UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of ARIZONA PUBLIC SERVICE COMPANY, et al. (Palo Verde Nuclear Generating Station, Units 1, 2 and 3)	))))))))	Docket Nos. STN 50-528 STN 50-529 STN 50-530

AFFIDAVIT OF F. W. HARTLEY
ON CONTENTION NO. 6B

STATE OF ARIZONA)
) ss.
County of Maricopa)

I, F. W. Hartley, being duly sworn, upon my oath state as follows.

1. I am employed by Arizona Public Service Company as Manager of Nuclear Operations.

2. In such capacity I am responsible for the day-to-day operation and maintenance of the Palo Verde Nuclear Generating Station ("PVNGS"). My resume is set forth in Attachment FWH-1.

3. This affidavit is made with reference to Intervenor Patricia Lee Hourihan's Contention No. 6B concerning the subject of ATWS.

4. ATWS is an acronym for "anticipated transients without scram."

5. Anticipated transients are deviations from normal operating conditions which can be foreseen as probable occurrences during the service life of a nuclear power plant.

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6. An ATWS event refers to the failure of the reactor protection system to shut down the reactor following the occurrence of an anticipated transient requiring reactor shutdown.

7. ATWS is an unresolved generic safety issue which has been included by the NRC Staff in its "Task Action Plans for Generic Activities," NUREG-0371 (November 1978), as Task No. A-9.

8. The NRC staff has issued its Safety Evaluation Report related to the operation of the Palo Verde Nuclear Generating Station Units 1, 2, and 3, NUREG-0857 (November 1981) and its Safety Evaluation Report related to the final design of the Standard Nuclear Steam Supply Reference System, CESSAR System 80, NUREG-0852 (November 1981).

9. The Staff's review of ATWS for PVNGS is set forth at pages 15-1 to 15-2 of the Safety Evaluation Report for PVNGS.

10. In its Safety Evaluation Report for PVNGS at page 15-2, the NRC Staff has identified two procedural requirements which in the Staff's view serve as an acceptable basis for operation of PVNGS pending completion of any plant modifications ultimately required by the Commission in its final resolution of ATWS as a generic safety issue.

11. As set forth at page 15A-29 of the PVNGS Final Safety Analysis Report, Joint Applicants have committed to meet the NRC Staff's ATWS procedural requirements set forth at page 15-2 of the Safety Evaluation Report for PVNGS.

12. As set forth at page 15A-29 of the PVNGS Final Safety Analysis Report, Joint Applicants have committed to have the required procedures implementing the Staff's requirements available for NRC review at least 60 days prior to fuel loading.

F. W. Hartley

Subscribed and sworn to before me this 13th day of January, 1982.

Lois A. Armstrong
Notary Public

My commission expires:
April 15, 1984

NAME: F. W. Hartley

ADDRESS: 7820 N. 107 Dr., Glendale, Az. 85307

EDUCATION & MILITARY SERVICE:

B.S. Degree in Management - Arizona State University
Retired USN Master Chief Steam Propulsion

PRIOR EMPLOYERS:

United States Navy
Connecticut Yankee Atomic Power Company
Northeast Utilities
Arizona Public Service

SUMMARY:

Thirty plus years experience in operation, maintenance and management of fossil and nuclear power plants. The past twenty-two years have been in the nuclear field - six in the Navy and sixteen in the commercial nuclear power field. Certified as a Navy Reactor operator in 1960 and an NRC Senior Reactor Operator License holder from 1967 to 1976.

PROFESSIONAL HISTORY:

5/81 - Present:

Manager of Nuclear Operations, Arizona Public Service Co.

10/76 - 5/81:

Manager of Palo Verde Nuclear Generating Station

1/76 - 10/76:

Superintendent of the Millstone Nuclear Power Station, Northeast Utilities, Organization size 250 personnel

12/1/69 - 1/76:

Superintendent of Connecticut Yankee Atomic Power Station, Haddam, Conn., Organization size 95 personnel

9/8/68 - 12/1/69:

Assistant Superintendent, Connecticut Yankee

10/1/67 - 9/8/68:

Operations Supervisor, Connecticut Yankee

3/21/66 - 10/1/67:

Shift Supervisor, Connecticut Yankee

12/20/62 - 3/15/66:

Nuclear Chief Operator and Engineering Watch Officer on the U.S.S. Long Beach (CGN-9), USN

5/1/60 - 12/1/62:

Chief Operator, Engineering Watch Officer and Shift Training Coordinator at AIW, Idaho FAlls, Idaho (National Reactor Testing Station). USN

6/1/59 - 5/1/60:

Nuclear Power academic and prototype schools, Vallejo, California and Idaho Falls, Idaho. USN

PROFESSIONAL AFFILIATIONS:

American Nuclear Society - Chairman of ANS 55.4, Member Executive Committee ROD Division

EEI - Member Nuclear Power Committee since 1968. Past Chairman, Nuclear Operating Experience TAsk Group under the Nuclear Power Subcommittee.

Founder and past Chairman, Western States Plant Managers Association

PERSONAL:

Height 5' 10", Weight 175 lbs.

Heath - Excellent

Marital Status - Married

Children - Four

NUREG-0857

Safety Evaluation Report

related to the operation of
Palo Verde Nuclear Generating Station,
Units 1, 2, and 3

Docket Nos. STN 50-528, STN 50-529, and STN 50-530

Arizona Public Service Company, et al.

U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

November 1981

15 ACCIDENT ANALYSES

15.1 INTRODUCTION

The analyses of normal operation, anticipated transients and generic accidents are provided in the CESSAR System 80 FSAR. Staff evaluations for those transients and accidents within CESSAR Scope are provided in the CESSAR SER.

15.2 NORMAL OPERATION AND ANTICIPATED TRANSIENTS

The staff evaluation is presented in the CESSAR SER.

15.3 LIMITING ACCIDENTS

Staff evaluations for the following accidents 15.3.1 through 15.3.8 are presented in the CESSAR SER.

- 15.3.1 Steam Line Breaks
- 15.3.2 Feedwater System Pipe Breaks
- 15.3.3 Reactor Coolant Pump Shaft Seizure
- 15.3.4 Reactor Coolant Pump Shaft Break
- 15.3.5 Inadvertent Opening of a Pressurizer Safety Value
- 15.3.6 Double-Ended Break of a Letdown Line Outside Containment
- 15.3.7 Steam Generator Tube Rupture
- 15.3.8 Loss-of-Coolant Accident

15.3.9 Anticipated Transients Without Scram

A number of plant transients can be affected by a failure of the scram system to function. For a pressurized water reactor, the most important transients affected include loss of normal feedwater, loss of electrical load, inadvertent control rod withdrawal, and loss of normal electrical power. In September 1973, the staff issued WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," establishing acceptance criteria for anticipated transients without scram. In conformance with the requirements of Appendix A to WASH-1270, and as discussed in the CESSAR SER, Section 15.3.9, Combustion Engineering submitted an evaluation of anticipated transients without scram in Topical Report CENPD-158, "Topical Report Anticipated Transients Without Scram." On December 9, 1975, the staff issued a report, "Status Report on Anticipated Transients Without Scram for Combustion Enegineering Reactors." In response, Combustion Engineering issued Revision 1 to CENPD-158 in May 1976. A reevaluation of the potential risks from anticipated transients without scram (ATWS) has been published in NUREG-0460, Volume 1 through 4. The status of this NUREG is described below:

(1) In March 1980 the 4th Volume of NUREG-0460 was issued by the NRC staff. The recommendations included design criteria for plants such as PVNGS and recommended rulemaking to establish such criteria.

(2) The NRC staff presented its recommendations on ATWS to the Commission, including the recommendation for rulemaking, in September 1980.

(3) After deliberation, the Commission will act on the matter. Whether it will agree to rulemaking is speculative at this time. If rulemaking is initiated by the Commission, the staff would expect that any rule adopted would include an implementation plan for all classes of plants.

As discussed in the CESSAR SER, all reference plants, including PVNGS 1-3, would be required to provide plant modifications in conformance with ATWS criteria and schedular requirements provided in the rule or as adopted by the Commission. The following discussion presents the bases for operation of PVNGS 1-3, prior to the adoption of a rule.

In NUREG-0460, Volume 3, the staff states: "The staff has maintained since 1973 (for example, see pages 69 and 70 of WASH-1270) and reaffirms today that the present likelihood of severe consequences arising from an ATWS event is acceptably small and presently there is no undue risk to the public from ATWS. This conclusion is based on engineering judgment in view of: (a) the estimated arrival rate of anticipated transients with potentially severe consequences in the event of scram failure; (b) the favorable operating experience with current scram systems; (c) the limited number of operating reactors." In view of these considerations and the staff expectation that the necessary plant modifications will be implemented in one to four years following a commission descision on anticipated transients without scram, the staff has generally concluded that pressurized water plants can continue to operate because the risk from anticipated transient without scram events in this time period is acceptably small. As a prudent course, in order to further reduce the risk from anticipated transient without scram events during the interim period before completing the plant modification determined by the Commission to be necessary, the staff, as discussed in the CESSAR SER, required that the following steps be taken:

(1) Develop emergency procedures to train operators to recognize anticipated transient without scram event, including consideration of scram indicators, rod position indicators, flux monitors, pressurizer level and pressure indicator, and any other alarms annunciated in the control room with emphasis on alarms not processed through the electrical portion of the reactor scram system.

(2) Train operators to take actions in the event of an anticipated transients without scram, including consideration of manually scramming the reactor by using the manual scram button, prompt actuation of the auxiliary feedwater system to assure delivery to the full capacity of this system, and initiation of turbine trip. The operator should also be trained to initiate boration by actuation of the high pressure safety injection system to bring the facility to a safe shutdown condition.

The staff considers these procedural requirements an acceptable basis for interim operation of the facility based on our understanding of the plant response to postulated anticipated trannsients without scram events.

The applicant has committed to develop emergency procedures for and train operators to respond to anticipated transients without scram per requirements (1) and (2) above. The staff finds this acceptable.

Proposed Rules

Federal Register
Vol. 46, No. 226
Tuesday, November 24, 1981

This section of the FEDERAL REGISTER contains notices to the public of the proposed issuance of rules and regulations. The purpose of these notices is to give interested persons an opportunity to participate in the rule making prior to the adoption of the final rules.

DEPARTMENT OF AGRICULTURE

Rural Electrification Administration

7 CFR Part 1701

Proposed Rescission of REA Bulletin 81-7:381-11

AGENCY: Rural Electrification Administration, USDA.

ACTION: Proposed rule.

SUMMARY: The Rural Electrification Administration (REA) proposes to amend Appendix A—REA Bulletins to provide for the rescission of REA Bulletin 81-7:381-11, "Changes or Corrections in Line Construction," which has become obsolete. The primary purpose of REA Bulletin 81-7:381-11 is to provide REA Form 216, "Construction Change Order." Since REA Form 216 was rescinded in an effort to eliminate unnecessary REA forms, REA Bulletin 81-7:381-11 is considered to be unnecessary.

DATE: Public comments must be received by REA no later than January 25, 1982.

ADDRESS: Submit written comments to the Director, Engineering Standards Division, Rural Electrification Administration, Room 1270, South Building, U.S. Department of Agriculture, Washington, D.C. 20250.

FOR FURTHER INFORMATION CONTACT: Mr. Edwin N. Limberger, telephone (202) 447-7040. A Draft Impact Analysis has been prepared and is available from the Director, Engineering Standards Division, at the above address.

SUPPLEMENTARY INFORMATION: Pursuant to the Rural Electrification Act, as amended (7 U.S.C. 901 et seq.), REA proposes to amend Appendix A—REA Bulletins to provide for the rescission of REA Bulletin 81-7:381-11, "Changes or Corrections in Line Construction." Since no significant effect on the economy will occur, since no significant increase in cost for consumers, subscribers, industries or Government will result, and since no significant impact on economic conditions will be caused, this action has been determined to be "not major."

The Regulatory Flexibility Act (Pub. L. 96-354) is not applicable to this action; therefore, a Regulatory Flexibility Analysis will not be prepared.

This proposed action is intended to eliminate an unnecessary bulletin, thereby saving the Government the cost of periodic revisions.

This program is listed in the Catalog of Federal Domestic Assistance as 10.850—Rural Electrification Loans and Loan Guarantees, 10.851—Rural Telephone Loans and Loan Guarantees and 10.852—Rural Telephone Bank Loans.

All written submissions made pursuant to this action will be made available for public inspection during regular business hours at the above address.

Dated: November 18, 1981.

Harold V. Hunter,

Administrator.

[FR Doc. 81-33672 Filed 11-23-81; 8:45 am]

BILLING CODE 3410-15-M

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

Standards for the Reduction of Risk From Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Commission is considering three alternatives for amending its regulations to require improvements in the design and operation of light-water-cooled nuclear power plants to reduce the likelihood of failure of the reactor protection system to shut down the reactor (scram) following anticipated transients and to mitigate the consequences of anticipated transients without scram (ATWS) events. This will reduce the overall risk of nuclear power plant operation. The consequences of this regulation will be to require electric utilities to install certain equipment in nuclear power plants and, possibly, to implement a reliability assurance program.

DATES: Comment period expires April 23, 1982. Comments received after April 23, 1982, will be considered if practical to do so, but only those comments received on or before this date can be assured of consideration.

ADDRESSES: Comments should be submitted in writing to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch. All comments received and all referenced and other NRC documents relevant to the ATWS issue will be available for public inspection in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C. Copies of referenced NRC reports may be purchased from the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

FOR FURTHER INFORMATION CONTACT: David W. Pyatt, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, (301) 443-5960.

SUPPLEMENTARY INFORMATION: Concern regarding protection against anticipated transients without scram (ATWS) events has long been a subject of extensive and continuing study by the NRC staff. The significance of ATWS for reactor safety is that some ATWS events could result in melting of the reactor fuel and the release of a large amount of radioactive fission products.

The principal benchmark for deciding whether and to what extent nuclear power plants should be modified because of ATWS-related safety concerns is set forth in subsection 161i(3) of the Atomic Energy Act. That section grants to the Commission the authority to "prescribe such regulations or orders as it may deem necessary * * * in order to protect health and to minimize danger to life or property." Throughout the history of regulating nuclear reactors, the dual concept of preventing accidents and mitigating their consequences should they occur, i.e., defense in depth, has been used to achieve this ojective. Thus, conservative design, construction, testing, maintenance and operation of plants are required so that accidents will not happen (i.e., have a low probability of occurrence). Then, to provide defense in depth, the capability to mitigate their consequences is required for accidents that are postulated to occur even though the design is required to include measures to prevent them.

ATWS accidents are a cause for concern because a mismatch can develop between the power generated in the reactor and the power dissipated in controlled ways if the scram system fails to shut down the reactor following a fault in the normal heat dissipation functions (transient events). The power mismatch can threaten the integrity of the barriers that confine the fission products. A core meltdown accident, in some cases accompanied by a failure of containment and a very large release of radioactivity, is a possible outcome of some ATWS accident sequences. Thus, the consequences of some postulated ATWS accidents are unacceptable.

There have been roughly one thousand reactor years of experience accumulated in foreign and domestic commercial light-water-cooled reactors without an ATWS accident. This experience suggests that the frequency of ATWS accidents is less than or of the order of once in a thousand reactor years. There have been several precursor events, i.e., faults detected that could have given rise to ATWS events. This suggests that the frequency of ATWS accidents, though less than once in a thousand reactor years, may not be very much less. Such frequencies are too high for accidents of the severity described above. Thus the NRC has determined that reductions must be made in the frequency, severity, or both the frequency and severity of ATWS accidents.

The Nuclear Regulatory Commission has under consideration three proposed alternative rules, each intended to reduce the risk posed by ATWS accidents. Two of these originated within the NRC, and are described below. The third is set out in a petition for rulemaking filed by twenty utilities ("Electric Utilities Petition," PRM 50–29, 45 FR 73080, November 4, 1980 and the supplement to the petition published on February 3, 1981, 46 FR 10501). The utilities' petition will not be reproduced here; however, the current period for the utility petition is hereby reopened to run concurrently with that of the two NRC proposed rules for the purpose of comparing and contrasting the utility petition with the two proposed rules published herein. Both of the NRC-proposed rules mandate improvements in ATWS prevention and mitigation. They differ in scope, approach, and criteria.

The first NRC-proposed rule is known as the staff rule and is a direct outgrowth of NUREG–0460, "Anticipated Transients Without Scram for Light Water Reactors," Volumes 1-4. It was submitted to the Commission for consideration in an early version in SECY 80–409, September 4, 1980, and in final form in SECY 80–409C, November 7, 1980. The second NRC-proposed rule is a recent proposal by former NRC Chairman Joseph M. Hendrie.[1] Dr. Hendrie's aim in starting afresh was to try an approach that would make licensees look carefully at their plants for ATWS-related vulnerabilities and then fix these vulnerable areas, employing systems analysis or reliability techniques.

The Commission believes that the likelihood of severe consequences arising from an ATWS event during the two to four year period required to implement a rule is acceptably small. This judgment is based on (a) the favorable experience with the operating reactors, (b) the limited number of operating nuclear power reactors, (c) the inherent capability of some of the operating PWRs to partially or fully mitigate the consequences of ATWs events, (d) the partial capability of the recirculation pump trip feature to mitigate ATWS events that has been implemented on all BWRs of high power level, and (e) the interim steps taken to develop procedures and train operators to further reduce the risk from some ATWS events. On the basis of these considerations, the Commission believes that there is reasonable assurance of safety for continued operation until implementation of a rule is complete. The implementation schedule contained in this rule balances the need for careful analysis and plant modifications with the desire to carry out the objectives of the rule as soon as possible.

Paperwork Reduction Act

A request for clearance of any application and reporting requirements of the alternative finally selected will be submitted to the Office of Management and Budget under the Paperwork Reduction Act (Pub. L. 96–511). At the time, the SF–83 "Request for Clearance," Supporting Statement, and related documentation submitted to OMB will be available for inspection and copying for a fee in the NRC Public Document Room at 1717 H Street NW., Washington, D.C.

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that neither of these alternative proposed rules will, if promulgated, have a significant economic impact on a substantial number of small entities. The alternative proposed rules affect only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 21. Since these companies are dominant in their service areas, this proposed rule does not fall within the purview of the Act.

First NRC-Proposed Rule (the Staff Rule)

The review and evaluation by the NRC staff of the information that has been developed over the past ten years on ATWS events and of the manner in which they should be considered in the design and safety evaluation of nuclear power plants is contained in the report "Anticipated Transients Without Scram for Light Water Reactors," NUREG–0460, Volumes 1 through 4. There are two primary factors in the staff's evaluation. The first is the degree of assurance that ATWS events can be prevented, which depends on the reliability of current reactor protection systems. The second is the capability of existing reactor designs to mitigate the consequences of ATWS events.

The reliability of current reactor protection systems has been estimated based on the operating experience to date and reliability analyses. However, the very high level of reliability required is difficult to demonstrate with confidence because it depends on accurately determining the rate of common cause failures. Common cause failures involve failures of multiple components resulting from a single cause or event. Reactor protection systems are carefully reviewed to identify and eliminate all but the most unlikely common cause failures. However, one connon cause failure in the reactor trip portion of the protection system of a commercial nuclear power reactor has occurred during approximately 1000 reactor-years of operating experience. The failure was detected during normal surveillance and corrected before any event requiring a reactor scram occurred. There has also been one partial failure to scram in a commercial power reactor, which occurred at low power and resulted in no core damage or radiation release.

Common cause failures have also occurred in other systems in nuclear

[1] See the memorandum of Chairman Joseph M. Hendrie to Commissioners Gilinsky, Bradford, and Ahearne, "ATWS," dated June 9, 1981. A copy is available for inspection and copying for a fee in the Commission's Public Document Room at 1717 H Street NW., Washington, D.C.

power plants and other potential common cause failures in reactor protection systems have been identified. Because of the low rate of occurrence of common cause failures, operating experience is not, and cannot be, sufficient to conclusively determine on a statistical basis whether reactor protection systems are reliable enough to make the probability of unacceptable consequences from ATWS events acceptably small. The prediction of common cause failures is as much art as it is science. System reliability analyses that attempt to predict the nature and frequency of common cause failures suffer from problems of completeness and accuracy, particularly when the desired failure rate is extremely small. While quantitative estimates of protection system reliability provide important information, the conclusion as to the adequacy of protection system reliability must be based on engineering judgment. The NRC has concluded that the reliability of current reactor protection systems has not been demonstrated to be adequate and most likely is not adequate.

The probability of severe consequences resulting from ATWS events is also affected by the capability of nuclear power plants to mitigate ATWS events. This capability varies depending on the design of the reactor system and the status of systems and the values of system process variables at the time the event occurs. The capability of a plant to mitigate ATWS events can be assessed by analysis. However, uncertainties in the design characteristics of the reactor, the probability of failure of the mitigating systems and the probability that the values of system process variables will be different from those assumed in the analysis all combine to produce uncertainty in the results. Therefore, the difficulty of demonstrating a capability to adequately mitigate ATWS events is similar to the difficulty of demonstrating that ATWS events can be prevented. Based on analyses performed to date, however, it is clear that, in most cases, present reactor designs have inadequate capability to mitigate the consequences of many postulated ATWS events should they occur.

Having concluded that improvements are needed to reduce the probability of severe consequences from ATWS events, the staff developed four alternatives, three of which would reduce this probability by increasing increments and would require increasing amounts of modifications. The alternatives were first described in Volume 3 of NUREG-0460 and again in slightly revised form in Volume 4. The intent of the proposed rule is to adopt a combination of the alternatives recommended in Volume 4 (except for one change for reactors designed by Westinghouse and licensed to operate before 1984). The proposed rule would implement the requirements in a different manner from that described in Volume 4 of NUREG-0460. The form of the requirements in the proposed rule is also different from that recommended in NUREG-0460 in that the proposed rule specifies acceptance criteria for ATWS mitigating systems while the required mitigating systems are specified in Volume 4.

Alternative 1 is to make no modifications at all. As discussed, the NRC has concluded that the reliability of current reactor protection systems is insufficient with respect to ATWS and that the probability of ATWS events is sufficiently great to warrant improvements. Therefore, this alternative is not represented in the proposed rule.

Alternative 2, as modified in the proposed rule, would increase the reliability of the reactor trip portion of reactor protection systems and improve the capability of existing systems to mitigate some ATWS events. Reliability of the reactor trip systems would be increased by the addition of supplementary protection systems that would be independent and diverse from the reactor trip portion of the current reactor protection systems. Diversity would be achieved by the use of components from different manufacturers, by the use of components having different principles of operation or power sources, and by the use of components in different operating modes (normally energized vs. normally deenergized). This alternative would not provide increased reliability of the reactivity control portion of the protection system, i.e., the control rods and control rod drives. However, in the case of reactors designed by General Electric it was proposed to increase the reliability of a portion of the control rod drive system, i.e., the control rod drive scram discharge volume. The capability to mitigate ATWS events would be improved by providing actuation circuitry that is separate from the reactor protection system for some existing systems such as primary system relief valves, turbine trip, and auxiliary feedwater in PWRs and the recirculation pump trip in BWRs. This alternative is very similar to the proposed rule offered by the utility group.

The staff proposed in Volume 4 of NUREG-0460 to implement only Alternative 2 for the ten older plants that began operation before late 1969. Because of their unique characteristics, the staff believed that more extensive modifications would not be appropriate for these plants. The proposed rule does not explicitly address these plants (except in the implementation schedule), but the intent is to consider any exemptions from the acceptance criteria of the proposed rule for these older plants based on analyses by the licensees and evaluations similar to those conducted under the Commission's systematic evaluation program (SECY-77-561 October 1977) in context with the overall safety of these facilities.

Alternative 3, as modified in the proposed rule, would increase the reliability of the reactor trip portion of the reactor protection system for some plants and provide for the mitigation of most ATWS events. The reliability of the protection system would be increased in the same manner as in Alternative 2. However, this increased reliability of the reactor protection system would not be required in plants that have a greater capability to mitigate ATWS events. The mitigation of most ATWS events in PWRs was expected to be accomplished as in Alternative 2, except that means would be required to isolate the containment early in an ATWS event upon detection of radiation released from failed fuel. The mitigation capability of BWRs was expected to be increased by providing automatic initiation of the Standby Liquid Control System and increase its flow capacity. Considering the state of design and construction, and a balancing of public safety benefits against economic cost the Commission proposes in this first rule that plants receiving an operating license before 1984 should be required to implement Alternative 3 as modified in the proposed rule.

Alternative 4, as modified in the proposed rule, would increase the reliability of the reactor trip portion of the reactor protection system of all plants and provide for the mitigation of almost all ATWS events. The reliability of the protection systems would be increased in the same manner as in Alternative 2. The mitigation of virtually all ATWS events was expected to be substantially increased by additional pressure relief capacity in the reactor coolant system. The mitigation capability of BWRs was expected to be increased by the addition of high capacity neutron poison injection systems. In balancing public safety benefits against economic cost, the Commission proposes in this first rule that these extensive design changes could only be practically incorporated in plants not near completion and not to be licensed before 1984.

The proposed requirements in Volume 4 of NUREG-0460 were in the form of specific design changes. The proposed rule also specifies the design changes required to improve the reliability of the protection system and the response for containment isolation, but the changes in mitigation capability are required through the specification of acceptance criteria, criteria for evaluation models, and mitigating system design criteria. The specification of criteria requires licensees and applicants to demonstrate that the designs of their plants are in compliance and thus provides more assurance that the safety objective is being attained. This form also allows the designer more flexibility in design and a greater potential for minimizing costs.

Although the ultimate safety objective is to limit the release of radioactivity to the environment, the acceptance criteria in the proposed rule are directed toward ensuring the integrity of the reactor coolant system and the reactor core following ATWS events. The staff recognizes that failure to satisfy these acceptance criteria does not necessarily result in severe radiological consequences and has considered the additional safety margin in developing the proposed rule. In formulating the proposed rule, the Commission has considered the need to compare for each plant the offsite doses that might result from ATWS events with 10 CFR Part 100 guidelines. Based on conservative generic calculations performed by the staff, there is reasonable assurance that calculated offsite doses from ATWS will be within the Part 100 dose guidelines if the acceptance criteria of the proposed rule are met. Accordingly, the Commission has decided that applicants and licensees will not be required to calculate the potential offsite radiological doses resulting from an ATWS event under § 100.11. If only these guidelines for calculated offsite doses were specified, the flexibility for the designer would be increased, but the attainment of the safety objective would be more difficult to demonstrate. If systems designs were specified, the flexibility of the designer would be reduced, and the demonstration that the safety objective had been attained would be generic rather than for specified plants. Prior attempts at such a generic demonstration have been unsuccessful, as discussed above.

The level of safety, that is, whether most or virtually all ATWS events can be mitigated, is specified through the criteria for acceptable evaluation models. Since the parameters in the evaluation model are uncertain to some degree and some may vary over the lifetime of the plant, the level of safety is determined to a large extent by the degree of conservatism in the parameters used in the evaluation models, which affect the conservatism of the calculated consequences of postulated ATWS events. The proposed rule specifies that realistic values of parameters may be used when the value is known with reasonable accuracy, but that parameters with large uncertainties must be conservatively treated. The intent is to obtain realistic analyses of the course of ATWS events, yet predict the consequences conservatively. In order to ensure that the consequences of most ATWS events will be within the acceptance criteria, the proposed rule specifies that the value used for parameters that vary over the lifetime of the plant (the most significant of these in the moderator temperature coefficient) must be a value that is not exceeded over most or virtually all of the plant lifetime. In the case of the moderator temperature coefficient, the value used in the evaluation model that was less negative than the value expected to be experienced during 90 or 99 percent of the design lifetime of the plant would ensure that the consequences of most or virtually all ATWS events would not violate the acceptance criteria.

Although improvements in the capability to mitigate ATWS events provide a significant increase in the level of safety, there is some uncertainty associated with this conclusion. This uncertainty derives from the uncertainty in the reliability of mitigating systems and in the evaluation models used to define them. Because of this uncertainty the staff believes that improvements in reactor protection system reliability should also be required. These modifications to present reactor protection systems, as with any modifications to a nuclear plant, have the potential for introducing unrecognized failure modes that could result in a decrease in the level of safety. A careful design process in conjunction with the quality assurance, verification, and test programs is necessary to ensure that this will not occur. However, the implementation of these improvements in reliability in some plants is to be accomplished within two years, and such a short design and installation schedule might compromise the design program. In plants such as those designed by Westinghouse, which have a capability to mitigate nearly all ATWS events and where the level of safety is already high, the Advisory Committee on Reactor Safeguards (ACRS) recommended omitting the requirement for improvements in the protection system reliability. Thus, the proposed rule allows the protection system improvements to be omitted if more conservative values of the parameters, such as moderator temperature coefficient are used in the evaluation models and the capability to comply with the acceptance criteria is demonstrated. In plants licensed after January 1, 1984 or later the time available to design and install the modifications to the protection system is sufficient to ensure that the design process would not be compromised and improvements in the protection systems of all of these plants is required by the proposed rule.

One plant modification that would be required by the proposed rule is already being implemented on boiling water reactors. In an order dated February 21, 1980, licensees of BWR plants were directed not to operate after December 31, 1980, without a recirculation pump trip installed. Licensees have also been directed (IE Bulletin No. 80-17 dated July 3, 1980, and NUREG-0737, "Clarification of TMI Action Plan Requirements") to ensure that operating procedures and operator training address the actions to be taken in the plants as now designed if an ATWS did occur. These requirements are prudent measures that will reduce the risks from ATWS events during the interim period before the plant modifications determined by the Commission to be necessary can be installed.

In particular cases, additional requirements or earlier implementation may be appropriate. For example, candidates would be those existing nuclear power plants that are considered to be at high risk sites owing to a combination of population density, meteorological conditions and other factors.

The proposed rule would provide for implementation of the requirements in stages in order to gain the greatest increase in safety in the shortest time and at the least cost. The modifications to improve the reliability of the protection system and the mitigating system actuation circuitry would be required within two years of the effective date of the rule. In order to accomplish this, descriptions of the modifications are to be submitted for review by the NRC within one year of the effective date of the rule.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy

Reorganization Act of 1974, as amended, and section 553 of title 5 of the United States Code, notice is hereby given that adoption of the following amendments to 10 CFR Part 50 is contemplated.

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for 10 CFR Part 50 reads as follows:

Authority: Secs. 103, 104, 161, 182, 183, 68 Stat. 936, 937, 948, 953, 954, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233); secs. 202, 206, 88 Stat. 1244, 1246 (42 U.S.C. 5842, 5846), unless otherwise noted. Section 50.78 also issued under sec. 122, 68 Stat. 939, 42 U.S.C. 2252). Sections 50.80–50.81 also issued under Sec. 184, 68 Stat. 954, as amended, Secs. 50.100–50.102 issued under sec. 186, 68 Stat. 955; (42 U.S.C. 2236). For the purposes of sec. 223, 68 Stat. 958, as amended; (42 U.S.C. 2273), § 50.54(i) issued under sec. 161i, 68 Stat. 949; (42 U.S.C. 2201(i)), and §§ 50.70–50.71 and § 50.78 issued under sec. 161o, 68 Stat. 950, as amended; (42 U.S.C. 2201(o)) and the Laws referred to in Appendices.

2. A new § 50.60 is added to read as follows:

§ 50.60 Acceptance criteria for protection against anticipated transient without scram events for light-water-cooled nuclear power plants.

(a) *Definitions.* (1) "Anticipated Transient Without Scram" (ATWS) means an anticipated operational occurrence as defined in Appendix A of this part followed by the failure of the reactor protection system specified in General Design Criterion 20 of Appendix A of this part.

(2) "ATWS evaluation model" means the calculational framework for evaluating the behavior of the nuclear power plant during a postulated ATWS event.

(3) "ATWS mitigating systems" means those systems including associated controls, instruments, power supplies and other systems assumed to function when evaluating the behavior of the nuclear power plant following an ATWS event.

(b)(1) *Acceptance Criteria.* Each light-water-cooled nuclear power plant must be designed, constructed, and operated so that the consequences of postulated anticipated transient without scram (ATWS) events calculated in accordance with an ATWS evaluation model approved pursuant to paragraph (b)(4) of this section conform to the following criteria:

(i) *Primary system pressure.* The calculated reactor coolant system (RCS) pressure and temperature resulting from postulated ATWS events must be limited so that either (A) the calculated maximum primary stress anywhere in the RCS pressure boundry does not exceed that permitted by the "Level C Service Limit" as defined in Article NB-3000 of Section III of the ASME Boiler and Pressure Vessel Code[1] and the calculated deformation of RCS components is limited so that the operability of components necessary to safely bring the reactor to and maintain it at a cold shutdown condition is not impaired, or (B) the integrity or operability of RCS components must be demonstrated based on conservative assessments of tests conducted to determine the integrity or operability of components under the conditions accompanying postulated ATWS events and based on the likely condition of the components over their design life.

(ii) *Fuel integrity.* The calculated damage to the reactor core as a consequence of postulated ATWS events, including oscillations of power and flow, must be limited to ensure that the core geometry is not distorted to an extent that would impair core cooling or safe shutdown.

(iii) *Radioactivity release.* The calculated release of radioactivity from the fuel rods to the reactor coolant system during postulated ATWS events must not exceed one percent of the radioactivity within the fuel rods of a pressurized water reactor or ten percent of the radioactivity within the fuel rods of a boiling water reactor.

(iv) *Containment.* The calculated containment pressure, temperature, and humidity resulting from postulated ATWS events must not exceed the design values of the containment structure and components or the contained mitigating systems, equipment and components. For boiling water reactor pressure suppression containments, the relief or safety valve discharge line flow rates and suppression pool water temperatures must be limited so that steam quenching instability will not result in destructive vibrations.

(v) *Long-term shutdown and cooling.* The reactor design must permit the reactor to be safely brought to and maintained at a cold shutdown condition following postulated ATWS events without insertion of control rods.

(2) *Evaluation Model Criteria.* (i) ATWS evaluation models must, with reasonable accuracy or acknowledged conservatism, represent the actual characteristics of the facility modeled and each significant physical phenomenon that would occur in the reactor and related systems during the course of the modeled event. Evaluation models must represent the effect of the failures in mitigating systems that are a direct consequence of the ATWS event being modeled. For facilities issued operating licenses on or after January 1, 1984, and not standardized to a facility at the same site that was issued an operating license before January 1, 1984, evaluation models must also represent the effect of the likely random single failures of active components in mitigating systems.

(ii) The value of parameters that vary over the lifetime of the facility or represent the characteristics of mitigating systems that are permitted by procedure to be inoperable for any period during operation must be selected so that values that would result in violation of the acceptance criteria would not be expected to occur during

(A) Most of the design lifetime of facilities issued operating licenses before January 1, 1984 or of facilities standardized to a facility at the same site that was issued an operating license before January 1, 1984.

(B) Almost all of the design lifetime of facilities issued operating licenses on or after January 1, 1984, except facilities standardized to a facility at the same site that was issued an operating license before January 1, 1984.

(3) *Mitigating System Criteria.* ATWS mitigating systems must be independent, separate and diverse from the reactor protection system. ATWS mitigating systems must be designed, qualified, monitored and periodically tested to ensure continuing functional capability under the conditions accompanying postulated ATWS events, including natural phenomena such as earthquakes, storms, tornadoes, hurricanes, and floods expected to occur during the design life of the plant. ATWS mitigating systems must be automatically initiating when the conditions monitored reach predetermined levels and continue to perform their function without operator action unless it can be demonstrated that an operator would have adequate information and would reasonably be expected within the time available to take the proper corrective action.

(4) *Evaluation models.* Each applicant or licensee shall submit evaluation models as defined in paragraph (b)(2) of this section, together with the description and results of the analyses and test necessary to verify the validity of the assumptions made in preparing such evaluation models to the Nuclear Regulatory Commission for approval by (within six months of the effective date of the rule) or prior to issuance of an operating license, whichever is later.

[1] See § 50.55a for approval of this incorporation by reference.

(5) *Plans for compliance.* Each applicant or licensee shall submit a description of all measures to be taken to ensure compliance with the criteria set forth in paragraph (b)(1), (b)(2) and (b)(3) of this section together with such proposed changes in technical specifications and license amendments as may be necessary to ensure compliance with these criteria to the Nuclear Regulatory Commission as follows:

(i) For all light-water-cooled nuclear power plants for which operating licenses have been issued on or before August 22, 1969, no later than (eighteen months after the effective date of the rule).

(ii) For all light-water-cooled nuclear power plants for which operating licenses have been issued after August 22, 1969, no later than (one year after the effective date of the rule) or prior to issuance of an operating license, whichever is later.

(6) *Implementation.* Each applicant or licensee shall implement those measures necessary to ensure compliance with the criteria set forth in paragraph (b)(1) of this section on the following schedule:

(i) For all light-water nuclear reactor power plants for which operating licenses have been issued on or before August 22, 1969, by dates agreed upon with the NRC. These dates must be submitted for approval not later than (three years after the effective date of the rule).

(ii) For all light-water-cooled nuclear reactor power plants for which operating licenses have been or may be issued after August 22, 1969, but before (three years after effective date of the rule), all modifications shall be completed prior to startup following the first refueling that begins (three years after effective date of the rule).

(iii) For all light-water-cooled nuclear reactor power plants licensed on or after (three years after effective date of the rule), all modifications shall be completed prior to issuance of an operating license.

(c) *Additional requirements*—(1) *Actuation:* In addition to those requirements set forth in paragraph (b) of this section, each light-water-cooled nuclear power plant except as provided in paragraph (c)(2) of this section, must be provided with:

(i) Actuation circuitry for ATWS mitigating systems that is independent and diverse from the reactor protection system;

(ii) Prompt automatic containment isolation initiated by a significant source of radiation in the containment resulting from failure of the fuel rods following postulated ATWS events;

(iii) Those modifications necessary to reduce the common mode failure potential of the control rod scram discharge volume in plants designed by the General Electric Company including diverse scram discharge volume level sensing devices; and

(iv) Those modifications necessary to provide a supplementary reactor trip system that is diverse from the reactor trip portion of the current reactor protection system.

(2) *Exemption.* Pressurized light-water-cooled nuclear power plants issued operating licenses before January 1, 1984 or standardized to a facility at the same site that was issued an operating license before January 1, 1984 need not comply with the requirements of paragraph (c)(1)(iv) if the facility conforms to the requirements of paragraph (b) of this section except that the fraction of the design lifetime used to determine the value of parameters must be greater than that specified in paragraph (b)(2)(i) of this section.

(3) *Submittal.* A description of the measures together with such proposed changes in technical specifications or license amendments as may be necessary to ensure compliance with the criteria set forth in paragraph (c)(1) of this section must be submitted to the Nuclear Regulatory Commission no later than (nine months after the effective date of the rule) or prior to issuance of an operating license, whichever is later.

(4) *Implementation.* Those measures required under paragraphs (c)(1) of this section must be completed:

(i) For all light-water cooled nuclear reactor power plants for which operating licenses have been or may be issued after August 22, 1969 but before (two years after effective date of the rule), all modifications shall be completed prior to startup following the first refueling that begins (two years after effective date of the rule).

(ii) For all light-water cooled nuclear reactor power plants licensed on or after (two years after effective date of the rule), all modifications shall be completed prior to issuance of an operating license.

(d) *Dose calculations.* Applicants or licensees are not required to calculate the potential offsite radiological doses resulting from an anticipated transient without scram event under § 100.11 of this chapter.

Second NRC-Proposed Rule (the Hendrie Rule)

The essence of the second NRC-proposed rule is that power reactor licensees would be required to implement a reliability assurance program to seek out and rectify reliability deficiences in those functions and systems that prevent or mitigate ATWS accidents. To cover the possibility that the reliability assurance programs might fail to correct an obscure reliability defect, some additional requirements for ATWS mitigation would be selectively mandated. These improvements in ATWS tolerance of reactor plants have been chosen to afford an opportunity to learn from experience without incurring a substantial likelihood of an unacceptable radiological release.

The NRC is exploring the possibility that the regulation of reactor safety may evolve toward regulating the process by which licensees ensure public heath and safety and away from licensing the details of plant design and operation. Programs like the reliability assurance program in this proposed rule offer promise of growing into a formal, auditable way the NRC can determine that licensees are doing a satisfactory job of ensuring public health and safety. A number of diverse regulatory initiatives are supportive of this tread. Among them are the requirements on licensee staffing and organization, the proposal that licensees employ probabilistic risk assessment methods as design and operations management tools, and the pilot studies of independent design reviews.[2]

The necessity for and content of the proposed rule is based on (1) operating experience to date with power reactor scram systems, (2) system reliability analysis, (3) the qualitative findings of reactor risk assessment, and (4) ATWS accident analysis.

There has been one partial failure of the scram system in a commercial power reactor. It occurred at Browns Ferry Unit 3 on June 28, 1980. Although the particular scram system failure mode that caused the event is very unlikely to cause a severe radiological release accident, the event and the reviews resulting from it revealed a number of reliability deficiencies in the BWR scram systems. These are now being rectified by the industry subject to the review and approval of the NRC staff. One objective of the proposed reliability assurance program is to institutionalize within the licensed industry the thorough evaluation and implementation

[2] See, for example, "NRC Action Plan Developed As A Result of the TMI-2 Action Plan" NUREG-0660, "Policy on Proceeding With Pending Construction Permit and Manufacturing License Applications," SECY-81-20D, and "Use of Independent Design Reviews (IDEs) in the Regulatory Process," SECY-81-16. Copies of these references are available for public inspection in the Commission's Public Document Room, 1717 H Street NW., Washington, D.C.

of the lessons of experience with functions important to ATWS prevention or mitigation.

Reliability deficiencies in safety systems differ substantially in the kind and frequency of opportunities to detect and repair them. Some faults are self-announcing and thus elicit prompt repair. Others show up in each surveillance test. Some faults may not be revealed by routine surveillance tests. For instance, the reliability defect responsible for the partial scram failure at Browns Ferry could not have been detected in routine surveillance tests of the scram system.

System reliability calculations by the Electric Power Research Institute and others have shown that component failures of reactor scram systems that are detected and corrected in each surveillance test are very unlikely to cause ATWS events. Other system failure modes can only be detected in some but not all surveillance tests. Still others show up only in some or all genuine demands upon the system. Some reliability defects cannot be detected even in genuine system demands unless triggered by other failures. Examples of the latter category are the hydraulic design deficiencies in the BWR scram discharge system revealed by the incident at Browns Ferry. Such blind spots in the experience base for safety systems can conceal serious flaws in reliability. Thus a second objective of the reliability assurance program is to conduct a thorough analysis of the startup test program, the surveillance test program, and the record of system functional experience to identify and—where feasible—close loopholes through which design deficiencies, construction deficiencies, vulnerability to test or maintenance error, or component failures might escape detection and thus correction for considerable periods of time.

Studies initiated in response to the Browns Ferry partial scram failure indicated that two auxiliary systems [3] that serve the scram sytem as well as other systems, could have caused partial or complete scram failures. This discovery is suggestive of a class of common cause failures that might compromise the safety of a reactor. Failures in auxiliary systems might cause the initiating transient as well as degrade the reliability of the scram system, or they might contribute to the scram failure and also could compromise the availability of one of the systems required to mitigate an ATWS event, or both.

Thus, a third objective of the reliability assurance program is to search out and evaluate the potential common cause failures that might contribute to failure in two or more systems whose reliability is important to ATWS accident sequences. This search should embrace not only auxiliary systems but also human factors via test, maintenance, and operations; technical specifications dealing with equipment availability; and environmental conditions in the plant.

A fourth objective of the reliability assurance program is to search out and evaluate the susceptibility of the redundant divisions of each safety system important to ATWS prevention or mitigation to common cause failure. Concern with common cause failure modes of the scram system has been central to the history of the ATWS controversy.

A common cause failure of an electrical nature has already occurred in a reactor scram system in a commercial nuclear power plant (Kahl reactor) that could have resulted in its failure to operate on demand. That failure was detected during normal surveillance and rectified. A similar common cause failure was detected and corrected in the startup testing of the Monticello reactor. Estimates of the upper limits of the frequency of ATWS events for the commercial power reactor industry are of the order of 10^{-3} per reactor year. The NRC staff has concluded that operating experience is not sufficient to determine conclusively on a statistical basis whether reactor scram systems are reliable enough to make the probability of unacceptable consequences from ATWS events sufficiently small.

The improvements emanating from the proposed reliability assurance program will make ATWS accidents less likely and the systems that mitigate ATWS events more reliable. Nevertheless, it is necessary to ensure that mitigating systems will render the outcome of most ATWS events acceptable. The principle of defense in depth calls for reactor plants to be designed and operated in such a way that a rare ATWS accident can be tolerated.

The requirements for ATWS tolerance in light-water cooled commercial power reactors are intended to afford an opportunity to learn from experience without placing the public health and safety in jeopardy. The first occurrence of an ATWS precursor due to any particular failure mode will result in studies like those now being made in response to the Browns Ferry incident. These can be counted on to make a recurrence of that failure mode much less likely in the future. *

Calculations of the expected consequences of very severe reactor accidents have been made in the *Reactor Safety Study* (WASH-1400)[4] and other studies. The results indicate that the accidents that could realistically be expected to result in lethal radiation doses outside the plant site are those denoted as release category 1, 2 or 3 accidents in the notation of WASH-1400. These are also the accidents that are expected to cause substantial offsite property damage.

Studies of ATWS accidents in pressurized water reactors (PWRs) suggest that only a small percentage of reactor scrams are limiting transients. That is, only a small fraction of the opportunities for ATWS accidents occur under circumstances that most severely challenge the ATWS tolerance of the plant. In addition, the qualitative findings of PWR risk assessment studies suggest that even the most limiting classes of ATWS accidents in PWRs are unlikely to produce a release category 1, 2 or 3 radiological outcome.

In boiling water reactors (BWRs) a substantial fraction of scrams take place under circumstances that can lead to a limiting transient. BWRs are least forgiving of those ATWs events in which the reactor is isolated. Even if reactor isolation does not cause the transient in the first place, the effects of a failure to scram are likely to trigger reactor isolation. Furthermore, BWR risk assessment studies suggest that ATWS accidents may give rise to release category 1, 2 or 3 (as described in WASH-1400) outcomes.

These arguments suggest that PWRs may already achieve the minimum ATWS tolerance necessary to supplement the reliability assurance program, whereas improvements should be mandated for BWRs to strengthen their provisions for ATWS mitigation. However, a more careful analysis of ATWS-tolerance is required in the proposed rule to provide the basis for and form of actions to be taken by licensees.

In PWRs, the limiting transient with respect to ATWS is a complete interruption in the delivery of feedwater to the steam generators at full power. Should the scram fail to shut the reactor

[3] The auxiliary systems of note are the vent system serving the scram discharge volumes, and the compressed air system, serving the air-operated scram valves.

[4] Microfiche copies are available for purchase from the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

down, the continued power generation and the declining heat removal, as the secondary coolant boils away, causes a surge in pressure of the reactor coolant. The severity of this pressure excursion is a sensitive function of the moderator temperature coefficient, the capacity of the relief valves attached to the reactor coolant system, and the speed with which the auxiliary feedwater system starts. The pressure surge will subside as the power decreases due to the increasing moderator temperature. Subsequent reactor coolant replenishment and reactivity control is provided by the high pressure injection (HPI) system, which pumps cooling water containing a reactivity poison into the reactor coolant system.

The most severe test of the ATWS tolerance of a PWR lies in its survival of the pressure excursion and in the successful start of the auxiliary feedwater and high pressure injection systems. The possible outcomes of the pressure excursion are (1) the reactor coolant system and interfacing equipment are undamaged, (2) the reactor coolant system remains intact but instruments on the pressure boundary fail or the valves for the HPI system are damaged, (3) the reactor coolant system is ruptured producing a loss-of-coolant accident (LOCA) to containment, (4) steam generator tubes rupture causing a primary-to-secondary LOCA or a LOCA to other interfacing systems, or (5) combinations of (2), (3) or (4). The first outcome is clearly preferred. The second outcome makes it clear that care must be taken to ensure that the operators have sufficient information about the status of the reactor to manage the recovery. Should the HPI pressure boundary valves all seize in the closed alignment, the core will melt. This is one of several paths from ATWS to a contained core melt accident. A LOCA to containment is likely to be mitigated by the Emergency Core Cooling System (ECCS) even though the initial pressure conditions are outside the design envelope for ECCS analysis. Thus no core melt is expected (although a contained core melt is a remote possibility), and a core melt with missile damage to containment is a still more remote possibility. Steam generator tube rupture can provide a leakage path to the outside atmosphere that bypasses containment. However, ECCS is likely to be successful, so the core would not melt. All but one steam generator can very probably be isolated, thus terminating a minor release.

The severe release category 1, 2, or 3 events occur only for a core melt and a gross above-ground failure of containment. This is not among the more probable outcomes of even the most severe and damaging pressure excursions associated with ATWS in PWRs.

Analysis of ATWS transients by the NRC staff and the reactor suppliers suggest that Westinghouse reactors have sufficient relief capacity so that pressure excursions expected of limiting ATWS transients will not be damaging, provided that the auxiliary feedwater system starts promptly. Combustion Engineering and Babcock and Wilcox reactors may be subject to severe pressure excursions even with prompt start of the auxiliary feedwater system, should the ATWS accident take place when the moderator temperature coefficient is unfavorable. The NRC staff has argued in NUREG-0460 that these plants should install additional relief capacity to improve their ATWS tolerance. The industry has argued that such modifications are very expensive, will produce substantial occupational exposures to radiation to those installing them, and are unnecessary because the plants already have sufficient tolerance of the pressure excursion, according to their analyses in proprietary reports.

The NRC, in reassessing its position, has concluded that the minimum ATWS tolerance necessary to complement the reliability assurance program does not dictate additional pressure relief capacity in CE and B&W plants in light of the several mitigating factors noted above. However, there are a number of other safety-related incentives to alter the provisions for reactor coolant pressure reduction or relief in PWRs. These include deliberate depressurization to enable low-pressure safety injection in small LOCAs and feedwater transients with scram, to avoid the melt-through of reactor vessels while at elevated pressure, and to enable the ECCS accumulators to extend the point of no return for the restoration of AC power in station blackout accidents. The NRC expects to take up the case for and against altered pressure relief provisions for PWR reactor coolant systems in the forthcoming rulemakings on severe accidents.

The required ATWS tolerance of PWRs rests: (1) upon the prompt start of the auxiliary feedwater system, (2) the availability of instruments necessary for the operators to diagnose the ATWS accident sequence and successfully maneuver the plant to minimize the release of radiation, (3) the training of operators, (4) the availability of the high pressure injection system, and (5) the integrity of reactor coolant pressure boundary valves through which a LOCA would bypass containment and could not be isolated.

In some PWRs, the very rapid autostart of the auxiliary feedwater system following a feedwater transient can overcool the reactor if the scram is successful. In such plants, the rapid start logic may be interlocked to take place only if the scram fails. However, such interlocks must not degrade the reliability of the auxiliary feedwater system for the more frequent loss-of-feedwater transients in which the scram is successful and in which a delayed autostart of auxiliary feedwater is appropriate. The identification of the required instrumentation and the training of operators may be made a part of the reliability assurance program, and the verification that the instruments and the critical pressure boundary valves on the reactor coolant system have the required tolerance for the limiting pressure excursions would be part of the ATWS tolerance requirements.

The moderator temperature coefficient, which strongly influences the severity of the reactor coolant pressure excursion for limiting ATWS transients, is at its least favorable value during the early months of operation with the first fuel load. The early months of plant operation are also characterized by a higher-than-average frequency of transients and safety system failures as the plant is shaken down and the plant personnel gain experience with the equipment. Therefore, much of the risk associated with ATWS accidents is expected to be concentrated in the first months of plant operation. One mitigating factor is the less-than-equilibrium inventory of fission products accumulated in the fuel at this time. Nevertheless, PWR reactor licensees would be required to propose and implement particularly stringent limiting conditions of operation in the technical specifications to constrain operation when combinations of the unavailability of mitigating or preventive equipment, the prevailing moderator temperature coefficient, and the power level encroach upon the tolerance of the plant for the pressure excursions to be expected of limiting ATWS transients.

In large, modern boiling water reactors, a transient with failure to scram from full power is very likely to cause, or may follow, the isolation of the reactor, notably a trip of the main steam isolation valves. If the reactor coolant recirculation pumps continue to run, the power level will remain high and a severe pressure excursion will take place. Even if the reactor coolant system survives the pressure surge, the very high steam flow will rapidly heat the suppression pool and pressurize the containment. In addition, the high-pressure coolant injection (HPCI) may not suffice to cool the core; overheating and core damage may follow. Ultimately the containment is expected to rupture due to overpressure while the core sustains damage. Continued core coolant replenishment is questionable after containment rupture. A large radiological release is a plausible outcome. A necessary mitigating feature is thus a prompt automatic trip of the recirculation pumps to avoid the pressure excursion and diminish the power and the consequent steam flow to the suppression pool. Given a trip of the recirculation pumps, the reactor power will stabilize at roughly 30% power until the reactor coolant boils down and steam bubbles (void formation) in the core throttle the chain reaction. Thereafter, a static or oscillatory equilibrium will be maintained in which the reactor sustains the average power necessary to boil off however, much reactor coolant is delivered, up to about 30% power. Analysis shows that HPCI or main feedwater can adequately cool the core to avoid extensive core damage. However, the power delivered to the suppression pool will be greater than the pool cooling system can dissipate. Therefore, containment overpressure failure remains a distinct possibility unless the reactor is shut down, either by control rod insertion or by liquid reactivity poison injection. Well before the containment is significantly pressurized, the suppression pool will approach saturation, and the steam condensation will become unstable. Chugging steam condensation may threaten containment intergrity or pressure suppression and thus shorten the time available to shut down the reactor without unacceptable consequences. In limiting transients, the failure of the main feedwater system may be the initiator of, or companion of the initiating event. The HPCI is a single-train system. The fault or human error that precipitates the initial transient might also disable the HPCI. In addition, system reliability analyses have indicated that HPCI may fail or be unavailable in as many as from 1% to 10% of the cases in which a demand is made of the system. This may be insufficient reliability for the mitigation of a potentially serious accident having a frequency of occurrence that might be as high as once in a thousand reactor years. A second diverse system, the Reactor Core Isolation Cooling system (RCIC) should be expected to autostart and run, delivering coolant to the reactor. The flow rate delivered by the RCIC is lower than that of the HPCI. If the RCIC is the sole operative means of replenishing reactor coolant, the adequacy of core cooling, rather than the heat deposited in the suppression pool, is likely to be the factor limiting the time allowed to shut down the reactor without unacceptable consequences. The RCIC can successfully cool the reactor once it is shut down, and it can slow the boiloff of reactor coolant in the reactor.

The NRC has concluded that the liquid reactivity poison injection system in large, modern BWRs must have a start time and poison injection rate such that either of two redundant trains of high-pressure reactor coolant replenishment systems, either of which may be expected to be available under ATWS conditions, can successfully mitigate ATWS transients. The two trains may be the HPCI and RCIC.

The criteria of successful mitigation are: (1) The containment temperature and pressure must remain within the design envelope, (2) the core must retain coolable geometry, and (3) neither prompt fatalities nor serious offsite property damage are predicted by analyses whose conservatism is compatible with that employed in WASH-1400.[5]

Concern has been expressed that the RCIC, though capable of meeting these success criteria, does not prevent the automatic depressurization of the reactor coolant system. Operator action is necessary in less than ten minuted to override the automatic depressurization or to throttle low pressure ECCS should the depressurization occur. The NRC staff does not wish to force an alteration of the logic governing the automatic depresurization system (ADS) which might compromise the reliability of the ADS in non-ATWS events. Options to resolve these competing concerns will be evaluated by the NRC staff during the comment period. We are interested in receiving comments on the potential effects of the three proposed rules on this subsystem (high pressure makeup) of the BWR design.

Several factors complicate the analysis of the ATWS-tolerance of BWR plants. The delivery of main feedwater, which may be available in some ATWS accident sequences, may dilute liquid poison and increase the power level in ATWS events, thus threatening successful mitigation. In some sequence variants, operators might be tempted to depressurize the reactor to enable low pressure reactor coolant injection but, in so doing, disable turbine-driven coolant injection systems or otherwise compromise possible avenues of successful ATWS mitigation. The reliability assurance program must entail a thorough investigation of such ATWS accident sequences, of the instrument indications available, and of the possible range of operator actions. Operator training should familiarize operators with the optium strategies and alert them to serious errors that could occur in dealing with ATWS accidents.

BWR reactor operators may be subject to a strong disincentive to actuate the Standby Liquid Control (SLC) system because of the costly nature of spurious SLC actuations. They may also be inclined to override an autostart of the SLC if they doubt that an ATWS indication is genuine or the failure of the scram system is irreparable. The NRC recognizeas the ligitimacy of the concern with the cost of spurious SLC actuation

To deal with these conflicting concerns, the NRC proposes to require the automatic start of the SLC system under circumstances diagnosed to be ATWS sequences. Licensees are free to employ reliability engineering methods to minimize the likelihood of spurious actuations under non-ATWS circumstances provided these provisions do not compromise the reliability of the essential SLC safety function in genuine ATWS sequences.

In light of the analysis and operator training associated with the reliability assurance program, it is not deemed necessary to preclude provisions for manually overriding the autostart of the SLC. As part of the reliability assurance program, a thorough analysis is to be made of the circumstances in which an operator might be tempted to override a genuinely needed SLC actuation. Consideration should be given to improved instrumentation if the correct diagnosis of such sequences is ambiguous. Operators must be trained to give first priority to safety rather than to the availability of the plant for power generation. The anticipation that repeated manual scrams or quick fixes in the control cabinets may succeed in inserting the control rods would be an unacceptable justification for overriding SLC actuation.

In conjunction with this form of the rule, the NRC does not deem it necessary that the SLC meet the single-failure criterion as well as the indicated

[5] Microfiche copies are available for purchase from the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

success criteria. In the very unlikely event of an ATWS event and a failure of automatic and manual starts of the SLC system, a fallback strategy is available through manual rod insertion and intervention in the reactor protection system control cabinets. Nevertheless, the SLC must not depend upon a single division of an auxiliary system the failure of which would also compromise the reliability of the scram system or of the recirculation pump trip or precipitate the initiating transient.

BWRs must also operate under specified Limiting Conditions of Operation that constrain power generation under circumstances in which equipment unavailability compromises the reliability of systems important to ATWS prevention or mitigation.

The older lower-power-level reactors may differ significantly in the levels of ATWS-tolerance provided. These plants would be required to submit analyses of their ATWS-tolerance for review by the NRC staff.

The dual approach of ATWS tolerance and the reliability assurance program provides defense in depth. Each allows the other to be implemented without highly conservative margins. The margin provided by ATWS-tolerance allows realistic cost-benefit considerations to govern the selection and schedule of implementation for fixes suggested by the reliability assurance program.

The very costly accident at Three Mile Island has demonstrated that the protection of a licensee's investment in a reactor plant provides a powerful economic incentive to search out and correct reliability defects in the functions that protect a reactor core from damage. These economic considerations, together with a realistic evaluation of offsite risks affecting public health and safety, are sufficient to determine the scope and schedule of the more expensive or intrusive alterations in plant operation or design emerging from the reliability assurance program.

The reliability assurance program is not to be a paper study to demonstrate to the NRC staff that the plant is already safe enough. The role of probabilistic evaluations is secondary to the qualitative search for and evaluation of specific types of reliability defects. The reliability assurance program is not intended primarily to assist NRC staff review. Rather, it is to be integrated into the conduct of plant management, personnel training, and the conduct of operations. It is intended to strengthen the responsibility for safe design and operation of the plant resting with the licensee and to relieve the NRC staff of much of the detailed involvement in experience review and the selection of procedural or hardware backfits in the context of ATWS risks. For this reason, the proposed rule emphasizes criteria for the sound implementation of the reliability assurance program and limits the staff review to these criteria, together with the conventional review and approval of the license amendments associated with changes in design or operation.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and section 553 of title 5 of the United States Code, notice is hereby given that adoption of the following amendments to 10 CFR Part 50 is contemplated.

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for 10 CFR Part 50 reads as follows:

Authority: Secs. 103, 104, 161, 182, 183, 68 Stat. 936, 937, 948, 953, 954, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233); secs. 202, 206, 88 Stat. 1244, 1246 (42 U.S.C., 5842, 5846), unless otherwise noted. Section 50.78 also issued under sec. 122, 68 Stat. 939, 42 U.S.C. 2152. Sections 50.80–50.81 also issued under sec. 184, 68 Stat. 954, as amended; (42 U.S.C. 2234). Secs. 50.100–50.102 issued under sec. 186, 68 Stat. 955; (42 U.S.C. 2236). For purposes of sec. 223, 68 Stat. 958, as amended; (42 U.S.C. 2273), § 50.54(i) issued under sec. 161i, 68 Stat. 949; (42 U.S.C. 2201(i)), and §§ 50.70–50.71 and § 50.78 issued under sec. 161o, 68 Stat. 950, as amended; (42 U.S.C. 2201(o)) and the Laws referred to in Appendices.

2. A new § 50.60 is added to read as follows:

§ 50.60 Standards for the reduction of risk from Anticipated Transients Without Scram (ATWS) events for light-water-cooled nuclear power plants.

Each light-water-cooled commercial power reactor licensee shall establish and maintain a reliability assurance program for functions associated with the prevention and mitigation of Anticipated Transients Without Scram (ATWS) employing state-of-the-art methods and procedures to identify vulnerabilities to failure. Each licensee is responsible for the implementation of cost-effective improvements to reduce ATWS risk. Defense in depth must be maintained by operating commercial power reactors only in modes that afford an opportunity to learn from experience with ATWS events without severe radioactivity releases. Specific acceptance criteria are delineated below.

(a) *Initial reliability assurance program.* The initial reliability assurance program must include an analysis and classification of the principal determinants of the radiological severity of each class of ATWS accident sequences in terms of the initial plant conditions, the type of initiating transient, the failure mode of the reactor protection system, and the state of operability or inoperability of other active systems affecting the outcome. This analysis must be employed in each of the following elements of the reliability assurance program:

(1) Training of licensed reactor operators in the diagnosis and prognosis of the several ATWS accident sequences. Operators must be trained to make productive use of their time during ATWS accidents to effect mitigation. Consideration must be given to improving instrumentation, displays, and emergency procedures to minimize the likelihood that misdiagnosis or delayed diagnosis of ATWS sequences may substantially increase the radiological severity of the outcome.

(2) An analysis of hypothetical errors in or erroneous departures from proper test and maintenance procedures for systems whose reliability is important to ATWS prevention or mitigation. Consideration must be given to improved designs, test equipment, procedures, and personnel training to minimize the likelihood that the reliability of these systems will be compromised by errors in test and maintenance.

(3) An analysis of the blindspots in the experience base with systems important to ATWS prevention or mitigation through which reliability defects might escape detection for considerable periods of time. Hypothetical reliability deficiencies must be classified by (i) kind (design deficiency, construction deficiency, vulnerability to test or maintenance error, active or passive failure), (ii) affected components or subsystems, (iii) severity of the reliability deficiency, and (iv) the frequency and kind of opportunity to detect the deficiency. A test program covering startup or one-time-only tests, tests associated with periodic plant overhauls, and inservice surveillance tests must be developed and implemented so that the mean time to detect the deficiencies is reduced to the extent reasonably achievable.

(4) An analysis of the susceptibility of the plant to common cause failures of two kinds: those in which a single root cause degrades the reliability of redundant divisions of a safety system

important to ATWS prevention or mitigation, and those in which a single root cause degrades the reliability of two or more systems whose concurrent failure contributes to a severe ATWS accident sequence. The kinds of root causes to be considered are those listed in paragraph (a)(3)(i) of this section. Consideration must be given to improved design or operation to reduce vulnerability to common cause failures.

(b) *Continuing reliability assurance program.* Each commercial power reactor licensee shall maintain a continuing reliability assurance program for functions important to ATWS prevention and mitigation that includes the following:

(1) Configuration control for designs, procedures, and technical specifications to assure consistency with the initial reliability assurance analyses.

(2) Procedures for updating affected portions of the initial reliability assurance analysis for, and prior to, departures from the controlled design, procedures, or technical specifications. Applications for license amendments to implement these changes must include a brief analysis of the impact of the change on the reliability of systems important to ATWS risk.

(3) An experience feedback system to review operational and test data on relevant systems in the licensed plant and the relevant experience at plants having a similar system design. Each operational occurrence must be reviewed for clues to oversight or errors in the reliability assurance analyses. The initial reliability assurance analyses and cost-benefit analyses based thereon are to be updated when the experience feedback system reveals oversights or limitations in these studies.

(c) *Design and operation for ATWS tolerance.* (1) Boiling water reactor licensees receiving an operating license after August 22, 1969, shall:

(i) Provide equipment to trip automatically the reactor coolant recirculation pumps under conditions indicative of an ATWS event,

(ii) Provide equipment to automatically deliver liquid reactivity poison so that either of two independent reactor coolant replenishment system trains expected to be available during ATWS events can successfully bring the reactor to stable hot shutdown. The poison injection system must not depend for its function on a single division of an auxiliary system whose failure could precipitate the transient, degrade the reliability of the scram system, or defeat the recirculation pump trip, and

(iii) Provide a reliable scram discharge volume system.

(2) Pressurized water reactor licensees receiving an operating license after August 22, 1969, shall:

(i) Provide for the prompt, automatic start of the auxiliary feedwater system under circumstances indicative of a transient entailing loss of main feedwater and a failure to scram,

(ii) Ensure that the instruments necessary for the diagnosis of and recovery from ATWS accident sequences will not be disabled by the effects of such accidents, and

(iii) Ensure that those reactor coolant system pressure boundary valves through which high-pressure injection can reach the reactor remain functional after limiting ATWS transients and those valves whose integrity is essential to the avoidance of unisolatable, uncontained loss of coolant accidents retain their integrity throughout limiting ATWS transients.

(3) Commercial light-water-cooled power reactor licensees not covered in paragraphs (c)(1) or (c)(2) of this section shall submit an analysis of the ATWS tolerance of their plants.

(4) Each commercial power reactor licensee shall prepare, submit for review and approval, and implement Limiting Conditions of Operation that proscribe operation in, and mandate expeditious retreat from, operation under conditions that compromise the ATWS tolerance of the plant. Limiting Conditions of Operation should also minimize operation under conditions in which the ATWS tolerance of the plant would be severely tested by a limiting ATWS event. Consideration of the prevailing plant parameters as well as equipment operability is appropriate in the Limiting Conditions of Operation.

(5) For the purposes of this pargraph, the ATWS tolerance of a plant is inadequate if any of the more limiting transients, followed by a total failure of the scram system, result in any one of the following:

(i) Containment pressure or temperature above the design values,

(ii) Loss of coolable geometry in the core, or

(iii) Releases of radioactive material that may realistically cause any offsite prompt fatalities or serious offsite property damage.

(6) Applicants or licensees are not required to calculate the potential offsite radiological doses resulting from an ATWS event under § 100.11 of this chapter.

(d) *Schedule of implementation and reporting requirements.* (1) Plans for the implementation of the reliability assurance program called for in paragraphs (a) and (b) of this section must be filed with the NRC for review and approval. Holders of operating licenses, applicants for operating licenses, and those expecting to file an application for an operating license within one year of [the effective date of the rule] shall file the reliability assurance program plan at a time to be agreed upon by the NRC staff. The time afforded for plan development will be not less than one year [from the effective date of the rule]. Those holders of construction permits who file an application for an operating license on or after [one year from the effective date of the rule] shall file the reliability assurance program plan at the time of operating license application. The plans must identify (i) the ways the reliability assurance program will be integrated into the engineering and operations management of the plant, (ii) the reporting and approval requirements internal to the licensee's organization, (iii) the plans for information evaluation and exchange among licensees as part of the experience feedback function, (iv) the criteria for reporting to the NRC, (v) the criteria for the adoption and scheduling of alterations to plant design or operation emerging from the reliability assurance program, and (vi) the date at which the initial reliability assurance studies can be completed. A brief summary of findings and plans for the resolution of reliability deficiencies must be filed with the NRC upon completion of the initial reliability assurance studies. Subsequent discoveries of reliability deficiencies in the plant must be reported in accord with prevailing practices for reporting licensee events. The reliability assurance program will be subject to audit by the NRC. It is not expected that the NRC will engage in routine review and approval of the program unless a pattern suggestive of noncompliance is observed.

(2) Applicants for or holders of operating licenses subject to paragraph (c)(1) or (c)(2) of this section shall file with the NRC plans for the implementation of the requirements of paragraph (c) of this section [within one year of the effective date of the rule] or upon license application, whichever is later. The full implementation of the requirements of paragraphs (c)(1), (c)(2), and (c)(4) of this section must be completed:

(i) For all light-water cooled nuclear reactor power plants for which operating licenses have been or may be issued after August 22, 1969, but before [three years after effective date of the rule], all modifications shall be completed prior to startup following the first refueling that begins (three years after effective date of the rule).

(ii) For all light-water cooled nuclear reactor power plants licensed on or after (three years after effective date of the rule), all modifications shall be completed prior to issuance of an operating license.

(3) Holders of operating licenses subject to paragraph (c)(3) of this section shall file with the NRC plans for the accomplishment of the ATWS tolerance assessment called for in paragraph (c) of this section [within one year of the effective date of the rule]. Such licensees shall file the results of these studies, together with proposed changes, if any, in design, procedures, and technical specifications to assure ATWS-tolerance, and a proposed implementation schedule shall be filed with the NRC for review and approval [within three years of the effective date of the rule.]

Dated at Washington, D.C., this 19th day of November, 1981.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 81-33942 Filed 11-23-81; 8:45 am]

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FEDERAL RESERVE SYSTEM

12 CFR Parts 207, 220, and 221

[Docket No. R-0372]

Proposal To Revise Criteria for Initial and Continued Inclusion on the List of OTC Margin Stocks

AGENCY: Board of Governors of the Federal Reserve System.

ACTION: Proposed amendments.

SUMMARY: The Board proposes to amend the requirements set forth in Regulations G, T and U for inclusion and continued inclusion on the List of OTC Margin Stocks ("OTC List"). Brokers and dealers may not extend credit on stocks which are traded over-the-counter unless such stocks appear on the OTC List. Loans by banks and other lenders that are used to purchase stocks that appear on the OTC List are subject to the Board's margin requirements.

The proposed amendments would modify three areas in the existing rules for initial and continued OTC List eligibility. First, they would permit equity securities of foreign issuers and American Depository Receipts ("ADRs") to be considered for OTC List inclusion. Second, the proposals would replace certain criteria which must currently be met in the alternative and replace them with mandatory requirements. Finally, existing financial criteria would be relaxed to more closely resemble requirements established by major exchanges.

DATE: Comments should be received by January 29, 1982.

ADDRESS: Comments, which should refer to Docket No. R-0372, may be mailed to the Secretary, Board of Governors of the Federal Reserve System, 20th Street and Constitution Avenue, N.W., Washington, D.C. 20551, or delivered to Room B-2223 between 8:45 a.m. and 5:15 p.m. Comments received may be inspected at Room B-1122 between 8:45 a.m. and 5:15 p.m., except as provided in § 261.6(a) of the Board's Rules Regarding Availability of Information [12 CFR 261.6(a)].

FOR FURTHER INFORMATION CONTACT: Robert S. Plotkin, Assistant Director, Laura Homer, Securities Credit Officer, or Jamie Lenoci, Financial Analyst, Division of Banking Supervision and Regulation (202-452-2781).

SUPPLEMENTARY INFORMATION: In July 1969, the Board adopted criteria for including stocks on the OTC List. In discussions leading to the selection of such criteria, the Board indicated generally that (a) stocks to be included on the List should have market characteristics similar to exchange-listed securities, (b) manipulation by issuers to be included or excluded from the OTC List should be made as difficult as possible, and (c) fluctuations in the number of stocks on the List should be minimized.

The changes now proposed in the OTC List criteria are the result of a review of the OTC margin stock listing and continued listing requirements in light of recent developments in the securities markets in general, the OTC Market in particular, and staff experience with administering the requirements. It is believed that revising the criteria is especially appropriate at this time because of a recent decision to revise the List three times a year commencing in 1982, rather than twice a year as is the current practice. This has been a frequent recommendation of the securities industry. The following is a discussion of the specific proposals to amend OTC List criteria.

A. Deleting Requirement That Issuer be Organized Under the Laws of the United States or a State

As early as 1964, when the SEC first recommended a broadening of the Federal Reserve's margin authority to encompass over-the-counter stocks, the Board indicated that securities, to be eligible for credit at a broker, should meet the prerequisites of (1) market depth, (2) a reliable system of quotations, and (3) sufficient issuer disclosures. The National Association of Securities Dealers Automated Quotation System ("NASDAQ"), now in operation for ten years, has greatly improved the efficiency of the OTC market and has addressed the first two concerns of the Board. The SEC, over the past few years, has improved and strengthened its disclosure rules, so that financial information on foreign as well as domestic issues is available to the public in a comprehensive and timely fashion. In addition, the National Association of Securities Dealers ("NASD") requires that its domestic and foreign issuers file financial data with it as a prerequisite for trading on NASDAQ.

None of the approximately one-hundred eighty (180) foreign stocks currently in the NASDAQ system can be placed on the OTC List, as they do not meet the existing criterion, which requires all OTC List candidates to be "organized under the laws of the United States or a State." A growing number of requests have been received from both investor groups and the general public to include foreign OTC stocks on the OTC List. When the Board first adopted its criteria for inclusion on the List, there was insufficient financial disclosure for foreign issues. This problem has now been remedied. Furthermore, foreign issues can and do list on national exchanges and are therefore automatically eligible for margin credit.

In this connection, the Board also proposes to allow American Depository Receipts ("ADRs") to be eligible for inclusion on the OTC List. ADRs are receipts issued against securities of foreign issuers deposited in an American depository, and are exempt from registration under Section 12 of the '34 Act. There are approximately sixty (60) ADRs currently in NASDAQ. The Board would allow ADRs to be considered for inclusion on the OTC List, provided the foreign securities against which the ADRs are issued are registered pursuant to Section 12 of the '34 Act, which imposes certain reporting requirements upon the foreign issuer. This approach is consistent with the policy currently employed by stock exchanges with respect to exchange listings[1] and with the Securities and Exchange Commission's current proposal to allow ADRs to be designated as national market system securities.[2]

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[1] An exchange will list ADRs only if the underlying foreign security also is listed and therefore registered under Section 12.

[2] SEC Release No. 34-18131.

DOCKETED USNRC OCT 14 1981 Office of the Secretary Docketing & Service Branch

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

Before Administrative Judges:
James P. Gleason, Chairman
Paul W. Purdom
Glenn O. Bright

SERVED OCT 14 1981

In the Matter of: PENNSYLVANIA POWER & LIGHT CO. and ALLEGHENY ELECTRIC COOPERATIVE, INC. (Susquehanna Steam Electric Station, Units 1 and 2)	Docket Nos. 50-387 OL 50-388 OL October 12, 1981

MEMORANDUM AND ORDER ON SUMMARY DISPOSITION MOTIONS

The Board has previously set forth the law applicable to motions for summary disposition (see Board Memorandum and Order, dated March 16, 1981). We see no need to repeat it herein, and thus will proceed with our evaluation of some motions pending. We have previously communicated our decision on some of the motions discussed herein.

1. The applicant filed a motion for summary disposition of contention 5, and subsequently received a Staff response supporting the Applicants' Motion. The Board has received no response from

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any of the other parties in this proceeding.

Background: Contention 5, as accepted by the Board reads as follows:

> 5. Certain models used by the applicants to calculate individual and population radiation doses are inaccurate and obsolete. The deficiencies are compounded by the arbitrary selection of data from inappropriate sources for the purpose of formulating these models.
>
> Specifically:
>
> a. the milk transfer coefficient for iodine has been underestimated (see Health Physics, 35, pp 413-16, 1978);
> b. the models use factors which correct alpha particle dose in rads to rems which are far too low (see Health Physics, 34, pp. 353-60, 1978);
> c. the models use factors which underestimate the radiation effect, on a per rad basis, for the very low energy beta and gamma radiations, as from H-3 and C-14 (see Health Physics, 34, pp. 433-38, 1978).

There being no models prescribed by regulation for such calculations, the Board accepted the contention as a challenge to the models specified. We now consider these.

The general thrust of Intervenor's contention is that the models used by Applicant are inaccurate and obsolete. Both the Applicant and the Staff dispute this, stating that the models used by the Applicant are set forth in NRC Regulatory Guide 1.109. These models were developed by the Staff and by experts at Battelle Pacific Northwest Laboratories and such national laboratories as

Oak Ridge and Argonne. These models, which are used to calculate both the maximum hypothetical individual and general population doses from exposure to radioactive liquid and airborne releases from routine operation of a commercial nuclear power reactor, are subject to continuing peer review and verification by other federal agencies such as the U.S. Environmental Protection Agency and the Bureau of Radiological Health. (Staff affidavit, Branagan at 4). In September 1977, a group of experts, meeting to evaluate models used for the environmental assessment of radionuclide releases, concluded that the transport models given in Regulatory Guide 1.109 are adequate for demonstrating compliance with Appendix 1 of 10 CFR Part 50. (Branagan at 4). As these models are adequate for demonstrating compliance with the numerical guides in Appendix 1, they are conservatively adequate for demonstrating compliance with the higher standards for protection of the public against radiation hazards found in 10 CFR 20. The NRC models are therefore not inaccurate or obsolete.

Data used in these models are taken from reports by scientists at national laboratories, peer-reviewed articles in scientific journals, and recommendations of nationally- and internationally-known radiation protection organizations. (Branagan at 4). The models do not, therefore, use data arbitrarily selected from inappropriate sources.

The Staff also argues, and the Board must agree, that from the standpoint of the protection of public health and safety it is the degree of conservatism in an overall model that is important in determining whether doses have been overestimated or underestimated, rather than the degree of conservatism in individual parameters. The challenged models contain many parameters, some of which depend, in turn, on a number of subparameters. Thus, due to the complexity of the models, an apparent lack of conservatism in one parameter does not necessarily constitute a *prima facie* showing that a model will underestimate doses. (Branagan at 2-3).

Subpart a of the contention alleges that the value of the "milk transfer coefficient" for iodine has been underestimated, and refers to a note contained in *Health Physics*. The Staff points out that the values in Regulatory Guide 1.109 are within the range of values given in the referenced note, both for cow's milk and for goat's milk. (Also see Applicant's Bronson Affidavit at 3). The author of the article referred to, Dr. F. Owen Hoffman, subsequently published a more comprehensive report (NUREG/CR-1004) which included a statistical analysis of the entire NRC model, as well as its individual parameters. The conclusion of this report is that doses to real individuals from ingestion of milk containing radio-iodine are more likely to be overestimated than underestimated. (Branagan at 5-6).

Subpart b alleges that the factors which are used to convert alpha radiation doses from rads to rems give results which are too low. Intervenor cites an article by Rossi, et al., entitled, "Leukemia Risk from Neutrons" (Health Physics, 34, pp. 353-60(1978)), presumably as support for their argument. The primary thrust of the article is the effect of neutrons, and deals with alpha radiation and other types of high linear energy transfer radiation in only a speculative way. (Branagan at 7). Furthermore, alpha particles are not a significant source of dose to offsite individuals from exposure to radioactive effluents from routine reactor operations. (Branagan at 8).

Subsection c alleges that factors used to convert dose to radiation effect underestimate the effect for very low energy beta and gamma radiation, as from H-3 and C-14, and cites an article by Bond, et al., in support of their argument. (Health Physics, 34, pp. 433-38, 1978)).

This "quality factor" ("Q") allows doses of different biological effectiveness to be added and measured relative to a reference standard. In 1969, the International Commission on Radiation Protection (ICRP) established through research and study that a Q-value of 1 was the best estimate for low linear energy transfer beta and gamma matters. Since that time, all major national and international advisory and regulatory groups have used a Q of 1 for low-LET beta and gamma radiation. The NRC has established, by regulation in 10 CFR 20.4 (c)(2), a Q of 1 for low-LET beta and gamma radiation. (Bronson affidavit, paragraph 3, 5, 6).

The cited article asserts that different kinds of radiation in the low-LET range (0.2 to 3.5 kiloelectron-volts per micrometer) may have a biological effect varying by as much as a factor of 4, and proposes that a reference radiation be chosen in the mid-point of this range. If this were done, some kinds of low-LET radiation would have their Q go up or down by a factor of approximately 2. (Bronson affidavit, paragraph 4).

The Susquehanna dose estimates for low-LET beta and gamma radiation were arrived at using a model developed for the NRC by Battelle Pacific Northwest Laboratories. This particular model uses a Q of 1.7. Thus, the instant dose estimates use essentially the same Q as is proposed in the referenced article. Thus, even if the redefinition of Q were made as proposed in the article, this would not require a change in the dose estimates for low-LET beta and gamma emitters released from Susquehanna. (Bronson affidavit, paragraph 7.)

Conclusion

The Board has closely reviewed the affidavits tendered by the Applicant and Staff in support of Applicants' motion for summary disposition of Contention 5 in this proceeding. None of the facts, as discussed above, were controverted by any of the parties in the case. The Board finds that the Applicant and Staff have met the burden of showing the absence of a genuine issue of material fact, and are entitled to judgment as a matter of law. The Applicant's motion is granted.

2. The Applicant filed a motion for summary disposition of contention 8 which is supported by the Staff. No response was received from any other party.

Background: The contention reads as follows:

> The Applicant has not demonstrated adequately a compliance with the requirements of the Standard Review Plan, 5.3.3, "Reactor Vessel Integrity", Part 11.6. As a result the reactor vessels may not survive the thermal shock of cool emergency water after blowdown without cracking.

The Applicant's motion makes the following argument:

a) The recommended criterion of the Standard Review Plan is that the vessels remain leaktight enough to support adequate core cooling after thermal shock; b) that the critical location for cracking to occur in the plants pressure vessels was determined by detailed stress analysis on materials with essentially identical wall thicknesses and basic dimensions; c) that using the ASME Code and other relevant test data and results, the analysis performed by the reactors manufacturer demonstrates that the vessels not only meet the criterion at its most critical point but indicate that the area would survive with considerable margin the stress that could cause cracking to occur.

The Staff's supporting response brings out the information that the design, material, fabrication, inspection and operation of the Applicant's pressure vessels conform to the Standard Review Plan as well as Appendices G & H of 10 CFR Part 50 respecting fracture toughness requirements. In addition, the supporting information argues that independent research efforts support the conclusion that the integrity of the vessel could survive a thermal shock from cooling water during a large break loss-of-coolant accident.

Conclusion:

Based on the foregoing information, which is supported by affidavits, the Board concludes the Applicant has fulfilled the criterion of the Standard Review Plan as well as the other regulatory requirements in relation to the plant's pressure vessel integrity. In view of the information supplied by the Applicant and Staff, and in the absence of any contrary or contradictory information from any party, the Board finds that contention 8 presents no genuine issue of a material fact and therefore grants the Applicant's motion for summary disposition.

3. The applicant has filed motions for summary disposition of all of contention 7 and these motions have been supported by a Staff response. No other party has responded.

Background: Contention 7(a), as accepted by the Board reads as follows:

> 7. The Nuclear Steam Supply System of Susquehanna 1 and 2 containsnumerous generic design deficiencies, some of which may never be resolvable, and which, when reviewed together, render a picture of an unsafe nuclear installation which may never be safe enough to operate. Specifically:
>
> (a) The presure suppression containment structure may not be constructed with sufficient strength to withstand the dynamic forces realized during blowdown.

The Applicant has provided a brief description of the containment structure and its function. Basically, this structure consists of two parts: the wetwell, which contains the water used to condense steam resulting from a blowdown of the primary system; and a drywell, which is situated directly above the wetwell and contains the reactor vessel and associated piping, valves and equipment. The two chambers are separated by a horizontal diaphragm slab. (Affidavit of George R. Abrahamson in Support of Summary Disposition of Contention 7 (a) para. 4-5).

There are two mechanisms by which hydrodynamic loads can be placed upon the containment structure. These are 1) actuation of one or more of the 16 steam relief valves (SRV), and 2) a loss-of-coolant (LOCA). Although the end effect is the same, i.e., the quenching of steam from the primary coolant system, the mechanisms which govern how the hydrodynamic loads are applied differ, and they must be considered separately. We consider these below.

SRV Discharge Loads

The Applicant has described, succinctly but adequately, the complex phenomena which occur during blowdown through SRVs. (Id., par. 6, 20-22). To determine the pressures on the pool boundary during this process, an extensive test program was undertaken by PP&L at Kraftwerk Union (KWU), a German firm with extensive experience in nuclear reactor steam discharge phenomena. Tests were made using an actual SSES SRV system which simulated the simultaneous activation of all sixteen SRVs, which is the case that gives the highest loads on the containment structure. (Id., para. 23).

The test program covered the range of reactor operating conditions, including various parameters such as length of discharge line, pool temperature, steam pressure, etc. (Id., para. 26). Pressure measurements, the principal data obtained from the tests, showed peak values of the order of 15 psig, with a main frequency of about 6 Hz. (Id., para. 29). These pressure histories at the pool boundary were used as input to a computer model of the containment. The resulting calculations show that the SRV blowdown loads on the pool boundary produce stresses in the containment floor and walls that are within the structures' design values. (Id., paras. 31, 48).

Further tests to measure loads caused by SRV blowdown on submerged structures in the containment pool were done for PP&L by SRI International (SRI). Using the peak pressure and oscillation frequency observed in the KWU tests, the SRI tests confirmed that the loads on submerged structures in the pool are well below design loads (Id., paras. 33-34).

LOCA Loads

Applicant describes the physical setup of the downcomer system which discharges steam to the suppression pool following a LOCA. (Id., para. 7). A full description of the phenomena observed during LOCA blowdown, which are similar but not identical to those observed during SRV actuation is also set forth. (Id., paras. 36-38).

Loads on the containment during LOCA blowdown were investigated by SRI, using a setup that was prototypical of SSES. A number of tests were performed covering a range of pool temperatures, steam flows and break sizes, including the break size corresponding to the design basis accident. (i.e. a break in a 28-inch diameter recirculation line). (Id., paras. 43, 44). Measurements were made of the pressures in the drywell, wetwell air space and wetwell water space (Id., para. 41).

The pressure histories obtained for the wetwell airspace and the drywell indicate that the wetwell pressure rises to 25.2 psig and the drywell pressure rises to 37.7 psig. The recirculation line break produces the most rapid flow of steam into the drywell, and the drywell and wetwell pressures for the recirculation line break are greater than for smaller breaks. (Id., para. 47).

Comparison of Hydrodynamic Loads with Containment Design Capacity and Test Level

The design pressure for the SSES containment is 53 psig for both the wetwell and drywell. The SSES containment has already been tested by pressurizing it to 61 psig with air. These pressures are greatly in excess of the maximum pressure of 37.7 psig produced in a recirculation line break. (Id., para. 49), and the 15 psig produced during the SRV discharge. (Id., para. 29). The differential pressure across the containment diaphragm slab produces load stresses that are within the allowable range (Id., para. 39).

The experimental test results and the accompanying computer calculations show that the SSES containment can withstand the hydrodynamic loads from both SRV discharges and LOCAs with ample safety margin. (Id., paras. 48, 50). Therefore, the SSES containment can withstand the dynamic forces realized during blowdown with an ample margin of safety. (Id., para. 2).

The Staff supported the Applicants' motion by its conclusions that dynamic loads used by the Applicant to assess the design capacity to withstand such loads were conservative when reviewed against the Commission's generic acceptance criteria, and further that the Staff had concluded that the dynamic forces realized during blowdown had been considered and that the containment structures had sufficient strength to withstand those forces.

Conclusion:

The Board has reviewed the affidavit supporting Applicant's Motion for summary disposition of Contention 7(a) in this proceeding. None of the facts have been controverted by any of the parties in the case. The Board finds that the Applicant has met the burden of showing the absence of a genuine issue of a material fact, and is entitled to judgment in its favor as a matter of law. Contention 7(a) is dismissed.

Background: Contention 7(b) states that:

> (b) the cracking of stainless steel piping in BWR coolant water environments due to stress corrosion has yet to be prevented or avoided.

In support of their motion for summary disposition of this part of the contention, applicant submitted affidavits from Joseph C. Lemaire (Lemaire affidavit) and Walter J. Rhoads (Rhoades affidavit). The former affidavit generally describes the problem and various studies, experiments, etc., undertaken to understand the mechanisms of intergranular stress corrosion cracking (IGSCC) and determine means to eliminate or mitigate the problem. The latter affidavit addresses the specific means taken at Susquehanna to cope with the problem.

First, the Board is satisfied that the mechanisms which produce cracks in 304 stainless steel are understood. While the incidence of cracks has been low, the discovery of hairline cracks in BWR's in late 1974 and early 1975 triggered an intensive effort to discover its cause. From this effort, it has been determined that the cause was IGSCC, and that it occurred in the sensitized region of the weld heat affected zone. (HAZ). (Lemaire affidavit, paras 8, 11-13). The investigation also determined that three conditions must be present for cracking. These are: 1) tensile stress in excess of the local yield stress, 2) suitable environmental conditions (i.e., dissolved oxygen, and 3) use of susceptible material. Stress corrosion will not occur if any one of these three conditions is absent or reduced below a critical value. (Lemaire affidavit, paras. 14-28).

Second, methods of eliminating one or more of the required factors for IGSCC have been experimentally verified. (Lemaire affidavit, paras. 29-31). These are: 1) solution heat treatment (eliminates residual stress and sensitization), 2) corrosion-resistant cladding, 3) residual stress improvement, (field application of induction heating to relieve stress), 4) ferrite control in weld metal, 5) use of limited-carbon type 304 stainless steel), and 6) use of ASME code in design which limits design stresses. (Lemaire affidavit, paras. 32-43).

Third, Applicants, after being made aware of this potential problem in 1975, have undertaken an extensive program to effectively eliminate IGSCC in the Susquehanna system. This has been accomplished through a number of means. Oxygen levels in the reactor primary coolant will be controlled by a mechanical deaerator. Extensive use has been made of solution heat treatment and induction heating stress relief. Critical piping and weldments have been made of carbon-limited stainless steel and weld metal. Redesign of some elements of the system to eliminate crevices, stagnant reaches and built-in stress point has been made. (Rhoade affidavit, paras 4-12).

Fourth, austenitic stainless steel has a high ductility behavior, which renders sudden, brittle-type fracture highly unlikely. In other words, for a significant crack, the component would leak before it broke. This has been verified through experience, analysis and experimentaion. (Lemaire affidavit, paras. 9, 10). The Susquehanna plant has a continuous, on-line leak detection system capable of sensing small leaks and small leak changes, such that small leaks can be detected before critical crack length is achieved. (Lemaire affidavit, para. 10, Rhoades affidavit, para. 13).

The Staff supported the applicant's motion on the grounds that its program to reduce and evaluate incidents of intergranular stress corrosion cracking conforms to the in-service inspection and leak detection requirements of NUREG-1313, Revision 1, which was developed subsequent to receiving recommendations from NRC and General Electric study groups on the problem. After a careful review of the information submitted, the Board finds that a substantial case for summary disposition may be present. However, the Board has some reservations which precludes its finding that no genuine issue of a material fact exist in this part of contention 7. The principle issue, we believe that should be ventilated during the hearing without excluding any others covered by this part of the intervener's contention has two aspects: first, the use of low-carbon stainless steel while apparently investigated thoroughly on an experimental basis deserves further information on the record concerning operating experience, if any, in its application; and secondly, there should be some further illumination of the efficacy of the applicant's leak detection system in the areas of concern here. Accordingly, the Board denies this part of applicant's summary disposition of contention 7.

Background: Contention 7 (c) reads as follows:

> c. BWR core spray nozzles occasionally crack, a problem which reduces their effectiveness.

No cracking of core spray nozzles has ever been reported to General Electric Co., nor is GE aware of any such cracking. Cracking of these nozzles would not be expected in view of the relatively low cyclic thermal stress in these nozzles and the successful overall performance of core spray nozzles throughout four hundred reactor years of service. (Affidavit of Joseph C. Lemaire, para. 4).

The Staff supported Applicant's motion and confirmed the information that no cracking of BWR core spray nozzles has ever been reported.

Cracking has occurred in other parts of the core spray system, namely in external lines, safe ends, internal core spray piping and core spray spargers.

This cracking was determined to result from intergranular stress corrosion in the type 304 high-carbon stainless steel which was used in these components. At Susquehanna either low-carbon type 304L stainless steel or Alloy 600, both of which are highly resistant to intergranular stress corrosion, is used (Id., paras. 5-8).

In summary, cracking has not been reported in BWR core spray nozzles, and infrequent cracking in other components of the system occurred in materials substantially different from those used at Susquehanna. The Board finds no genuine issue of material fact to be heard here and Applicants are entitled to a decision in their favor as a matter of law. Appliants' Motion for Summary Disposition of Contention 7(c) is therefore granted.

Background: Contention 7(d) reads as follows:

> (d) The ability of Susquehanna to survive anticipated transients without scram (ATWS) remains to be demonstrated. In this regard, reliance on probabilistic numbers, as 10^{-7} per year, is unwise and unsafe.

Unresolved generic safety issues, of which ATWS is one, are rarely litigated absent a showing of special circumstances involving a specific plant. No such showing has been presented in this case. In resolving this issue, then, the Board must be guided by the Appeal Board's ruling in Gulf State Utilities Co.

(River Bend Station, Units 1 and 2), ALAB-444, 6 NRC 760 (1977), and Virginia Electric and Power Company (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 NRC 245 (1978). These rulings basically hold that the primary consideration is whether the Staff review of an unresolved issue is adequate.

Technical resolution of the ATWS problem has been completed by the Staff, and a proposal for rulemaking has been submitted to the Commission. (SECY-80-409, dated September 4, 1980). In this situation, the Appeal Board has upheld a Licensing Board's ruling that a facility could operate safely, even though ATWS was an unresolved safety issue, pending final Commission action. (Northern States Power Co. (Monticello, Unit 1), ALAB-611, 12 NRC 301, 304 (1980)). A requisite here is compliance with the Staff's interim requirements.

After a thorough study of the ATWS problem in boiling water reactors, the Staff has established the following interim requirements:

1. Applicants must develop emergency operating procedures to recognize and handle an ATWS event; and

2. train operators to take such actions to respond to an ATWS event; and

3. Install an automatic trip of the recirculation pumps.

The reason for the first two requirements are self-evident. Automatic trip of the recirculation pumps has a two-fold beneficial effect. First, it minimizes the pressure rise in the vessel in the first few seconds of the event so that the reactor coolant system pressure is maintained within acceptable limits by the relief valve. Second, it reduces the reactor thermal power output, which in turn minimizes the peak suppression pool temperature and containment pressure.

Conclusion:

Applicant has implemented, or is implementing on a continuing basis, requirements 1 and 2. (Affidavit of William L. Fiock, para. 11). It is in the process of installing an automatic recirculation pump trip. (Id., paras 9, 10). The Board therefore finds that the Susquehanna plant can be operated with no undue risk to the public from an ATWS event. Applicants' motion for summary disposition for contention 7(d) is granted.

4. The Applicant filed a motion for summary disposition of part of contention 11 which is supported by the Staff and has not been responded to by any other party.

Background: The contention alleges that the Applicants have failed to provide adequately for safe on-site storage, for periods of up to 10 to 15 years, of spent fuel and by violating the standards for protection against radiation in 10 CFR 20.1 and 20.105(a), the project creates an unreasonable risk of harm to the health and safety of the petitioners and their property.

The Applicants' motion makes the following points: 1a) With respect to the spent fuel facilities, each unit has its own storage facility, located in the reactor building and consisting of a water-filled reinforced concrete basin lined with stainless steel with racks for storing the fuel, cranes and material handling equipment, a heat exchanger for cooling the water purity and pumps to circulate the water; b) the pool walls are six foot thick reinforced concrete; a leak detection system is provided, the liner's corrosion is insignificant and repairs can be made, if necessary, when fuel is in the pool; c) the fuel racks are designed to withstand any significant degradation; d) the fuel pool cooling system has several backup systems and four independent sources of make-up water for evaporative losses, if they became necessary; e) that alarms indicating high pool water temperatures, high or low water levels and high area radiation are provided in the control room; f) the design of the spent fuel racks will asure the spent fuel remains in a sub-critical condition under both normal and abnormal conditions; g) that the major components of the system are protected against any credible seismic event, and the possibility of the system being impacted by aircraft, spacecraft or meteors is negligible, and h) the spent storage facilities can store spent fuel safely for at least the duration of the operating license period or until the year 2013.

2a) With respect to the capability of the spent fuel to be safely stored during this period, the Commission Advisory Committee on Reactor Safeguards has stated that safe interim storage of spent fuel can be provided well beyond a thirty-year period; b) that spent fuel in storage is best characterized by its inactivity and that decay heat from fission products decreases rapidly so that the margin of safety for the storage system increases with time in storage; c) that any credible failures in the cooling system would only result in slow temperature increases in view of the large volumes of water in the system; d) visual monitoring of the fuel in storage is possible and that monitoring of radiation levels of the pool water and of airborne radioactive materials above the pool is performed frequently; e) that Zircaloy cladding surrounding fuel pellets is an important containment barrier and such fuel has been stored successfully for periods of over twenty years; f) that the uranium oxide ceramic fuel pellets themselves provide a barrier to the leaching of radioactive material into basin water, and g) that encapsulation as a means of isolating defective or failed fuel in storage has been used routinely in Canada and could be used here if necessary.

The Staff's supporting response indicates that the spent fuel facilities of the Applicant including its operating systems and backups meet the recommendations of the appropriate regulatory guides and General Design Criterion. It further substantiates that the design of the fuel was adequate to withstand storage well in excess of the 10 to 15-year period referred to in contention 11 without a loss of integrity and that any corrosion in fuel rods during the lifetime of the plant would be of little significance.

Conclusions:

Based on the information contained in the motions and supporting affidavits and in the absence of any information to the contrary in the pleadings, the Board finds that the Applicant has not violated the Commission's standards for protection against radiation in the facility's on-site storage of spent fuel and that there is no genuine issue of a material fact presented by this contention. Accordingly, the Applicant's motion for summary disposition of this portion of contention 11 is granted.

FOR THE ATOMIC SAFETY AND LICENSING BOARD

James P. Gleason

James P. Gleason, Chairman
ADMINISTRATIVE JUDGE

Dated at Bethesda, Maryland
this 12th day of October, 1981

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of ARIZONA PUBLIC SERVICE COMPANY, et al. (Palo Verde Nuclear Generating Station, Units 1, 2 and 3)	)))))))	Docket Nos. STN 50-528 STN 50-529 STN 50-530

CERTIFICATE OF SERVICE

I hereby certify that copies of "Joint Applicants' Motion for Summary Disposition of Intervenor's Contention No. 6B" have been served upon the following listed persons by deposit in the United States mail, properly addressed and with postage prepaid, this 15th day of January, 1982.

Docketing and Service Section
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Chairman, Maricopa County Board of Supervisors
111 South Third Avenue
Phoenix, Arizona 85004

Dr. Richard F. Cole
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Atomic Safety and Licensing Appeal Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Ms. Patricia Lee Hourihan
6413 S. 26th Street
Phoenix, Arizona 85040

Robert M. Lazo, Esq.
Chairman, Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Dixon Callahan
Union Carbide Corporation
P.O. Box Y
Oak Ridge, Tennessee 37830

Atomic Safety and Licensing Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Stephen M. Sohinki, Esq.
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Edwin J. Reis, Esq.
Office of the Executive Legal Director
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Rand L. Greenfield, Esq.
Assistant Attorney General
P.O. Drawer 1508
Santa Fe, New Mexico 87504

Charles A Bischoff

Charles A. Bischoff