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

NOV 7 1980

MEMORANDUM FOR: Chairman Ahearne
Commissioner Gilinsky
Commissioner Hendrie
Commissioner Bradford

FROM: Robert B. Minogue, Director
Office of Standards Development
(Signed) T. A. Rehm

THRU: Executive Director for Operations

SUBJECT: PROPOSED RULEMAKING TO AMEND 10 CFR PART 50 CONCERNING ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) EVENTS (SECY 80-409)

As requested in the September 30, 1980 memorandum from Chairman Ahearne, the Federal Register Notice has been rewritten. Enclosure "A" to SECY-80-409 should be replaced with the revised Enclosure "A."

The discussion in the Notice has been rewritten to include:

1. a History extracted from Enclosure "C";

2. a revised Basis for the Proposed Rule;

3. the Content of the Rule derived from Enclosure "D";

4. an Implementation of Requirements.

The proposed rule has been rewritten to:

1. clarify the implementation and effective dates including the application to standardized plants;

2. substitute a limit on the radioactivity source term as acceptance criteria for radiological consequences rather than the dose guideline values of Part 100. This is consistent with the proposed new siting criteria:

Contact:
W. Minners, NRR
49-27581

8011210208

3. clarify that only likely failures, not all single failures, are to be considered in the evaluation of post-84 plants;

4. make the criteria applicable to all plants except where unique to one design; and,

5. add criteria for mitigating systems.

A summary of the requirements of the proposed rule and the proposed implementation schedule is also provided (Enclosure "K").

The Office of the Executive Legal Director has no legal objection.

Robert B. Minogue, Director
Office of Standards Development

Enclosures:
1. "A" - Notice of Proposed Rulemaking
2. "K" - Summary of Requirements and Implementation

cc: OPE
OGC
SECY

NUCLEAR REGULATORY COMMISSION

10 CFR PART 50

DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Acceptance Criteria for Protection Against Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants

AGENCY: U. S. Nuclear Regulatory Commission

ACTION: Proposed Rule

SUMMARY: Following extensive study of anticipated transients without scram (ATWS) events, the NRC has concluded that the probability of an ATWS event occurring, as well as the attendant consequences of such an event should one occur, are unacceptably high. Accordingly, the NRC is considering amending its regulations to require improvements in the design, construction, and operation of nuclear power plants in order to (1) reduce the probability that a nuclear power plant would fail to scram (rapidly shut down) following the occurrence of a transient event (i.e., an abnormal operating condition), and (2) mitigate the consequences of a failure to scram following a transient, should such an event occur.

DATES: The comment period expires (90 days after publication)

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 will be available for public inspection in the Commission's Public Document Room at 1717 H. Street, N.W., Washington, D. C.

FOR FURTHER INFORMATION CONTACT:

Medhat M. El-Zeftawy, Office of Standards Development, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, (301) 443-5921

SUPPLEMENTARY INFORMATION: History: The questions of whether and to what extent anticipated transients without scram (ATWS) events should be considered in the design and safety evaluation of nuclear power plants has been the subject of extensive and continuing studies by the NRC staff and the regulated industry. It has long been recognized that, should an ATWS event occur, the potential for release of a substantial amount of radioactive fission products as a result of the melting of reactor fuel is significant. It was not until 1969, however, that it became apparent that the probability of such an event occurring may in fact be higher than earlier analyses indicated. Early that year a consultant to the Advisory Committee on Reactor Safeguards (ACRS) pointed out that the reliability of the reactor protection system may be diminished as a result of a common mode failure (i.e., the failure of multiple components due to a common single cause).

The concern was that an extraordinarily high reliability of the reactor trip and reactivity shutdown systems was required, considering the relatively high rate of challenge by anticipated transients, the increasing number of nuclear power plants, and the desire to assure that the potentially severe consequences of failure were very unlikely. Attaining such a high reliability requires that the frequency of occurrence of failures of common mode failures be extremely low. This extremely low frequency was lower than could be confidently predicted by currently available reliability assessment methods. Thus, the practicality of attaining and demonstrating the required high reliability in a single reactor protection system subject to common mode failures was questionable. Because of this question, analyses of the consequences of postulated ATWS events were requested from reactor designers. After reviewing these preliminary and simplified analyses, the staff confirmed that the consequences of ATWS events could be severe in that several anticipated transients would require prompt action to shut down the reactor in order to avoid high pressure in the primary system and possible offsite effects. The staff's preliminary results on ATWS were discussed with the ACRS in September 1970.

In August 1971, the ACRS and the regulatory staff concluded that a design change to the proposed Newbold Island boiling water reactor units was appropriate to limit the possible consequences of ATWS. In April 1972, the staff transmitted to the ACRS a proposed set of positions and actions to be taken to implement the conclusions of the staff and ACRS studies on ATWS. In January 1973, as a result of further review and discussion, the staff transmitted to the ACRS an amended proposed position on the need for protection against ATWS. The ACRS responded in April 1973, agreeing with the amended position. In September 1973, the staff published the "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors" (WASH-1270) containing its position on ATWS.

After WASH-1270 was issued, reactor manufacturers in conjunction with the staff, began to develop acceptable methods of performing analyses of ATWS events. A draft industry standard, ANSI-N661, was written, which outlined general guidelines for the analysis of ATWS events in PWRs. In October 1974, each manufacturer of nuclear steam supply systems (NSSS vendors) submitted reports describing the analysis of ATWS events for their reactor designs. The staff reviewed these reports and in December 1975 issued status reports with evaluations of these analyses. The NSSS vendor transient analysis methods were generally acceptable except for the treatment of system failures and some system parameters. Subsequently, in mid-1976, applicants were requested to perform analyses for their plants using the methods developed by the NSSS vendors, modified as indicated in the staff status reports. In 1975, contemporary with the status reports, the staff published the Reactor Safety Study (RSS), WASH-1400*, which contained estimates of the probability and consequences of core-melt from various causes including ATWS events.

In 1976, the Electric Power Research Institute (EPRI) published a set of reports which had been submitted earlier and provided their assessment of the significance of ATWS events and supporting analysis and data. As a result of criticism of the staff position and the new information submitted by EPRI and published in the RSS, the staff reviewed and evaluated the information then available on the subject with particular emphasis on the material developed subsequent to the publication of the status reports.

The results of their review were published by the staff in April 1978 in the report "Anticipated Transients Without Scram for Light Water Reactors" NUREG-0460, Volumes 1 and 2*. In this report, the staff concluded that some

*Copies of these reports may be purchased from the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission Washington, D.C. 20555.

corrective measures to reduce the probability or consequences of ATWS were required because the reliability of reactor protection systems, and therefore the frequency of ATWS events resulting in core-melt, could not be shown to be a small contribution to the overall risk from nuclear power plants as estimated in the Reactor Safety Study. These reports were discussed with the ACRS in a series of meetings through the Fall of 1978. Late in 1978 the Risk Assessment Review Group reported to the NRC on its assessment of the RSS and the current state of risk assessment methodology. This group recommended that, in general, the use of probabilistic risk analysis methodology be avoided for the determination of absolute risk probabilities for subsystems unless an adequate data base existed and it were possible to quantify the uncertainties. The Nuclear Regulatory Commission accepted this recommendation and included it in a statement of policy issued in January 1979.

Based on the recommendations of the Risk Assessment Review Group; a review by the staff Regulatory Requirements Review Committee; and the information provided by nuclear utilities, architect-engineering, and reactor manufacturers at the 1978 ACRS meetings; the staff reexamined the approach recommended in NUREG-0460. The results of this reexamination were published in Volume 3 of NUREG-0460 in December 1978. In this report the staff concluded that the safety objective presented in the report was not satisfactory for use in regulatory decision making, but that engineering evaluation and judgment, supported by quantitative risk evaluation, should be used to determine the appropriateness of the various alternative plant modifications described. The staff recommended specific modifications to plants that were considered appropriate and in February 1979 requested that generic evaluations be performed by the industry to confir that these modifications would achieve the desired objectives.

The accident at Three Mile Island forced deferral of all NRC and most industry work on ATWS and the information submitted fell far short of the request. The information that was received did not confirm that the proposed modifications would achieve the objectives. In Volume 4 of NUREG-0460, published in March 1980, the staff revised its recommendations and proposed to require some specific modifications to plants immediately and to impose further requirements after conducting a rulemaking proceeding and issuing a regulation. Subsequently the staff decided to recommend that all ATWS requirements be imposed by regulation after a rulemaking proceeding.

Basis for Proposed Rule: The statutory basis for deciding whether, and to what extent, ATWS events should be considered in the design and safety evaluation of nuclear power plants is set forth in Section 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." Implementation of this direction in the regulations for nuclear power plants has been based on dual objectives, prevention of accidents and mitigation of their consequences should they occur. Thus, conservative design, construction, and operation of plants are required so that the accidents will be prevented (i.e., have a low probability of occurrence). Then, to provide defense in depth, the capability to mitigate the consequences of accidents that are postulated to occur is required even though the design includes measures to prevent them.

In specifying the requirements for preventing or mitigating accidents, not all accidents that can possibly occur are postulated to occur. One function of the regulations and requirements of the NRC is to specify which possible accidents and consequences are sufficiently probable to warrant preventive or mitigative measures. It is within this framework that the NRC has concluded that the probability of ATWS events ocurring over the lifetime of light-water-nuclear power plants and the potential magnitude of consequences arising from such events, should they occur, are sufficiently great to warrant the imposition of requirements designed to reduce the probability and mitigate the consequences of ATWS events.

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 that form the basis for this conclusion 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 mode failures. Common mode 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 mode failures. However, one common mode failure in the reactor trip portion of the protection system of a commercial nuclear power reactor has occurred during approximately 700 reactor-years of operating experience. The failure was detected during normal.surveillance and corrected before any event requiring a reactor scram occurred. Common mode failures have also occurred in other systems in nuclear power plants and other potential common mode failures in reactor protection systems have been identified. Because of the low rate of occurrence of common mode 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. Reliability analysis methods are also inadequate because they must treat common mode failures either in an arbitrary manner or use the highly uncertain estimates of common mode failure rates derived from operating experience. 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 staff has concluded that the reliability is inadequate.

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 value of system process variables at the time the event occurs. The capability of a plant to mitigate ATWS events can be assessed by analyses. 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 analyses, all combine to produce uncertainty in the results. Therefore, the difficulty in demonstrating a capability to adequately mitigate ATWS events is similar to the difficulty in demonstrating that ATWS events can be prevented. However, based on analyses performed to date, 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.

Content of the Proposed Rule: Having concluded that improvements to reduce the probability of severe consequences from ATWS events should be made, the staff developed four alternative sets of requirements that would provide increasing reduction in this probability 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 also 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 are also different from those 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 staff has concluded that the reliability of current reactor protection systems is insufficient 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 was di -se 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.

The staff proposed in Volume 4 to implement only Alternative 2 for the ten older plants which 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 and increasing the flow capacity of the Standby Liquid Control System. Considering the state of design and construction, and a balancing of pubic safety benefits against economic cost, the staff concludes 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 virtually 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 require significant design changes. The mitigation capability of PWRs 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 staff concluded 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 explicitly 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 design of their plant is 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 optimizing 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 assuring the integrity of the reactor coolant system and the continued cooling of the 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. If only criteria 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 were 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, whether the mitigation of most or virtually all ATWS events, is specified through the criteria for acceptable evaluation models. Since the parameters in the evaluation model are all uncertain to some degree and some 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 assure that that 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 is the moderator temperature coefficient) must be 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 assure 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. Such 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 assure 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 in or after 1984, the time available to design and install the modifications to the protection system is sufficient to assure 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. BWR licensees have also been directed (IE Bulletin No. 80-17 dated July 3, 1980) to assure 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 which will reduce the risk from ATWS events during the interim period before the plant modifications determined by the Commission to be necessary and included in a final rule, 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. Identification of these sites is a subject of another Commission action and any additional ATWS requirements for these units would be subsequently considered.

Concurrent with this publication of the proposed rule for comment, the staff is also publishing a proposed regulatory guide for comment. This regulatory guide provides guidance on the evaluation models, mitigating system design requirements, and other licensing requirements.

Implementation of Requirements: The proposed rule provides 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 by July 1, 1982. In order to accomplish this, descriptions of the modifications are to be submitted for review by the NRC by July 1, 1981. Since these modifications involve instrumentation, control and logic circuits that are for the most part outside of the containment, most of the installation could be accomplished with the plant operating or during refueling outages and a two year design and installation schedule appears appropriate.

The proposed rule provides that implementation of any modifications required to meet ATWS acceptance criteria be completed by January 1, 1984 for pressurized water reactor plants and for the oldest boiling water reactor plants. All other boiling water reactor plants would be required to complete the modifications by July 1, 1982. In order to accomplish this, evaluation models are to be submitted for review by March 1981 and descriptions of the modifications are to be submitted for review by July 1981 (by December 1981 for the ten oldest plants). Since the modifications to the boiling water reactors would consist of piping changes to the Standby Liquid Control System, which is primarily outside containment, a two year design and installation schedule appears to be appropriate. Since any modifications determined to be required for pressurized water reactors would likely require the installation or modification of valves on the pressurizer, the more extended schedule for these plants is appropriate.

The NRC believes that the likelihood of severe consequences arising from an ATWS event during the period of this implementation 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) partial ATWS mitigative capability of the recirculation pump trips feature which has been implemented on most operating BWRs and which is required to be implemented on the remaining BWRs by December 31, 1980, 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 NRC believes that the implementation schedule in the proposed rule is acceptable and will minimize the risk of hasty modifications which may be counterproductive to safety.

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.

A new Section 50.49 is added to read as follows:

§50.49 Acceptance Criteria for Protection Against Anticipated Transient Without Scram Events for Light-Water-Cooled Nuclear Power Plants

(a) As used in this section:

(1) "Anticipated Transient Without Scram" (ATWS) means an anticipated operational occurrence as defined in Appendix A of this part followed by the 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) Each light-water-cooled nuclear power plant shall be designed, constructed, and operated such that the consequences of postulated anticipated transient without scram (ATWS) events calculated in accordance with an ATWS evaluation model, approved pursuant to paragraph (b) (2) 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 shall be limited so that either (A) the calculated maximum primary stress anywhere in the RCS pressure boundary 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 and the calculated deformation of RCS components is limited so that the operability of components necessary to safely bring and maintain the reactor at a cold shutdown condition is not impaired, or (B) the integrity or operability of RCS components shall 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, shall be limited to assure that the core geometry is not significantly distorted such as to impair core cooling or safe shutdown.

(iii) Radiation Release. The calculated release of radioactivity from the fuel rods to the reactor coolant system during postulated ATWS events shall not exceed one percent 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 shall 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 shall be limited so that steam quenching instability will not result in destructive vibrations.

(v) Long-term shutdown and cooling. The reactor design shall 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)(i) ATWS evaluation models shall, 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 shall 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 shall 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 shall 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) ATWS mitigating systems shall be independent, separate and diverse from the reactor protection system. ATWS mitigating systems shall be designed, qualified, monitored and periodically tested to assure continuing functional capability under the conditions accompanying postulated ATWS events including natural phenomena such as earthquakes, storms, tornadoes, and hurricanes, and floods expected to occur during the design life of the plant. ATWS mitigating systems shall be automatically initiated 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, as defined in paragraph (b) (2) of this section, together with the description and results of the analyses and tests necessary to verify the validity of the assumptions made in preparing such evaluation models, shall be submitted to the Nuclear Regulatory Commission for approval by March 1, 1981 or prior to issuance of an operating license, whichever is later.

(5) A description of all measures to be taken to ensure compliance with the criteria set forth in paragraph b(1) of this section together with such proposed changes in technical specifications and license amendments as may be necessary to ensure compliance with such criteria shall be submitted 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, such information shall be submitted no later than December 1, 1981.

(ii) For all light-water-cooled nuclear power plants for which operating licenses have been issued after August 22, 1969, such information shall be submitted no later than July 1, 1981 or prior to issuance of an operating license, whichever is later.

(6) Those measures necessary to ensure compliance with the criteria set forth in paragraph (b)(1) of this section shall be implemented on the following schedule:

(i) For all boiling water reactor power plants for which operating licenses have been issued on or before August 22, 1969, all modifications shall be completed by Janaury 1, 1984.

(ii) For all boiling water reactor power plants for which operating licenses have been or may be issued after August 22, 1969, all modifications shall be completed no later than July 1, 1982 or prior to issuance of an operating license, whichever is later.

(iii) For all pressurized water reactor power plants, all modifications shall be completed no later than January 1, 1984 or prior to issuance of an operating license, whichever is later.

(c) (1) 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, shall be provided with:

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

(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; and

(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) Facilities issued operating licenses before January 1, 1984 or facilities 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 shall be greater than that specified in paragraph (b)(2)(i).

(3) 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), shall be submitted to the Nuclear Regulatory Commission no later than July 1, 1981 or prior to issuance of an operating license, whichever is later.

(4) Those measures required under paragraph (c)(1) of this section shall be completed by July 1, 1982 or prior to issuance of an operating license, whichever is later.

All interested persons who desire to submit written comments or suggestion concerning the proposed rulemaking should send their comments to the Secretary of the Commission, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Docketing and Service Branch, or on before ____________.* Copies of comments received on the proposed amendments may be examined in the Commission's Public Document Room at 1717 H Street, NW, Washington, D.C.

(Sec. 161b and i, Pub. Law 83-703, 68 Stat. 948, Sec. 201, Pub. Law 93-438, 88 Stat. 1242 (42 U.S.C. 2201(b), 5841).)

Dated at ____________ this ___ day of ____________, 1980__.

For the Nuclear Regulatory Commission.

Samuel C. Chilk
Secretary of the Commission

*90 days after publication in the Federal Register

TABLE K-1

SUMMARY OF REQUIREMENTS

(*Indicates implicit requirement.)

VENDOR	ALT. 2	ALT. 3	ALT. 4
B&W, CE	SPS	SPS	SPS
	AMSAC	AMSAC	AMSAC
	Cont Isol	Cont Isol	Cont Isol
	Analysis	Analysis	Analysis
	--	Instr*	Instr*
	--	Safety Valve*	Safety Valve*
	--	--	
W	--	--	SPS
	AMSAC	AMSAC	AMSAC
	Cont Isol	Cont Isol	Cont Isol
	Analysis	Analysis	Analysis
	--	Instr*	Instr*
	--	--	--
GE	SPS	SPS	SPS
	AMSAC	AMSAC	AMSAC
	Cont Isol	Cont Isol	Cont Isol
	Analysis	Analysis	Analysis
	SD	SD	SD
	Logic*	Logic*	Logic*
	--	Instr*	Instr*
	--	SLCS-Auto* Incr. Cap	--
	--	--	SLCS-Auto* Hi-Cap

Enclosure "K-1"

PROPOSED ATWS RULE

SUMMARY OF REQUIREMENTS

KEY

AMSAC:	ATWS mitigation actuation circuitry. Diverse and independent from the reactor protection system to actuate: 1. PWR's - Turbine trip, auxiliary feedwater 2. BWR's - HPLI, SLCS, RPT
Analysis:	Analysis with acceptable evaluation models of performance following ATWS events
Cont Isol:	Containment isolation initiated by early detection of fuel failures
Instr:	Instrumentation necessary for shutdown that can withstand ATWS conditions
Logic:	Logic of control circuits to reduce vessel isolation events and runback feedwater
Safety Valves:	Additional safety valve relief capacity
SPS:	Supplementary protection system that is diverse and independent from the reactor trip portion of the reactor protection system -- B&W - BUSS, a diverse four-channel backup scram system CE - SPS, a diverse, four channel supplementary protection system W - MSS, a modified scram system that is diverse and independent from the RPS GE - ARI, ATWS rod injection that has separate sensors and redundant scram air header exhaust valves
SD:	Scram discharge volume for GE control rods that is less susceptible to common mode failure
SLCS-Auto:	Automatically initiated, Standby Liquid (neutron poison) Control System -- Incr-Cap: Capability to simultaneously inject with both pumps Hi-Cap: Single failure proof system, with approximately 400 gpm capacity

Enclosure: "K-2"

TABLE K-2

PROPOSED ATWS RULE

IMPLEMENTATION.SCHEDULE

YEAR	80	81			82	83
	10/1	3/1	7/1	12/1	7/1	
EVALUATION MODEL		0				
INITIAL CHANGES						
DESIGN			0			
INSTALL					Δ	
DESIGN FINAL CHANGES						
CURRENT PLANTS			0			
OLD PLANTS				0		
INSTALL FINAL CHANGES						
BWR						
CURRENT					Δ	
OLD						Δ
PWR						Δ

Enclosure "K-3"

PROPOSED ATWS RULE

PRESENTATION OUTLINE

COMMISSION BRIEFING

NOVEMBER 10, 1980

1. Description of Brookhaven National Laboratory Calculations for the Browns Ferry-Unit 3 Partial Failure to Scram (T. Speis, NRR, 30 minutes)

2. Discussion of Nuclear Safety Journal article "Anticipated Transients Without Scram" (W. Minners, NRR, 45 minutes)

3. Comparison of Utilities Petition with Proposed Rule (W. Minners, NRR, 15 minutes)