Goal Policy Statement, the Commission is reaching a first attempt at answering the question. Much credit should go to Chairman Palladino's efforts over the past 5 years to develop this policy statement. I approve this policy statement but believe it needs to go further. There are four additional aspects which should have been addressed by the policy statement.

Containment Performance

First, I believe the Commission should have developed a policy on the relative emphasis to be given to accident prevention and accident mitigation. Such guidance is necessary to ensure that the principle of defense-in-depth is maintained. The Commission's Advisory Committee on Reactor Safeguards has repeatedly urged the Commission to do so. As a step in that direction, I offered for Commission consideration the following containment performance criterion:

> In order to assure a proper balance between accident prevention and accident mitigation, the mean frequency of containment failure in the event of a severe core damage accident should be less than 1 in 100 severe core damage accidents.

Since the Chernobyl accident, the nuclear industry has been trying to distance itself from the Chernobyl accident on the basis of the expected performance of the containments around the U.S. power reactors. Unfortunately, the industry and the Commission are unwilling to commit to a level of performance for the containments.

The argument has been made that we do not know how to develop containment performance criteria (accident mitigation) because core meltdown phenomena and containment response thereto are very complex and involve substantial uncertainties. On the other hand, to measure how close a plant comes to the quantitative guidelines contained in this policy statement and to perform analyses required by the Commission's backfit rule, one must perform just those kinds of analyses. I find these positions inconsistent.

The other argument against a containment performance criterion is that such a standard would overspecify the safety goal. However, a containment performance objective is an element of ensuring that the principle of defense-in-depth is maintained. Since we cannot rule out core meltdown accidents in the foreseeable future, given the current level of safety, I believe it unwise not to establish an expectation on the performance of the final barrier to a substantial release of radioactive materials to the environment, given a core meltdown.

General Performance Guideline

While I have previously supported an objective of reducing the risks to an as low as reasonably achievable level, the general performance guideline articulated in this policy (i.e., ". . . the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation.") is a suitable compromise. I believe it is an objective that is consistent with the recommendations of the Commission's chief safety officer and our Director of Research, and past urgings of the Advisory Committee on Reactor Safeguards. Unfortunately, the Commission stopped short of adopting this guideline as a performance objective in the policy statement, but I am encouraged that the Commission is willing at least to examine the possibility of adopting it. Achieving such a standard coupled with the containment performance objective given above would go a long way toward ensuring that the operating reactors successfully complete their useful lives and that the nuclear option remains a viable component of the nation's energy mix.

In addition to preferring adoption of this standard now, I also believe the Commission needs to define a "large release" of radioactive materials. I would have defined it as "a release that would result in a whole body dose of 5 rem to an individual located at the site boundary." This would be consistent with the EPA's emergency planning Protective Action Guidelines and with the level proposed by the NRC staff for defining an Extraordinary Nuclear Occurrence under the Price-Anderson Act. In adopting such a definition, the Commission would be saying that its objective is to ensure that there is no more than a 1 in 1,000,000 chance per year that the public would have to be evacuated from the vicinity of a nuclear reactor and that the waiver of defenses provisions of the Price-Anderson Act would be invoked. I believe this to be an appropriate objective in ensuring that there is no undue risk to the public health and safety associated with nuclear power.

Cost-Benefit Analyses

I believe it is long overdue for the Commission to decide the appropriate way to conduct cost-benefit analyses. The Commission's own regulations require these analyses, which play a substantial role in the decisionmaking on whether to improve safety. Yet, the Commission continues to postpone addressing this fundamental issue.

Future Reactors

In my view, this safety goal policy statement has been developed with a steady eye on the apparent level of safety already achieved by most of operating reactors. That level has been arrived at by a piecemeal approach to designing, constructing and upgrading of the plants over the years as experience was gained with the plants and as the results of required research became available. Given the performance of the current generation of plants, I believe a safety goal for these plants is not good enough for the future. This policy statement should have had a separate goal that would require substantially better plants for the next generation. To argue that the level of safety achieved by plant designs that are over 10 years old is good enough for the next generation is to have little faith in the ingenuity of engineers and in the potential for nuclear technology. I would have required the next generation of plants to be substantially safer than the currently operating plants.

Separate Views of Commissioner Bernthal on Safety Goals Policy

I do not disapprove of what has been said in this policy statement, but too much remains unsaid. The public is understandably desirous of reassurance since Chernobyl; the NRC staff needs clear guidance to carry out its responsibilities to assure public health and safety; the nuclear industry needs to plan for the future. All want and deserve to see clear, unambiguous, practical safety objectives that provide the Commission's answer to the question, "How safe is safe enough?" at U.S. nuclear power plants. The question remains unanswered.

It is unrealistic for the Commission to expect that society, for the foreseeable future, will judge nuclear power by the same standard as it does all other risks. The issue today is not so much calculated risk; the issue is public acceptance and, consistent with the intent of Congress, preservation of the nuclear option.

In these early decades of nuclear power, TMI-style incidents must be rendered so rare that we would expect to recount such an event only to our grandchildren. For today's population of reactors, that implies a probability for severe core damage of 10^{-4} per reactor year; for the longer term, it implies something better. I see this as a straightforward policy conclusion that every newspaper editor in the country understands only too well. If the Commission fails to set (and realize) this objective, then the nuclear option will cease to be credible before the end of the century. In other words, if TMI-style events were to occur with 10–15 year regularity, public acceptance of nuclear power would almost certainly fail.

And while the Commission's primary charge is to protect public health and safety, it is also the clear intent of Congress that the Commission, if possible, regulate in a way that preserves rather than jeopardizes the nuclear option. So, for example, if the Commission were to find 100 percent confidence in some impervious containment design, but ignored what was inside the containment, the primary mandate would be satisfied, but in all likelihood, the second would not. Consistent with the Commission's long-standing defense-in-depth philosophy, both core-melt and containment performance criteria should therefore be clearly stated parts of the Commission's safety goals.

In short, this pudding lacks a theme. Meaningful assurance to the public; substantive guidance to the NRC staff; the regulatory path to the future for the industry—all these should be provided by plainly stating that, consistent with the Commission's "defense-in-depth" philosophy:

(1) Severe core-damage accidents should not be expected, on average, to occur in the U.S. more than once in 100 years;

(2) Containment performance at nuclear power plants should be such that severe accidents with substantial offsite damages are not expected, on average, to occur in the U.S. more than once in 1,000 years;

(3) The goal for offsite consequences should be expected to be met after conservative consideration of the uncertainties associated with the estimated frequency of severe core-damage and the estimated mitigation thereof by containment.[1]

The term "substantial offsite damages" would correspond to the Commission's legal definition of "extraordinary nuclear occurrence." "Conservative consideration of associated uncertainties" should offer at least 90 percent confidence (typical good engineering judgment, I would hope) that the offsite release goal is met.

The broad core-melt and offsite-release goals should be met "for the average power plant"; i.e., for the aggregate of U.S. power plants. The decision to fix or not to fix a specific plant would then depend on achieving "the goal for offsite consequences." As a practical matter, this offsite societal risk objective would (and should) be significantly dependent on site-specific population density.

The absence of such explicit population density considerations in the Commission's 0.1 percent goals for offsite consequences deserves careful thought. Is it reasonable that Zion and Palo Verde, for example, be assigned the same theoretical "standard person" risk, even though they pose considerably different risks for the U.S. population as a whole? As they stand, these 0.1 percent goals do not explicitly include population density considerations; a power plant could be located in Central Park and still meet the Commission's quantitative offsite release standard.

I believe the Commission's standards should preserve the important principle that site-specific population density be quantitatively considered in formulating the Commission's societal risk objective: e.g., by requiring that for the *entire* U.S. population, the risk of fatal injury as a consequence of U.S. nuclear power plant operations should not exceed some appropriate specified fraction of the sum of the expected risk of fatality from all other hazards to which members of the U.S. population are generally exposed.

I am further concerned by the arbitrary nature of the 0.1 percent incremental "societal" health risk standard adopted by the Commission, a concept grounded in a purely subjective assessment of what the public might accept. The Commission should seriously consider a more rational standard, tied statistically to the average variations in natural exposure to radiation from all other sources.

Finally, as noted in its introductory comments, the Commission long ago committed to "move forward with an explicit policy statement on safety philosophy and the role of safety-cost tradeoffs in NRC safety decisions." While this policy statement may not be very "explicit", as discussed above, it contains nothing at all on the subject of " 'safety-cost' tradeoffs in NRC safety decisions." For example, *is* $1,000 per person-rem an appropriate cost-benefit standard for NRC regulatory action? While I have long argued that such fundamental decisions are more rightly the responsibility of Congress, the NRC staff continues to use its own ad-hoc judgment in lieu of either the Commission or the Congress speaking to the issue.

In summary, while the Commission has produced a document which is not in conflict with my broad philosophy in such matters, I doubt that the public expected a philosophical dissertation, however erudite. It is a tribute to Chairman Palladino's efforts that the Commission has come this far. But the task remains unfinished.

[1] Interestingly enough, the Commission has adopted proposed goals similar to the above core-melt and containment performance objectives—without clearly saying so. Taken together, the Commission's: (1) 0.1 percent offsite prompt fatality goals; (2) proposed 10^{-4} per-reactor-year "large offsite release" criterion; (3) commitment "to provide reasonable assurance . . . that a severe core-damage accident will not occur at a U.S. nuclear power plant," though they may be ill-defined, can be read to be more stringent than the plainly stated criteria suggested above.

The Commission relies upon several factors in directing the Licensing Boards and, where appropriate, the staff to consider carefully the applicability of § 50.47(c)(1) for the limited period necessary to finalize a response to the recent *GUARD* decision. Because the Commission has not determined how, or even whether, to define what constitutes adequate arrangements for offsite individuals who have been exposed to dangerous levels of radiation, the Commission believes that until it provides further guidance on this matter, Licensing Boards (or, in uncontested matters, the staff) should first consider the applicability of 10 CFR 50.47(c)(1) before considering whether any additional actions are required to implement planning standard (b)(12). Such consideration is particularly appropriate because the *GUARD* decision leaves open the possibility that modification or reinterpretation of planning standard (b)(12) could result in a determination that no prior arrangements need to be made for off-site individuals for whom the consequences of a hypothetical accident are limited to exposure to radiation.

In considering the applicability of 10 CFR 50.47(c)(1), the Licensing Boards (and, in uncontested cases, the staff) should consider the uncertainty over the continued viability of the current meaning of the phrase "contaminated injured individuals." Although, that phrase currently includes members of the offsite public exposed to high levels of radiation, the *GUARD* court has clearly left the Commission the discretion to "revisit" that definition in a fashion that could remove exposed individuals from the coverage of planning standard (b)(12). Therefore, Licensing Boards (and, in uncontested cases, the staff) may reasonably conclude that no additional actions should be undertaken now on the strength of the present interpretation of that term.

Moreover, the Commission believes that Licensing Boards (and, in uncontested cases, the staff) could reasonably find that any deficiency which may be found in complying with a finalized, post-*GUARD* planning standard (b)(12) is insignificant for the purposes of 10 CFR 50.47(c)(1). The low probability of accidents which might cause extensive radiation exposure during the brief period necessary to finalize a Commission response to *GUARD* (as the San Onofre Licensing Board found, the probability of such an accident is less than one in a million per year of operation), and the slow evolution of adverse reactions to overexposure to radiation are generic matters applicable to all plants and licensing situations and over which there is no genuine controversy. Both of those factors weigh in favor of a finding that any deficiencies between present licensee planning (which complies with the Commission's pre-*GUARD* interpretation of 10 CFR 50.47(b)(12)) and future planning in accordance with the final interpretation of planning standard (b)(12) as a response to the *GUARD* decision will not be safety significant for the brief period in which it takes licensee to implement the final standard.

In addition, as a matter of equity, the Commission believes that Licensing Boards (and, in uncontested cases, the staff) could reasonably find that there are "other compelling reasons" to avoid delaying the licensees of those applicants who have complied with the Commission's pre-*GUARD* section 50.47(b)(12) requirements. Where applicants have acted in good faith reliance on the Commission's prior interpretation of its own regulation, the reasonableness of this good faith reliance indicates that it would be unfair to delay licensing while the Commission completes its response to the *GUARD* remand.

Finally, if Licensing Boards find that these factors adequately support the application of 10 CFR 50.47(c)(1), then those Licensing Boards could conclude that no hearings would be warranted. Therefore, until the Commission concludes its *GUARD* remand and instructs its boards and its staff differently, the Licensing Boards could reasonably find that any hearing regarding compliance with 10 CFR 50.47(b)(12) shall be limited to issues which could have been heard before the Court's decision in *GUARD* v. *NRC*.

Dated at Washington, D.C. this 16th day of May, 1985.

For the Commission.

Samuel J. Chilk,

Secretary of the Commission.

50 FR 32138
Published 8/8/85

10 CFR Part 50

Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: Policy statement.

SUMMARY: This statement describes the policy the Commission intends to use to resolve safety issues related to reactor accidents more severe than design basis accidents. Its main focus is on the criteria and procedures the Commission intends to use to certify new designs for nuclear power plants. This policy statement is a revision of the "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation" that was published for comment on April 13, 1983 (48 FR 16014). An advance notice of proposed rulemaking, "Severe Accident Design Criteria," published on October 2, 1980 (45 FR 65474) is being withdrawn by a notice published elsewhere in this issue.

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SUPPLEMENTARY INFORMATION: This policy statement sets forth the Commission's intentions for rulemakings and other regulatory actions for resolving safety issues related to reactor accidents more severe than design basis accidents. The main focus of this statement is on decision procedures involving staff approval or, optionally, Commission certification of new standard designs for nuclear power plants. It also provides guidance on decision and analytical procedures for the resolution of severe accident issues for other classes of future plants and for existing plants (operating reactors and plants under construction for which an operating license has been applied). Severe nuclear accidents are those in which substantial damage is done to the reactor core whether or not there are serious offsite consequences. On October 2, 1980, the Commission issued an advance notice of proposed rulemaking, "Severe Accident Design Criteria," that invited public comment on long-term proposals for treating severe accident issues (45 FR 65474). By another notice published elsewhere in this issue the Commission is withdrawing this advance notice of proposed rulemaking.

This policy statement is a revision of the "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation" published for public comment on April 13, 1983 (48 FR 16014). Twenty-six letters of comment on the proposed policy statement were received. The nuclear industry generally supported the proposed policy statement and suggested several modifications. Much of the criticism of the proposed policy statement by environmental groups and other interested persons focused on a perception of over-reliance on probabilistic risk assessment, especially when coupled with the Commission's "Safety Goal Development Program" (48 FR 10772, March 14, 1983). The Policy Statement was revised as a result of these suggestions and criticisms as well as comments by the Advisory Committee on Reactor Safeguards.

Many changes have already been implemented in existing plants as a result of the TMI Action Plan (NUREG-0660 and NUREG-0737),[1] information resulting from NRC- and industry-sponsored research, and data arising from construction and operating experience. On the basis of currently available information, the Commission concludes that existing plants pose no undue risk to public health and safety and sees no present basis for immediate action on generic rulemaking or other regulatory changes for these plants because of severe accident risk. The Commission has ongoing nuclear safety programs that include: the resolution of new and several other Unresolved Safety Issues and Generic Safety Issues; the Severe Accident Source Term Program; the Severe Accident Research Program; operating experience and data evaluation regarding failure of certain Engineered Safety Features and safety-related equipment, human errors, and other sources of abnormal events; and scrutiny by the Office of Inspection and Enforcement to monitor the quality of plant construction, operation, and maintenance. Should significant new safety information become available, from whatever source, to question the conclusion of "no undue risk," then the technical issues thus identified would be resolved by the NRC under its backfit policy and other existing procedures, including the possibility of generic rulemaking where this is justifiable.

One important source of new information is the experience of NRC and the nuclear industry with plant-specific probabilistic risk assessments. Each of these analyses, which provide a detailed assessment of possible accident scenarios, has exposed relatively unique vulnerabilities to severe accidents. Generally, the undesirable risk from these unique features has been reduced to an acceptable level by low-cost changes in procedures or minor design modifications. Accordingly, when NRC and industry interactions on severe accident issues have progressed sufficiently to define the methods of analysis, the Commission plans to formulate an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possibly significant risk contributors that might be plant specific and might be missed absent a systematic search. Following the development of such an approach, an analysis will be made of any plant that has not yet undergone an appropriate examination and cost-effective changes will be made, if needed, to ensure that there is no undue risk to public health and safety. In implementing such a systematic approach, plants under construction that have not yet received an Operating License will be treated essentially the same as the manner by which operating reactors are dealt with. That is to say, a plant-specific review of severe accident vulnerabilities using this approach is not considered to be necessary to determine adequate safety or compliance with NRC safety regulations under the Atomic Energy Act, or to be a necessary or routine part of an Operating License review for this class of plants.

Regarding the decision process for certifying a new standard plant design—an approach the Commission strongly encourages for future plants—the Policy Statement affirms the Commission's belief that a new design for a nuclear power plant can be shown to be acceptable for severe accident concerns if it meets the following criteria and procedural requirements:

- Demonstration of 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 CP Rule (10 CFR 50.34(f); 47 FR 2286);
- Demonstration of technical resolution of all applicable Unresolved Safety Issues and the medium- and high-priority Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems;
- Completion of a Probabilistic Risk Assessment (PRA) and consideration of the servere accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety; and
- Completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by PRA.

Custom designs that are variations of the present generation of LWRs will be reviewed in future construction permit applications under the guidelines identified for approval or certification of standard plant designs.

Because this policy statement is just one part of a larger program, including the Severe Accident Research Program, for resolving severe accident issues, the NRC staff is publishing concurrently with this Policy Statement a report on "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulation" (NUREG-1070). In this report the Policy Statement is reprinted along with other information and appendices that provide perspective on the development and implementation of this policy and how it relates to other features of the Severe Accident Program. A copy of NUREG-1070 will be available for inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C. Copies of NUREG-1070 may be purchased by calling (202) 275-2060 or (202) 275-2171 or by writing to the Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, D.C. 20013-7082 or the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161

Policy Statement

A. Introduction

The focus on severe accident issues in this Policy Statement is prompted by the staff's judgment that accidents of this class, which are beyond the substantial coverage of design basis events, constitute the major risk to the public associated with radioactive releases from nuclear power plant accidents. A fundamental objective of the Commission's severe accident policy is that the Commission intends to take all reasonable steps to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur.

On April 13, 1983, the U.S. Nuclear Regulatory Commission issued for public comment a "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation" (48 FR 16014). The public comments have been reviewed, and, on the basis of further study and consultation, the Commission is issuing the present Policy Statement as a guide to regulatory decision making on the treatment of severe accident issues for existing and future nuclear reactors[2] with special focus on procedures for staff approval or, optionally, Commission certification of new standard plant designs.[3]

In line with its legislative mandate to ensure that nuclear power plants should

[1] Documents referenced in this Policy Statement are available for inspection at the NRC's Public Document Room, 1717 H Street, NW, Washington, D.C.

[2] The term "nuclear reactor" is commonly used as a synonym for a nuclear power plant which, in addition to the Nuclear Steam Supply System, includes facilities and equipment denoted as Balance-of-Plant.

[3] For forward referenceability of a new standard design, the applicant is being afforded in this Policy Statement the flexibility of choosing between a Preliminary Design Approval (PDA), a Final Design Approval (FDA), or Design Certification (DC). The design approvals (i.e., a PDA or FDA) would be issued following the completion of the staff's review and would be subject to challenge in individual licensing hearings. The Design Certification would be issued by the Commission following a rulemaking proceeding and could not be challenged in individual hearings.

pose no undue risk to public health and safety, the Commission has examined an extensive range of technical issues relating to severe accident risk that have been identified since the accident at Three Mile Island. Following implementation of numerous modifications of plant design and regulatory procedures as developed through the TMI Action Plan (NUREG–0660 and NUREG–0737) and other Commission deliberations, the Commission concludes (based on current information and analyses) that existing plants do not pose an undue level of risk to the public. On this basis, the Commission feels there is no need for immediate action on generic rulemaking or other regulatory changes for these plants because of severe accident risk. However, the occurrence of a severe accident is more likely at some plants than at others. At each plant there will be systems, components or procedures that are the most significant contributors to severe accident risk. The intent of this policy statement is to provide utilities with basis for development of Commission guidance that will allow identification of these contributors and development of the appropriate course of action, as needed to assure acceptable margins of safety. In all cases, the commitment of utility management to the pursuit of excellence in risk management is of critical importance. The term "risk management" includes accident prevention, accident management to curtail or retard its progression, and consequence mitigation to further limit its effects on public health and safety. The Commission plans to formulate an approach for a systematic safety examination of existing plants to determine whether particular accident vulnerabilities are present and what cost-effective changes are desirable to ensure that there is no undue risk to public health and safety. In implementing such a systematic approach, plants under construction that have not yet received an Operating License will be treated essentially the same as the manner by which operating reactors are dealt with. That is to say, a plant-specific review of severe accident vulnerabilities using this approach is not considered to be necessary to determine adequate safety or compliance with NRC safety regulations under the Atomic Energy Act, or to be a necessary or routine part of an Operating License review for this class of plants.

The main purposes of this Policy Statement follow:

• To clarify the procedures and requirements for licensing a new nuclear plant;

• To re-examine the need for the generic rulemaking proceeding contemplated in the TMI Action Plan commitment (NUREG–0660, Task II.B.8) on degraded core accidents, currently referred to as severe nuclear reactor accidents;

• To avoid unnecessary delays of plants now under construction;

• To close out for now severe accident issues for existing plants (those in operation and under construction) without imposing further backfits unless this can be justified by new safety information; and,

• To achieve improved stability and predictability of reactor regulation in a manner that would merit improved public confidence in our regulatory decision making.

The policies presented in this statement will lead to amendment of NRC regulations, standard review plans for licensing actions, or other decision procedures and criteria as part of NRC's ongoing Severe Accident Program. This Policy Statement makes allowance for such changes as the result of the development of new safety information of significance for design and operating procedures.

In accordance with the activities, views, and policy developments discussed in this policy Statement, the Commission believes that it is possible to complete its ongoing reviews of new plant designs with an expectation of fully resolving the severe accident questions in the course of the review. This belief is predicated on the availability of results from the ongoing NRC, Industry Degraded Core Rulemaking Program (IDCOR), and vendor research and insights from the Zion, Indian Point, Limerick, and other risk analyses. The review of standard designs for future CPs provides incentive to industry to address severe accident phenomena. Indeed, since July 1983, the staff has completed the reviews and has issued Final Design Approvals (FDAs) for two standard designs (General Electric Company's BWR/6 Nuclear Island Design, GESSAR II; and Combustion Engineering Incorporated's System 80 Design, CESSAR). A severe accident review by the NRC staff of the GESSAR II design for forward referenceability is nearly complete. The review included assessment of alternative design changes for severe accident risk reduction. In addition, the staff has been involved with pretendering review of an application for Westinghouse Electric Corporation's advanced pressurized water reactor design RESAR–SP/90. In January 1984, the NRC found the RESAR–SP/90 application for a Preliminary Design Approval acceptable for docketing and in May 1984 the application was docketed. Also, work has been continuing between NRC and the Electric Power Research Institute (EPRI) on their "LWR standardized Future Plant Design Evaluation Program."

It is assumed in this Policy Statement that, over the next 10 to 15 years, utility and commercial interest in the United States will focus on advanced light water reactors that involve improvements but are essentially based on the technology that was demonstrated in the design, construction, and operation of more than 100 of these plants in the United States. This policy should not be viewed as prejudicial to more extensive changes in reactor designs that might be demonstrated during or beyond that time period. Indeed, the Commission encourages the development and commercialization of any standard designs that might realize safety benefits, such as those achieved through greater simplicity; slower dynamic response to upset conditions involving accident precursor events; passive heat removal for loss-of-coolant accidents; and other characteristics that promote more efficient construction, operation, and maintenance procedures to enhance safety, reliability, and economy.

B. Policy for New Plant Applications

1. Introduction

No new commercial nuclear reactors have been ordered in the United States since December 1978. However, the Commission has received several applications for reference design approvals that are currently under review. A reference design is one of the options in the Commission's standardization policy. When approved by the NRC staff, a reference design could be incorporated by reference in a new CP application and, ultimately, in an Operating License (OL) application. During the corresponding CP and OL reviews, the NRC staff would not duplicate that portion of its review encompassed by its reference design approval. Therefore, even in the absence of new CP applications, in order to provide guidelines for the current reference design reviews, the Commission has recognized the need to promptly establish the criteria by which new designs can be shown to be acceptable in meeting severe accident concerns. The Commission now believes that there exists an adequate basis from which to establish an appropriate set of criteria. This belief is supported by current operating reactor experience, ongoing severe accident research, and insights from a variety of risk analyses. The resultant criteria and procedural requirements are listed below.

2. Criteria and Procedural Requirements

The Commission believes that a new design for a nuclear power plant (as well as a proposed custom plant) can be shown to be acceptable for severe accident concerns if it meets the following criteria and procedural requirements:

a. Demonstration of 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 CP Rule [10 CFR 50.34(f)];

b. Demonstration of technical resolution of all applicable Unresolved Safety Issues and the medium- and high-priority Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems;

c. Completion of a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety; and

d. Completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by PRA.

The fundamental criteria listed above apply to the staff's review of any new design. In addressing criteria (b) and (c), the applicant for approval or certification of a reference design shall consider a range of alternatives and combination of alternatives to address the unresolved and generic safety issues and to search for cost-effective reductions in the risk from severe accidents. No cost-benefit standard has currently been certified by the Commission, although one has been proposed for trial use (NUREG–0880, Rev. 1). Such a standard, if certified, could serve as a surrogate, not only for dollar costs and benefits of a decision option, but also for other adverse and beneficial effects (soft attributes) of social significance that cannot readily be quantified in commensurate units.

The following sections explain in more detail how these criteria are to be applied to the various types of reviews that the staff may encounter. It is intended that a new design would satisfy each of the fundamental criteria listed above before *final* approval or certification. It is recognized, however, that a new design can go through different stages or levels of approval before receiving this final approval or certification. For example, a reference design can obtain a Preliminary Design Approval (PDA) and then a Final Design Approval (FDA). The unique circumstances of each design review will, therefore, require flexibility in the application of the criteria listed above. In particular, the timing of the PRA requirement may differ considerably from one review to another. In addition, the licensee is required to ensure that the intent of the safety requirements is accomplished during procurement, construction and operation.

It is recognized that there are a diversity of PRA methods. These will continue to undergo evolutionary development as the results of research programs and reliability data from operating reactors become available and as innovative uses of PRA in safety decision contexts suggest better ways to achieve the benefits of these methods while guarding against their limitations or improper uses. While learning curves of these kinds will likely continue for a decade or more, it would nevertheless be constructive to consolidate this experience at various stages of PRA development and utilization. At the present stage of development, a number of positive uses of PRAs have been demonstrated, especially in identifying: (1) Those contributors to servere accident risk that are clearly dominant and hence need to be examined for cost-effective risk reduction measures and (2) those accident sequences that are clearly insignificant risk contributors and can therefore be prudently dismissed. In-between cases are more problematic.

Accordingly, within 18 months of the publication of this severe accident statement, the staff will issue guidance on the form, purpose and role that PRAs are to play in severe accident analysis and decision making for both existing and future plant designs and what minimum criteria of adequacy PRAs should meet. From experience to date, it is evident that PRAs could serve as a highly useful tool in assessing the risk-reduction potential and cost-effectiveness of a number of imaginative design options for new plants in comparison with design features of existing plants. The PRA guidance will describe the appropriate combination of deterministic and probabilistic considerations as a basis for severe accident decisions.

The proposed Commission Policy Statement on Severe Accidents issued on April 13, 1983 recognizes the need for striking a balance between accident prevention and consequence mitigation. In exploring the need for additional design or operational features in the next generation of plants to mitigate the consequences of core-melt accidents, the commission will strike a balance between accident prevention and consequence mitigation encompassing actions that improve understanding of containment building failure characteristics and design features or emergency actions that decrease the likelihood of containment building failures. Although not specifically designed to accommodate all of the hostile environments resulting from the complete spectrum of severe accidents, they can contain a large fraction of the radiological inventory from a portion of the spectrum of such severe accidents. For example, large, dry containments may be sufficiently capable of mitigating the consequences of a wide spectrum of core-melt accidents; hence, further requirements may be unnecessary or, at most, upgrading current requirements to gain limited improvements of their existing capability may be necessary. The Commission expects that these matters will continue to be subjects for study (e.g., in the NRC research program and in further plant-specific studies such as the Zion and Indian Point probabilistic risk assessments).

Integrated systems analysis will be used to explore whether other containment types exhibit a functional containment capability equivalent to that of large, dry containments. Although containment strength is an important feature to be considered in such an analysis, credits should also be given to the inherent energy and radionuclide absorption capabilities of the various designs as well as other design features that limit or control combustible gases.

It is clear that core-melt accident evaluations and containment failure evaluations should continue to be performed for a representative sample of operating plants and plants under construction and for all future plant designs. These studies should improve our understanding of the containment loading and failure characteristics for the various classes of facilities. The analyses should be as realistic as possible and should include, where appropriate, dynamic and static loadings from combustion of hydrogen and other combustibles, static pressure and temperature loadings from steam and non-condensibles, basemat penetration by core-melt materials, and effects on aerosols on engineered safety features. A clarification of containment performance expectations will be made including a decision on whether to establish new performance criteria for containment systems and, if so, what these should be.

The Commission also recognizes the importance of such potential contributors to severe accident risk as human performance and sabotage. 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. Likewise, the effectiveness of human performance will be emphasized in design and operating procedure development. A balanced focus will be paid to the negative impact of human performance on severe accident risk as well as its potentially positive contribution to halting or limiting the consequences of severe accident progression. Design features should be emphasized that reduce the risk of early containment failure, thus providing more time for the positive contributions of operator performance in curtailing severe accident consequences. Also, design features should be given special attention that serve to decrease the role of human error in the sequence of events leading to the initiation or aggravation of core degradation. In particular, methods of analysis and associated data bases are under development by the Commission's ongoing severe accident programs that will aid the analyses and corrective actions of both negative and positive human performance contributions to severe accident risk or its alleviation.

It is noted that some of the severe accident scenarios result in insignificant probability of offsite consequences, because of containment effectiveness. In this situation, there may be no clear basis for regulatory action because there is no substantial effect on public health or safety. However, the implementation of requirements to control occupational exposure should be considered along with the relatively small effects on public health and safety for these types of severe accidents. The resolution of cost-benefit issues in severe accident decision making is part of the NRC's Safety Goal Evaluation Program.

Although in the licensing of existing plants the Commission has determined that these plants pose no undue risk to public health and safety, this should not be viewed as implying a Commission policy that safety improvements in new plant designs should not be actively sought. The Commission fully expects that vendors engaged in designing new standard (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs. This expectation is based on:

• The growing volume of information from industry and government-sponsored research and operating reactor experience has improved our knowledge of specific severe accident vulnerabilities and of low-cost methods for their mitigation. Futher learning on safety vulnerabilities and innovative methods is to be expected.

• The inherent flexibility of this Policy Statement (that permits risk-risk tradeoffs in systems and sub-systems design) encourages thereby innovative ways of achieving an improved overall systems reliability at a reasonable cost.

• Public acceptance, and hence investor acceptance, of nuclear technology is dependent on demonstrable progress in safety performance, including the reduction in frequency of accident precursor events as well as a diminished controversy among experts as to the adequacy of nuclear safety technology.

• Further progress in severe accident risk reduction is a hedge against the possibility that current risk estimates with their broad ranges of uncertainty might unwittingly have been optimistically biased.

• Although the severe accident risk of an individual plant may be acceptable in terms of its direct offsite regional consequences for public health and safety, the aggregate probability (say, over a 30-year period) that one severe accident will occur in a large population of reactors holds a separate and additive significance. Such an event would yield adverse spillover consequences for innocent parties in other regions (i.e., nuclear-oriented utilities and their customers), not to mention a changed political environment for nuclear regulation itself affecting resource costs and programmatic activities.

3. Application of Criteria for Different Types of OL and CP Applications

a. *Application of Certification of Reference Designs with No Previous FDA.* In accordance with the Commission's standardization regulations and policy, a new reference design can be submitted for approval, first as a preliminary design and then as final design. Correspondingly, the staff will issue a Preliminary Design Approval and a Final Design Approval. A PDA is not, however, a prerequisite for an FDA. An applicant has the option to submit FDA-level information initially and proceed directly with an FDA review. These options remain unchanged by this Policy Statement.

After a PDA application is docketed, the preliminary design can be referenced in a new CP application. The corresponding OL application would than reference the approved final design (FDA). Of course, an approved design could also be referenced in a new CP application.

The use of an approved standard design in new CP/OL applications has received considerable attention under the Commission's legislative initiatives on single-step licensing. It should be noted that a two-step review process for a standard design approval is not, in itself, inconsistent with single-step licensing. To be most effective, single-step licensing presumes the existence of a previously approved design—essentially an FDA. This design could still be approved in a two-step process as long as both steps were completed in advance of the single-step licensing application.

The use of PRA in a two-step review process also raises a number of questions. Of particular concern is the timing of the PRA requirement because the completion of a comprehensive and detailed PRA may not be achievable in the absence of essentially complete and final detailed design information. Therefore, to require a complete PRA at the PDA stage would not be realistic. The Commission's recent experience, however, indicates that a substantial amount of design detail that would permit meaningful, limited, quantitative risk analysis does exist at the PDA stage. Because the Commission believes that risk analysis of this type would be a useful design tool, the Commission expects that it would be completed as part of the PDA application process. A complete risk analysis would not be a prerequisite for issuance of a PDA. However, if this risk analysis is not performed in the PDA process, it will have to be provided as part of any CP application referencing the design.

If the scope of the FDA reference design application is limited to an extent that would preclude the completion of a meaningful, comprehensive PRA, the requirement for a complete PRA may be waived. However, the applicant should still perform and submit supplementary risk analysis, to the extent practical, to demonstrate the adequacy of the proposed design. If a comprehensive PRA is not submitted for an FDA, a CP/OL applicant referencing the approved design would be required to submit a plant-specific PRA. For standard design approvals of restricted scope, additional limitations beyond the PRA aspects may exist. Use of such a standard design by the license applicant may be limited by its very nature to a two-step licensing process, namely, a Construction Permit and an Operating License issued separately. This would negate some of the benefits envisioned for an approved or certified design wherein a previously approved site could be matched with it in a one-step, combined CP/OL process.

The reference design must satisfy each of the criteria stated in Section B.2 before an FDA can be issued. For forward referenceability of a new standard design, the applicant is being afforded in this Policy Statement the flexibility of choosing between a Preliminary Design Approval (PDA), a Final Design Approval (FDA), or a Design Certification (DC). The design approvals (i.e., a PDA or FDA) would be issued following the completion of the staff's review and would be subject to challenge in individual licensing hearings. The Design Certification would be issued by the Commission following a rulemaking proceeding and could not be challenged in individual hearings. CPs or OLs, based on a reference design that has not been approved through rulemaking, shall be subject to any design changes arising from the rulemaking proceeding in accordance with the Commission's backfit policy and regulations. The design certification would be issued for a longer duration than a design approval. The specific requirements and procedures for obtaining design certifications or approvals will be established in a forthcoming revision to the Commission's Standardization Policy Statement.

b. Approval or Certification of Reference Designs Previously Granted an FDA. In 1983, the NRC staff issued two Final Design Approvals for reference designs. These designs were permitted to be incorporated by reference in OL applications where the corresponding CP application had referenced the PDA. However, the designs were not approved for incorporation in new CP applications. The Commission now believes that these designs are suitable for use in new CP and OL applications under the conditions specified below. Any significant changes to these designs, other than those resulting from the severe accident review, will require the designs to be considered under the provisions of Section B.3.a, i.e., as new designs.

(1) Each of the two reference design applicants with existing FDAs must request that their FDAs be amended to permit their designs to be referenced in new CP and OL applications. The request must either (i) include the information needed to satisfy each of the criteria stated in Section B.2, or (ii) provide suitable interface requirements to ensure that CP and OL applications referencing the design will satisfy each of the criteria in Section B.2. Requests in either case need not include an evaluation of how the design conforms to the Standard Review Plan (10 CFR 50.34(g)).

In the first case, the staff will amend the existing FDA upon receipt of the request to permit the design to be referenced in new CP and OL applications until the severe accident review is completed. The severe accident review must be successfully completed prior to the issuance of any new CP or OL whose applications reference the design. Upon the successful completion of the severe accident review, the staff will further amend the FDA to permit the design to be referenced in new CP and OL applications for a fixed period of time, such as five years.

In the second case, the staff will amend the existing FDA upon receipt of the request to permit the design to be referenced in new CP and OL applications for a fixed period of time, such as five years. The amended FDA will be conditioned as appropriate to ensure that new CP and OL applications referencing the design will satisfy each of the criteria in Section B.2. The severe accident review must be completed prior to the issuance of the new CP or OL.

(2) Criterion B.2.c requires the completion of a comprehensive PRA. If a comprehensive PRA cannot be completed owing the the limited scope of the design, the applicant shall perform supplementary risk analyses to the extent practical in support of the approval or rulemaking process. As noted above, the limited scope of plant design and PRA analysis would lead to a partial loss of benefits in that a two-step CP/OL licensing process would be required in lieu of a one-step process.

(3) With regard to completion of a comprehensive PRA for a reference design, the Commission recognizes that a PRA would be more meaningful if it were based on a substantial portion of the complete facility design. Therefore, if justified to the NRC staff, completion of the PRA by the FDA applicant may be waived. If a comprehensive PRA is not submitted by the FDA applicant for the FDA, a CP/OL applicant referencing the design would be required to submit a plant-specific PRA.

A reference design applicant previously granted an FDA can pursue the same options of design approval or design certification as described in the preceding section for reference designs with no previous FDA. The FDA would be issued following the completion of the staff's review and would be subject to challenge in individual licensing hearings. The Design Certification would be issued by the Commission following a rulemaking proceeding and could not be challenged in individual hearings. CPs or OLs, based on a reference design that has not been approved through rulemaking, shall be subject to any design changes arising from the rulemaking proceeding in accordance with the Commission's backfit policy and regulations. The design certification would be issued for a longer duration than a design approval. The specific requirements and procedures for obtaining design certifications or approvals will be established in a forthcoming revision to the Commission's Standardization Policy Statement.

c. A Reactivated Construction Permit Application. Because of the many complex factors involved, the criteria and procedures for regulatory treatment of reactivated Construction Permits will be a matter of separate consideration apart from this Severe Accident Policy Statement.

d. A New Custom Plant Construction Permit Application. It is the Commission's policy to encourage the use of reference designs in future CP applications. This does not, however, preclude the use of a custom design. Custom designs shall also be reviewed against the criteria identified in Section B.2. As a result of the circumstances and timing involved in the ongoing standard design review processes, the Commission expects that most, it not all, new CP applications incorporating a reference design would be based on essentially final design information. This will result in improved safety and regulatory practices, as well as reduced time to license and construct a nuclear power plant. To obtain as much of this benefit as practicable for a custom design application, the Commission will require a CP application for a custom design to include design information that is sufficiently final and complete to permit completion of an adequate plant-specific PRA. It is possible, however, that an applicant referencing an approved or certified design in lieu of a custom plant would have in prospect a significantly reduced licensing fee since staff effort would not be required—or much less would be required—for a rereview of the approved or certified design at the CP/OL stage save for those detailed changes to accommodate unique site features or other special circumstances (e.g., innovative equipment designs to meet new ASME or IEEE codes, etc.)

C. Policy for Existing Plants

1. Some General Principles of Policy Development

The Commission has licensed about 90 nuclear plants and expects to process applications to license approximately 30 additional plants. The Commission has considered at length the question of whether generic rulemaking should be undertaken or additional regulations should be issued at this time to require more capability in operating plants or plants under construction to improve severe accident prevention, consequence mitigation, or accident management that would halt or delay further core degradation.

The TMI accident led to a number of investigations of the adequacy of design features, operating procedures, and personnel of nuclear power plants to provide assurance of no undue risk regarding severe reactor accidents. The report "NRC Action Plan Developed as a Result of the TMI-2 Accident" (NUREG-

0660, May 1980) describes a comprehensive and integrated plan involving many actions that serve to increase safety when implemented by operating plants and plants under construction. The Commission approved items for implementation and these are identified in a report, "Clarification of TMI Action Plan Requirements" (NUREG–0737, November 1980). The staff issued further criteria on emergency operational facilities (NUREG–0737, Rev. 1), auxiliary feedwater system improvements (derived from NUREG–0667), and instrumentation (Regulatory Guide 1.97, Revision 2).

The TMI Action Plan led to the requirements of over 6,400 separate action items for operating reactors and five Near-Term Operating Licenses. About 90 percent of the action items approved for operating reactors are now complete and the remainder are expected to be finished by the end of fiscal year 1985. There were 132 different types of action items approved in the Action Plan (an average of 90 actions per plant). Of this total, 39 involved equipment backfit items, 31 involved procedural changes, and 62 required analyses and reports. It is impractical to quantify all of the safety improvements obtained by these many changes. Nevertheless, the cumulative effect is undoubtedly a significant improvement in safety.

Other information from NRC- and industry-sponsored research along with failure data from construction and operating experience have led to changes in existing plants. Also, the NRC/AEC has sponsored 11 plant-specific PRAs and the industry has sponsored many more. The evaluation of severe accident risk by the interrelated deterministic and probabilistic methods has identified many refinements of current design and operating practice that are worthwhile, but has identified no need for fundamental (or major) changes in design.

On the basis of currently available information, the Commission concludes that existing plants pose no undue risk to public health and safety and sees no present basis for immediate action on generic rulemaking or other regulatory changes for these plants because of severe accident risk. Moreover, the Commission has ongoing programs (described in NUREG–1070 and issued concurrently with this Policy Statement) that include: the resolution of Unresolved Safety Issues and other Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of both AC and DC electrical supply systems; the Severe Accident Source Term Program; the Severe Accident Research Program; operating experience and data evaluation regarding equipment failure, human errors, and other sources of abnormal events; and scrutiny by the Office of Inspection and Enforcement to monitor the quality of plant construction, operation, and maintenance. The Commission will maintain its vigilance in these programs to offset the uncertainty of whether significant safety issues remain to be disclosed. Industry research and foreign reactor experience are also meaningful sources of information.

One important source of new information is the experience of NRC and the nuclear industry with plant-specific probabilistic risk assessments is that each of these analyses, which provide a more detailed assessment of possible accident scenarios, has exposed relatively unique vulnerabilities to severe accidents. Generally, the undesirable risk from these unique features has been reduced to an acceptable level by low-cost changes in procedures or minor design modifications. Accordingly, when NRC and industry interactions on severe accident issues have progressed sufficiently to define the methods of analysis, the Commission plans to formulate an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributors (sometimes called "outliers") that might be plant specific and might be missed absent a systematic search. Following the development of such an approach, an analysis will be made of any plant that has not yet undergone an appropriate examination. The examination will include specific attention to containment performance in striking a balance between accident prevention and consequence mitigation. In implementing such a systematic approach, plans under construction that have not yet received an Operating License will be treated essentially the same as the manner by which operating reactors are dealt with. That is to say, a plant-specific review of severe accident vulnerabilities using this approach is not considered to be necessary to determine adequate safety or compliance with NRC safety regulations under the Atomic Energy Act, or to be a necessary or routine part of an Operating License review for this class of plants.

Should significant new safety information develop, from whatever source, which brings into question the Commission's conclusion that existing plants pose no undue risk, then at that time the specific technical issues suggesting undue vulnerability will undergo close examination and be handled by the NRC under existing procedures for issue resolution including the possibility of generic rulemaking where this is justifiable. However, NRC's experience suggests that safety issues discovered through operating experience programs, quality assurance programs or safety analyses often pertain to *unique* characteristics of a specific plant design and, therefore, are dealt with through plant-specific modifications of relatively modest cost rather than major *generic* design changes.

The Severe Accident Research Program as well as NRC's extensive severe accident studies of certain individual plants will aid in determining the extent to which carefully analyzed reference plants can appropriately serve as surrogates for a class of similar plants as the basis for any generic conclusions. These studies will also aid in identifying the desirable scope and approach for follow-up safety studies of individual plants. Any generic changes that are identified as necessary for public health and safety will be required through rulemaking and will be consistent with the Commission's backfit policy.

2. Policy for Operating Reactors

In light of the above principles and conclusions, the Commission's policy for operating reactors includes the following guidance:

- Operating nuclear power plants require no further regulatory action to deal with severe accident issues unless significant new safety information arises to question whether there is adequate assurance of no undue risk to public health and safety.
- In the latter event, a careful assessment shall be made of the severe accident vulnerability posed by the issue and whether this vulnerability is plant or site specific or of generic importance.
- The most cost-effective options for reducing this vulnerability shall be identified and a decision shall be reached consistent with the cost-effectiveness criteria of the Commission's backfit policy as to which option or set of options (if any) are justifiable and required to be implemented.
- In those instances where the technical issue goes beyond current regulatory requirements, generic rulemaking will be the preferred

solution. In other cases, the issue should be disposed of through the conventional practice of issuing Bulletins and Orders or Generic Letters where modifications are justified through backfit policy, or through plant-specific decision making along the lines of the Integrated Safety Assessment Program (ISAP) conception.[4]

• Recognizing that plant-specific PRAs have yielded valuable insight to unique plant vulnerabilities to severe accidents leading to low-cost modifications, licensees of each operating reactor will be expected to perform a limited-scope, accident safety analysis designed to discover instances (i.e., outliers) of particular vulnerability to core melt or to unusually poor containment performance, given core-melt accidents. These plant-specific studies will serve to verify that conclusions developed from intensive severe accident safety analyses of reference or surrogate plants can be applied to each of the individual operating plants. During the next two years, the Commission will formulate a systematic approach, including the development of guidelines and procedural criteria, with an expectation that such an approach will be implemented by licensees of the remaining operating reactors not yet systematically analyzed in an equivalent or superior manner.

3. Policy for Operating License Applications for Plants Currently Under Construction

The same severe accident policy guidance applies to applications for operating licenses (OLs) as stated above for operating nuclear power plants along with the following additional item. (This item also applies to any hearing proceedings that might arise for an operating reactor.)

• Individual licensing proceedings are not appropriate forums for a broad examination of the Commission's regulatory policies relating to evaluation, control and mitigation of accidents more severe than the design basis (Class 9). The Commission has announced a policy regarding Class 9 environmental reviews and hearings in its Statement of Interim Policy on "Nuclear Power Plant Accident Considerations Under the National Environmental Policy Act of 1969" (45 FR 40101, June 13, 1980), and expects to continue this policy. The environmental issues deal essentially with the estimation and description of the risk of severe accidents. The Commission believes that considerations which go beyond that to the possible need for safety measures to control or mitigate severe accidents in addition to those required for conformance with the Commission's safety regulations or conformance with the Clarification of TMI Action Plan Requirements,[5] should not be addressed in case-related safety hearings.

The Separate Remarks of Chairman Palladino and the Dissenting Views of Commissioner Asselstine are attached.

Dated at Washington, D.C., this 30th day of July 1985.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

Separate Remarks by Chairman Palladino

I believe the Commission is on the right course with this decision. The severe accident policy statement presented here is based on the arguments contained within it, the additional support of more detailed analysis in its companion document NUREG–1070, the massive support of the many other related works of this agency and others in this field, and a logical consistency with other actions of the Commission.

In simple terms, this policy statement says that existing plants pose no undue risk to public health and safety, and that there is no present basis for regulatory changes for these plants due to severe accident risk. This conclusion on reactor safety does not lead us to dismantle our regulatory program; rather we are maintaining a vigorous program of surveillance, analysis, and evaluation to foresee possible causes of accidents and prevent them. In this perspective, the Commission has ongoing nuclear safety programs that include: unresolved safety issues; severe accident, source term and research programs; operating experience and data evaluation, and the scrutiny of plant construction, operation and maintenance. Should significant new safety information become available, from whatever source, to question the conclusion of no undue risk, then the technical issues thus identified would be resolved by the NRC under its backfit policy or other existing procedures.

The level of risk found to be acceptable is well documented in the basic works of the agency on these related subjects. The calculated frequency of severe core damage, whether mean or median value, is on the order of 1 chance in 10,000 per reactor year. For most plants, only a fraction of the calculated severe core damage sequences are likely to progress to large scale core melt. Until now, few analysts have even tried to take that fraction into separate consideration, preferring even to refer to the previously calculated value as the core melt frequency. Of the core melt sequences, typically only 1 in 10, or less, are expected to yield large releases of radioactive material. On virtually every reactor site in the United States conditions are such that, even with a large release, there is only 1 chance in 10 of any early fatality—and so on. Thus, the wealth of risk estimates before us indicate that the risk is quite low.

It is often said that one should beware of too much trust in the point estimates of probabilistic risk assessments, that one should consider the uncertainties. This we do. But some then go on to demand exact quantitative definitions of the uncertainty. This demand is a form of bottom line fallacy.

Precise statements of uncertainty come only with large amounts of data. At the very low levels of risk with which we are dealing, the occurrence of actual events is, thankfully, very rare indeed. Thus, we cannot have exact quantitative estimates of uncertainty. But we can and must, continually, explore the sensitivity of our estimates and our decisions to the gaps in our knowledge. We have been doing that and we will keep at it.

In summary, present reactors pose no undue risk to public health and safety. This policy statement acknowledges that and indicates a willingness to permit continued operation of existing reactors as well as to license new reactors. This policy statement has been studied intensively for over three years. It has been reviewed carefully and endorsed by the Advisory Committee on Reactor Safeguards. It has not been lightly considered nor lightly decided. I am confident that the Commission has enunciated a sound regulatory policy.

Dissenting Views of Commissioner Asselstine

Summary

The foremost risk to the public from the operation of nuclear reactors derives from core meltdown accidents which can, through the release of substantial quantities of radioactive materials, result in the injury and death of a catastrophic number of people. This policy statement, which establishes Commission policies on these severe accident risks, represents one of the most fundamental regulatory decisions

[4] See "Integrated Safety Assessment Program (ISAP)," SECY 84–133, March 23, 1984.

[5] See 10 CFR 2.764(f) and "Statement of Policy: Further Commission Guidance for Power Reactor Operating Licenses," 45 FR 85236, December 24, 1980.

ever made by this agency. This statement, together with three other related regulatory decisions, will chart the future course of this agency and the nuclear industry on nuclear safety issues for many years to come. The three other decisions are the Commission's decision on the acceptability of the severe accident risk at the two operating Indian Point plants, the development of a backfitting rule incorporating a substantial safety threshold for the imposition of new requirements together with heavy reliance on quantitative cost/benefit analyses, and the development of a provisional, and ultimately a final, safety goal with numerical standards for evaluating the acceptability of nuclear accident risk. Taken together, these four Commission actions will set the framework for deciding whether the NRC and the industry will pursue existing and future significant safety issues, whether further improvements in safety will be pursued for both existing and future plants, and how such decisions will be made.

Unfortunately, the first two of these decisions by the Commission lead me to conclude that we are on the wrong course. My views opposing the Commission's Indian Point decision were set forth in considerable detail in the Commission's written decision (see CLI-85-06), and I will not rehearse those views here. Suffice it to say that the Commission's unsubstantiated and overly optimistic assumptions on the long-term acceptability of the severe accident risk posed to the public by those plants have now been extended by this policy statement to cover all existing and future nuclear powerplants in this country. In my judgment, the Commission's action today fails to provide even the most rudimentary explanation of, or justification for, these sweeping conclusions. As a basis for rational decisionmaking, the Commission's severe accident policy statement is a complete failure.

Existing Plants

I see at least four fundamental flaws in the Commission's policy statement as it applies to existing plants. First, while the policy statement reaches a positive conclusion on the acceptability of the severe accident risk posed by existing plants, it fails to articulate what that risk is; it fails to identify the relevant technical issues evaluated in assessing the acceptability of that risk; it fails to explain how those technical issues were considered and resolved by the Commission in reaching its positive conclusion; and it fails to demonstrate the technical support for that conclusion based on scientifically accepted principles and methodology.

Absent a detailed discussion of the severe accident risk posed by existing plants and of the reasoning and scientific basis supporting the Commission's conclusion on the acceptability of that risk, that conclusion must be viewed as nothing more than an unsubstantiated assertion deserving of little weight.

Second, the Commission's policy statement fails to provide any explanation of the Commission's treatment of uncertainties in evaluating the risk of severe accidents. The absence of virtually any explanation of how uncertainties have been treated in this policy statement further undermines the validity of the Commission's broad conclusions on the acceptability of the risk posed by severe accidents.

Third, the Commission fails to address in a clear and consistent manner the need to prevent further severe reactor accidents. Although the Commission's policy statement pays lip service to this goal, it fails to include the means to fulfill that objective.

Fourth, the Commission's policy statement places undue reliance on probabilistic risk assessments (PRA's) as a means for resolving severe accident questions for existing plants. This reliance fails to recognize present weaknesses in these assessments due to the limited number of PRA's available thus far, the variations among the existing PRA's, the absence of accepted guidelines on how to conduct PRA's and to evaluate them in making severe accident risk judgments, and the uncertainties inherent in attempting to extrapolate plant-specific PRA results to other plants.

Future Plants

The Commission's policy statement is equally flawed in its treatment of severe accident risk for future plants. First, the policy statement promises that the Commission will make final decisions in the near term on the acceptability of new plant designs for severe accident purposes. At the same time, the policy statement acknowledges that key elements in evaluating the acceptability of severe accident risk—criteria for the preparation and evaluation of PRA's, containment performance criteria, and criteria for evaluating the risk contributions due to sabotage and human performance—will not be available for some time. Thus, the Commission's approach is to agree to make final decisions on severe accident risk for future plants before the technical basis for evaluating the nature and acceptability of that risk is available.

Second, the policy statement does not go far enough in insisting upon reductions in the severe accident risk of future plant designs. Such reductions are much more readily achievable in new designs for as-yet unbuilt plants than for existing plants. While the Commission's policy statement urges reactor designers to make safety improvements in the designs of future plants, it does nothing to require that improvements be made.

Third, the Commission's policy statement retains the option of authorizing the start of construction of future plants based upon only limited plant design information, including the limited design information which would be needed to support issuance of a preliminary design approval (PDA). Past experience with nuclear powerplant design, construction and regulation has taught us the many pitfalls of the old design-as-you-build approach. By continuing to allow the start of plant construction with only limited design work complete, the Commission seems committed to repeating the mistakes of the past—mistakes which have led to the deferral of significant design issues until the construction and pre-operation stages and the need to modify work already in progress or completed.

Taken together, these flaws in the Commission's severe accident policy statement cast doubt upon the adequacy of the Commission's overall approach to dealing with severe accident risk and undermine the validity of the Commission's sweeping judgments of the acceptability of that risk for existing and future plants.

Discussion

Before elaborating on the major infirmities of this policy statement, it is useful to explain what we know about the severe accident risks to the public.

Risks

Risks are commonly defined as the product of the probability that an event will occur and the consequences of the event happening. In regulating the nuclear industry, the Commission makes extensive use of a methodology called probabilistic risk assessment (PRA). In conducting a PRA the analyst calculates the core meltdown probability and, given a particular core meltdown scenario, the analyst then estimates the consequences to the public. The Commission uses the bottom line of these PRA's in deciding whether to improve reactor safety or to relax the safety standards even though such PRA's do not consider all contributors to core meltdown risks or quantify all of the uncertainties.

A typical result of a PRA which is used by NRC in reaching safety decisions is the estimated core meltdown probability of about one in ten thousand (or 10^{-4}) per reactor year. However this probability estimate is often based on what is called the "median" value. It is important to understand just what the meaning of this bottom line number really is. Because of major inadequacies in the data base, because of the vast complexity of nuclear plants, because a tremendous number of assumptions must be made in calculating core meltdown probabilities, and because large scale core meltdown phenomena are poorly understood, no *one* calculation will yield a remotely meaningful probability of catastrophic consequences. Therefore, the PRA analyst must perform thousands of individual estimates of the core meltdown probability while randomly varying within chosen distribution patterns which themselves are not precisely known individual component failure probabilities, human error rates, and theoretical models that are thought to describe most of the important physical processes or engineering behavior. Any one of these individual estimates is as likely to be valid as the estimate resulting from any one of the other thousands of calculations. There is a crucial, but untenable, underlying assumption that all core meltdown sequences have been accounted for in the estimates. The analyst then scans all of the estimates and picks the probability value at which half the estimates are above the half are below. This number is called the median. It is, according to the Commission, the "best estimate". When calculated in this way, however, one cannot say with any confidence that this median value is the true core meltdown probability. Nonetheless, the Commission arbitrarily chooses this median number to use in making its regulatory decisions.[1]

The spread in the estimated core meltdown probabilities for a typical plant range from approximately one chance in one thousand (10^{-3}) per year to one chance in one hundred thousand (10^{-5}) per year, with a median value of one chance in ten thousand (10^{-4}) per year, give or take a few. However, there is no proof that the median of the calculated values reflects the actual risk any more than do the estimates of 10^{-3} per year or 10^{-5} per year.

Another typical result of PRA's is the prediction that about 1 out of 10 core meltdowns likely will result in lethal radiation doses to about 1,000 people. Such consequences of core meltdown accidents are attributable to degraded performance of the containment, which can come about in a variety of ways that are not precisely quantifiable. Because of these uncertainties in quantification, the fraction of core meltdown accidents which would lead to catastrophic consequences is actually a range of values. The range could be two or three times greater than the above estimate: or it could be two or three times less. Picking the minimum factor of 2 and assuming there are 100 operating reactors, the approximate range of chances of a catastrphic accident between now and the year 2000 would be anywhere between 0.2 (2 chances in ten) and 0.001 (one chance in a thousand).

Therefore, the information before the Commission indicates that there could be anywhere between a 20 percent chance and a 0.1 percent chance of an accident at a nuclear reactor in the next 15 years that would result in lethal doses to about 1,000 people.The range of chances could be larger than this if one considers all contributors to the core meltdown probability and all uncertainties. Likewise, the number of deaths could be larger or smaller. Admittedly, there are many ways of going about estimating the range of risks. However, if there is validated quantitative information on core meltdown risks that is better, it has not yet been demonstrated. Thus, because of the many uncertainties involved in calculating both the probabilities and the consequences of core meltdowns, one number does not give a true picture of the actual risk. A range of possibilities is a more accurate representation of our understanding of the issue.

A serious consideration of the core meltdown risks would consider this full range of calculated risks and would address forthrightly the question of whether this risk is acceptable or unacceptable, both for the immediate future and over the long term. The Commission's consideration of severe accident risks instead focuses on a median number, ignoring the actual range of of values and the uncertainties inherent in using a median number for decisionmaking.

Since the foremost risk to the public from the commercial nuclear industry derives from severe accidents, adopting a policy that seeks to resolve severe accident issues in a definitive manner is the most basic duty which can be undertaken by the Commission in meeting its responsibility to decide what constitutes acceptable risk to the public. The Commission claims in this policy statement to have examined an extensive range of technical issues relating to severe accident risks in reaching its judgment "that existing plants do not pose an undue level of risk to the public." The Commission's policy statement does not, however, incorporate an explanation, or for that matter even a description, of the most significant issues that have been resolved and the manner in which they were resolved. Nor does it include a description of the methods of analyses used in resolving the issues or decision criteria that were used for reaching the ultimate judgment. It is, therefore, impossible to discern the bases for the Commission's decision.

Uncertainties

A paramount concern regarding the acceptability of the risks to the public that must be resolved is how to reach a judgment on this issue in the face of enormous uncertainties which are up to 100 times the median value used by the Commission. Depending on how such uncertainties are factored into the decision, judgments could range from requiring substantial efforts to reduce core meltdown risks to doing nothing about them. Scientifically accepted data and methodology are not available at this time to reduce substantially those uncertainties so that, as the technical staff of the NRC has repeatedly told the Commission, it is "mandatory" to consider them in any application of risk assessments.

After being informed of the uncertainties in the risk estimates, the Commission simply ignores them. The Commission fails to provide any basis

[1] The practice of using median estimates was strongly criticized by our Advisory Committee on Reactor Safeguards during its July 11, 1986 meeting with the Commission. The ACRS recommended that mean rather than median estimates be used, and noted that use of median rather than mean estimates can result in a substantial underestimate of the effects of uncertainties in making reactor accident risk estimates. As indicated above, the median is that point on a spectrum at which half of the values fall above and half fall below. The mean is the average value of the spectrum of risks and is also called the "expected value."

Continued

Some PRA analysts base their estimates on the mean. However, the Commission has twice endorsed use of the median value. The first time was when the Commission endorsed WASH-1400 (Reactor Safety Study) in 1973 and the second time was when the Commission approved the provisional Safety Goal Policy Statement (NUREG-0880, Revision 1) in 1983.

for its decision to ignore these uncertainties. Absent some rational treatment of these uncertainties or a convincing justification for why they can be ignored, the public can have little confidence in the Commission's conclusion that the risks to the public from a severe accident at a nuclear powerplant are acceptable. The only available explanation of the NRC's approach to making decisions in the face of these significant uncertainties is given on pages 133 through 140 of NUREG-1070, "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulation", October 1984. About half of the pages are blank and the remainder are not much better. This discussion of uncertainties is inadequate and fails to provide a sufficient basis to justify the Commission's sweeping conclusions on the acceptability of the severe accident risk.

Another fundamental issue requiring resolution is the level of risk to the public that reasonably should be found acceptable. Beyond making a sweeping conclusion that the severe accident risk at the existing plants does not pose an undue risk to the public, the Commission fails to address this fundamental question. In fact, the Commission's technical staff is just now embarking on a prog[illegible]m of analysis that "will form part of the basis for a Commission judgment on the level of safety presently achieved by existing plants for severe accidents."[3] Since the Commission is just beginning this program, it cannot serve to justify the Commission's judgment on the acceptability of the severe accident risk.

In its Indian Point decision, the Commission adopted specific point estimates of core meltdown risks for the Indian Point reactors and found them to represent an acceptable level of risk. In the course of developing this policy statement the Commission expressed much interest in the bottom line results of *all* completed PRA's, whether the reported point estimates were the mean or median. The technical staff has repeatedly cautioned the Commission that such bottom line numbers are not credible. What then is the basis for the Commission's position that the level of severe accident risk posed by the existing plants is acceptable?

The Commission's decision-making process in developing this policy statement is simply to rely upon "point estimates" of the core meltdown risks without any consideration of the effects of the uncertainties. This approach can lead to a decision to doing nothing to reduce core meltdown risks. Factoring into the decision the uncertainties in estimating the level of core meltdown risks would lead to a decision to search for ways to reduce the risks. However, given the current political climate, there is little sympathy for backfitting existing plants. Thus, the Commission chooses to rely on a faulty number which supports the outcome they prefer and to ignore the uncertainties, those that are known and quantified and those that are not quantifiable.

What level of confidence does the Commission have in its judgment that core meltdown accidents present no undue risks to the public? The Commission nowhere expresses the degree of confidence it seeks to ensure that catastrophic accidents do not happen. Yet, the Commission's chief safety officer recently wrote: "In view of the large uncertainties surrounding methods of assessing severe accident risk, the *level of assurance* (or confidence) of no undue risk to the public is regarded as no less important than the estimated *level of risk* itself (emphasis in the original)." Letter from H.R. Denton, NRR, to A.E. Scherer, Combustion Engineering, Inc., dated December 28, 1984, subject "SECY-84-370, Severe Accident Policy".

Another problem with the Commission's policy statement is that it clearly contradicts what the Commission is doing in other areas. For example, in this policy statement the Commission states: "A fundamental objective of the Commission's severe accident policy is that the Commission intends to take all reasonable steps to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur." However, compare this statement with the Commission's proposed backfitting standard: "The Commission shall require the backfitting of a facility *only* when it determines, based on a systematic and documented analysis * * * that there is a substantial increase in the overall protection of the public health and safety * * * to be derived from the backfit *and* that the direct and indirect cost of implementation for that facility are justified in view of this increased protection." (emphasis added) The Commission has already defined a substantial increase in protection as meaning a backfit that would at least reduce the "point estimate" of the calculated core meltdown risks by half. Unless such a reduction can be "demonstrated", the Commission will not consider requiring the change. This is a much higher barrier to requiring improvement in reactor safety than the policy statement would have us believe is the Commission's policy.

Further, the Commission's provisional safety goal is not intended to regulate on the basis of preventing core damage accidents, as implied in the above purported fundamental objective. Rather, the safety goal assumes that the containment is an independent bulwark capable of limiting the external release of radioactivity to modest amounts for most core meltdown accidents. Thus, according to the Commission, there is no need to regulate on the basis of preventing core meltdowns. I am not as sanguine as the Commission on the acceptability of core meltdown accidents. Even if the containment happens to retain most of the radioactive fission products in the next severe accident, another accident equal to or more severe than that which occurred at Three Mile Island would be unacceptable to the public and the Congress and would be disastrous for the nuclear industry and the NRC.

But more importantly, the Commission's belief that the containment will retain all but modest amounts of radioactivity during most core meltdowns is not yet supportable based on scientifically accepted principles and methodology. There simply is no actuarial experience or direct experimental data on large scale core meltdown phenomena or containment performance characteristics given a core meltdown. In the past, estimates of the quantities of radioactive releases to the environment have been based on not much more than interpolations of extrapolations of approximations. It is for this reason the Commission has an ongoing program, which has cost a quarter of a billion dollars in the last few years, in an attempt to bring some science to estimating the core meltdown risks. However, even in this program the data being generated are from limited small scale tests.

Thus, a reading of this policy statement indicates that the Commission's claim that in developing this policy statement it has examined an extensive range of issues is incorrect. It shows rather that the Commission either examined the wrong issues or gave short shrift to the fundamental issues.

In failing to define accurately the level of severe accident risk at the existing plants and to address the need for

[3] See, NUREG-1070, "NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plant Regulation," October 1984, p. 27.

additional changes to the plants to make this risk acceptable for the long term, the Commission is repeating past failures to deal effectively with the severe accident question. The concept of the reactor containment originally evolved as a vessel to contain a full core meltdown. But in the mid-1960's, the reactor designers began placing high powered cores into roughly the same kind of containment. The decay heat of those higher powered cores was so high that the containment vessel could no longer be considered as an effective independent barrier to the release of the fission products evolved during a core meltdown. At that time, the Atomic Energy Commission's Advisory Committee on Reactor Safeguards (ACRS) began urging the development and implementation, in about two years, of safety features to protect against a loss of coolant accident in which the emergency core cooling system did not work. The AEC and the industry believed that sufficient data were available to justify with a high degree of confidence the adequacy of the then-existing safety standards. Therefore, the AEC ignored the advice of the ACRS.

Over the years, the AEC and the NRC after it have reiterated these sweeping and optimistic statements on severe accident risk. At the same time, the numerous technical flaws in the Commission's judgments have become readily apparent as more information and data regarding the level of safety of the reactors has become available.[4]

When all of the available data are considered, I believe it fair to say that the estimated uncertainties in the risk calculations today are as large as they were at least ten years ago. Yet, the Commission is once again sweeping aside these uncertainties in order to make the same unsubstantiated and overly optimistic generalizations about the acceptability of the current level of severe accident risk which have been proven wrong in the past.

Needed Improvements

A disciplined approach to deciding whether to require core meltdown risk reduction measures should not only specify the Commission's expectations on addressing uncertainties but it should also describe the Commission's policy on acceptable ways to perform cost-benefit analyses.

Further, guidance from the Commission is needed on whether to emphasize core meltdown prevention measures or core meltdown mitigation measures. Of course, in order to develop a policy on the latter (whether for existing plants or future plants), one must first identify the root causes of core meltdown risks. One must also develop a policy on containment performance expectations.

Unfortunately, the Commission refuses forthrightly to address these issues. An effective guide to regulatory decision-making on the treatment of severe accident issues requires an understanding of what is expected by way of containment performance, of the root causes of core meltdown risks, and of the methods for performing sound cost-benefit analyses. Yet all of these elements are missing from the Commission's policy statement. The Commission's actual decision-making guidance in this policy statement is limited to the statement that a new requirement might be imposed if it involves "low-cost changes in procedures or minor design modifications."

The Commission claims that PRA's identify the plant specific vulnerabilities that dominate the core meltdown risks. It is true that PRA's can identify *some* of the vulnerabilities to catastrophic accidents. But the Commission's rationale for relying upon PRA's in assessing core meltdown risks begs the questions: what of the uncertainties in PRA's? What of oversights in the analyses? What of the multitude of assumptions and approximations in the PRA's? What of the residual risks once the specific vulnerability has been fixed? These questions are germane to resolving severe accident issues. Yet they are not addressed in the Commission's policy statement.

Operational experience gives additional insight into the level of safety. Actuarial experience with reactor accidents indicates that the average core meltdown frequency is not above the upper limit of the PRA results. Core meltdown accidents involve multiple failures and a progression of events that make close calls somewhat identifiable. If the industry average of the core meltdown frequency were as high as 10^{-3} per reactor year, one would expect more close calls on core meltdowns than appear to have occurred within the more than 800 reactor years of U.S. nuclear power experience. But such actuarial inferences must be made cautiously in part because the operating reactors continue to surprise us. What actuarial experience we have is severely limited by our lack of detailed understanding of the performance of the plants, their designs, their weak spots, and because of the wide variations in the designs and in utility capabilities. Further, the usefulness of actuarial experience in drawing broad conclusions about commercial nuclear reactors is highly controversial and fraught with uncertainties.

The Commission argues that credit can be taken for the improvements implemented to address specific close calls such as the TMI accident, the Browns Ferry fire and the Rancho Seco transient. Each of these were previously unrecognized (or at best inadequately appreciated) accident sequences. This is also true of, for example, the Susquehanna station blackout event from a single failure, the Indian Point vulnerability to a single failure of a battery, and the so-called interfacing system LOCA's for boiling water reactors. None of these latter events were identified or highlighted through PRA's nor were they expected to be, given the level of detail that typically goes into a PRA and given the subjective nature of PRA's. Whether these latter events should be called close calls is arguable but their occurrences certainly suggest a need to consider the root causes of significant operating events and the collective meaning of those events before passing judgment on the acceptability of the level of safety achieved at existing power reactors. Common sense also suggests completing such an analysis before developing guidelines for the design of future reactors. Yet all of these concerns are swept aside in the Commission's policy statement.

The TMI Action Plan called for a large number of modifications to the operating plants. In addition to those modifications, the Action Plan committed to a rulemaking to consider to what extent, if at all, existing nuclear power plants should be required to deal effectively with damaged core and core meltdown accidents. There was to be a demarcation between those plants already operating or under construction and the next generation of future plants. Because the Commission perceived in 1980 that there would be a long hiatus in new plant orders, ample time existed to reconsider the General Design Criteria, the design bases, and the other regulations in light of all that had been learned through the years of experience with large power reactors, including the TMI accident. From this in-depth assessment of the strengths and weaknesses of the large power reactor

[4] Dr. David Okrent (who has been a member of the ACRS since 1963) has compiled a detailed account of the judgments made by the AEC and the NRC on severe accident risk and the technical flaws in those judgments. See David Okrent, *Nuclear Reactor Safety-On The History of the Regulatory Process*, University of Wisconsin Press, 1981, pp. 163-178.

designs and the approach taken by utilities toward constructing the plants. NRC would then be in a position to articulate safety principles that it expected to be incorporated into designs for future applications. Thus, the Commission in 1980 signaled there would be a significant step forward in advancing the protection of the public. The Commission in this policy statement takes several steps backwards.

One backward step discussed above is the Commission's decision to accept the core meltdown risks as they exist in the current generation of plants without even addressing some of the most fundamental issues. Another backward step is abandonment of the expressed desire for a fresh look at light water reactor safety for future designs and the insistence on improvements in the level of severe accident risks for any future plants. A third backward step in this policy statement is the return to the philosophy of the 1960's and 1970's that construction permits can be issued based on only partial design information.

For any future reactor orders, nuclear utilities themselves have expressed a desire for plant designs that are simpler, safer, and more forgiving. Both the Electric Power Research Institute (EPRI) and Edison Electric Institute (EEI) have impressed on the Commission the need for a fresh look at light water reactor technology. These utility sponsored organizations have also indicated that plant construction for new plants should not begin until there exists an essentially complete design for the plant. Yet none of these forward thinking requirements are to be found in the Commission's policy statement. Instead, the Commission states that it will be satisfied with mere refinements in the old designs and that it is willing to continue to approve partial designs for issuance of Construction Permits.

I cannot leave this latter point without a sad commentary on the Commission's priorities. One issue in this policy that commanded great interest within the Commission was how to circumvent its regulation that requires a comparison of a design to the staff's Standard Review Plan. This effort was motivated by the objections of one reactor vendor. Indeed, the Commission's efforts to use this policy statement as a vehicle to permit the reactor vendor to circumvent the Commission's regulations took precedence over any Commission consideration of such fundamental issues as the actual level of severe accident risk to the public, the acceptability of that risk and potential measures to reduce that risk.

A Rational Approach to Severe Accident Decisionmaking

What the Commission should have done in its policy statement is to set forth precisely and in understandable terms what our present estimation of the risk of severe accidents is, whether the Commission believes that risk to be acceptable or not, what specific technical support can be offered in support of that judgment, and how the relevant uncertainties have been treated. The Commission should also have come to grips with a central question in our regulatory program: that is, given our present state of knowledge concerning severe accident risks, should we continue to pursue possible improvements in severe accident prevention and mitigation? If the Commission does not believe that the present level of severe accident risk is acceptable for the remaining 40-year life of some existing plants, then the Commission should outline its program for bringing this long-term risk within acceptable bounds. Only through such a process can the technical community, other public policy makers and the public understand and accept the Commission's judgment on the severe accident risk question. Unfortunately, such an analysis is nowhere to be found in the Commission's policy statement.

Based upon the preceding discussion, I would have reached the following conclusion. First, the risk to the public posed by severe accidents at the existing plants is not acceptable for the full remaining operating lives of those plants. Therefore, the Commission should continue to pursue cost-effective risk reduction measures for these plants. I would apply the as-low-as-reasonably-achievable (ALARA) principle to reducing severe accident risk, subject only to the qualification that changes which would only result in trivial safety improvements need not be pursued. I would have simply acknowledged the obvious: that the public and the Congress will not tolerate, and the industry and the NRC cannot allow, another severe accident as serious as the Three Mile Island accident or worse. My views in this regard are identical to those expressed by the Kemeny Commission nearly six years ago:

> Whether in this particular case we came close to a catastrophic accident or not, this accident was too serious. Accidents as serious as TMI should not be allowed to occur in the future.
>
> The accident got sufficiently out of hand so that those attempting to control it were operating somewhat in the dark. While today the causes are well understood, 8 months after the accident it is still difficult to know the precise state of the core and what the conditions are inside the reactor building. Once an accident reaches this stage, one that goes beyond well-understood principles, and puts those controlling the accident into an experimental mode (this happened during the first day), the uncertainty of whether an accident could result in major releases of radioactivity is too high. Adding to this the enormous damage to the plant, the expensive and potentially dangerous cleanup process that remains, and the great cost of the accident, we must conclude that—whatever worse could have happened— the accident had already gone too far to make it tolerable.
>
> While throughout this entire document we emphasize that fundamental changes are necessary to prevent accidents as serious as TMI, we must not assume that an accident of this or greater seriousness cannot happen again, even if the changes we recommend are made. Therefore, in addition to doing everything to prevent such accidents, we must be fully prepared to minimize the potential impact of such an accident on public health and safety, should one occur in the future.

Report of the President's Commission on The Accident at Three Mile Island, p. 15.

In order to reduce the severe accident risk over time to acceptable levels, I would have undertaken four specific initiatives. First, I would have required a detailed search for plant-specific equipment and design vulnerabilities at each existing plant to identify and correct those weaknesses which constitutes significant contributors to the risk of a severe accident.

Second, I would have initiated a concerted effort to improve operational performance at the existing plants, with special emphasis on areas of weakness throughout the industry (maintenance and surveillance testing stand out as good examples) and on specific utilities with a history of marginal performance. The June 9, 1985 operating event at the Davis Besse nuclear powerplant once again demonstrated the dangers inherent in the combination of a marginal plant design and a utility with marginal operating performance.

Third, I would have initiated a comprehensive assessement of the level of safety and the existing plants have achieved. The object of this effort would be to identify the root causes of severe accident risks. This effort would also identify possible measures which offer the promise of significantly reducing severe accident risk by overcoming the adverse effects of equipment breakdowns, human error, design deficiencies and areas of present uncertainty which are likely to persist despite our best efforts to address my first two initiatives. Indeed, as the Commission's chief safety officer noted in a June 27, 1985 memorandum to the Executive Director for Operations:

I believe that the recent Davis-Besse event illustrates that, in the real world, system and component reliabilities can degrade below those we and the industry routinely assume in estimating core melt frequencies. Our regulatory process should require margins against such degradation and also to reflect the uncertainties in our PRA estimates.

Finally, for future plants, I would have explicitly required measures to improve the margin of safety against severe accidents in future plants and to address the mistakes of the past. Such measures could include requirements for greater simplicity in plant design, improved maintainability, and a requirement for essentially complete plant designs prior to the issuance of NRC approval for the start of plant construction.

I believe that these measures would be sufficient to bring the risk of severe accidents within acceptable bounds for the remaining operating lives of the existing plants and for the operating lives of any future plants. Moreover, such an approach would do much to restore public confidence in nuclear power and in the effectiveness of the NRC's regulatory process. It is unfortunate that the Commission has chosen another path. However, key decisions remain to be made by the Commission in adopting a final backfitting rule and a final safety goal. Those decisions represent a final opportunity to come to grips with many of the pivotal issues avoided in this policy statement. In that regard, it is encouraging that there appears to be an emerging consensus within the NRC senior technical staff and within the ACRS in favor of safety improvements to reduce severe accident risk both for existing and for future plants.

50 FR 43621
Published 10/28/85
Effective 10/28/85

Commission Policy Statement on Engineering Expertise on Shift

AGENCY: Nuclear Regulatory Commission.

ACTION: Policy Statement on Engineering Expertise on Shift.

SUMMARY: This Policy Statement presents the policy of the Nuclear Regulatory Commission (NRC) with respect to ensuring that adequate engineering and accident assessment expertise is possessed by the operating staff at a nuclear power plant. This Policy Statement offers licensees two options for providing engineering expertise on shift and meeting licensed operator staffing requirements.

Option 1 provides for elimination of the separate Shift Technical Advisor (STA) position by allowing licensees to combine one of the required Senior Reactor Operator (SRO) positions with the STA position into a dual-role (SRO/STA) position. Option 2 provides that a licensee may continue to use an NRC-approved STA program, with certain modifications, while meeting licensed operator staffing requirements.

EFFECTIVE DATE: October 28, 1985.

FOR FURTHER INFORMATION CONTACT: Clare Goodman, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: 301/492-4894.

SUPPLEMENTARY INFORMATION:

Background

Following the accident at Three Mile Island in March 1979, a number of studies were conducted to determine why the accident occurred, what factors might have contributed to its severity, and what the industry and the NRC could do to prevent the recurrence of the same or a similar accident. These studies concluded, among other things, that a number of actions should be taken to improve the ability of shift operating personnel to recognize, diagnose, and effectively deal with plant transients or other abnormal conditions.

To address these recommended improvements, the NRC initiated both short-term and long-term efforts. The short-term effort required that as of January 1, 1980, each nuclear power plant have on duty a Shift Technical Advisor (STA) whose function was to provide engineering and accident assessment advice to the Shift Supervisor in the event of abnormal or accident conditions. The STA was required to have a bachelor's degree in engineering or the equivalent and specific training in plant response to transients and accidents. The STA requirement was identified to licensees via NUREG-0578 (July 1979) [1] and NUREG-0737 (November 1980) and was later mandated by plant-specific Confirmatory Orders.

Concurrently, the NRC and industry embarked on a longer-term effort aimed at upgrading staffing levels and the training and qualifications of the operating staffs, improving the man-machine interface, and increasing capabilities for responding to emergencies. At the time the STA requirement was imposed, it was intended that use of the dedicated STA would be an interim measure only until these longer-term goals were achieved.

These long-term initiatives collectively result in an improvement in the capabilities and qualifications of the shift crew and their ability to diagnose and respond to accidents. These initiatives include shift staffing increases, training and qualification program improvements, hardware modifications, emphasis on human factors considerations, procedural upgrades, and development of extensive emergency response organizations to augment on-shift capabilities during abnormal conditions.

Draft Policy Statement

On July 25, 1983, the Commission published in the **Federal Register** (48 FR 33781) a Draft Policy Statement on Engineering Expertise on Shift to reassert the Commission's belief that engineering and accident assessment expertise must be available to the operating crew at all nuclear power plants.

The Draft Policy Statement on Engineering Expertise on Shift offered licensees of nuclear power plants and applicants for operating licenses two options for meeting the staffing requirements of 10 CFR 50.54(m)(2) and the requirement in NUREG-0737, Item 1.A.1.1 for a Shift Technical Advisor (STA). Option 2 gave them the opportunity to combine the licensed Senior Operators' (SRO) and Shift Technical Advisors' (STA) functions. Under Option 1, licensees that did not want to combine the SRO and STA functions could continue with their approved STA program in accordance with the description in NUREG-0737, "Clarification of TMI Action Plan Requirements."

Interested persons, applicants, and licensees were invited to submit written comments to the Secretary of the Commission. Following consideration of the comments, the Commission amended the Draft Policy Statement, as discussed in the following sections.

Comments on the Draft Policy Statement

A total of 34 responses were received and evaluated. The public comments related primarily to the combined SRO/STA position. The following discussion highlights the major points raised in the comments and the resolution of those comments. A detailed analysis of all public comments and their resolution was also prepared. (Copies of those letters and the detailed analysis of all the public comments are available for public inspection and copying for a fee at the NRC Public Document Room at 1717 H Street NW., Washington, DC.)

Of the 34 letters received, 18 included

[1] NUREG-series reports and other documents referenced in this notice are available for inspection or copying for a fee in the NRC Public Document Room, 1717 H Street NW, Washington, DC. The reports may be purchased from the U.S. Government Printing Office (GPO) by calling 202/275-2060 or by writing the GPO, P.O. Box 37082, Washington, DC 20013-7082. They may also be purchased from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161.