STATEMENT OF MATERIAL FACTS AS TO WHICH THERE EXISTS NO GENUINE ISSUE TO BE HEARD

1. Issue #6 in this proceeding states:

   Applicants should install an automated standby liquid control system to mitigate the consequences of an anticipated transient without scram.

2. New 10 CFR 50.62 (c)(4) requires that "(t)he SLCS initiation must be automatic and must be designed to perform its function in a reliable manner . . . for plants granted a construction permit prior to July 26, 1984 that have already been designed and built to include this feature."

3. Applicants were granted a construction permit on May 3, 1977.

4. The Perry Nuclear Power Plant is being designed and built such that the SLCS will be capable of automatic initiation. See August 13, 1982 letter from D. Davidson to A. Schwencer (Attachment 2).

5. Automation of the SLCS can be achieved at low cost. See Applicants' Supplemental Answer to Sunflower Alliance Interrogatory #22, Second Set, February 29, 1984.

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

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

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Commission is 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) event. The final rule requires the installation of certain equipment in nuclear power plants. It also encourages the development of a reliability assurance program for the reactor trip system on a voluntary basis. This will significantly reduce the risk of nuclear power plant operation.

EFFECTIVE DATE: July 26, 1984.

FOR FURTHER INFORMATION CONTACT: David W. Pyatt, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, (301) 443-7631.

SUPPLEMENTARY INFORMATION: An anticipated transient without scram (ATWS) is an expected operational transient (such as a loss of feedwater, loss of condenser vacuum, or loss of offsite power to the reactor) which is accompanied by a failure of the reactor trip system (RTS), a part of the protection system, to shut down the reactor. The reactor trip system consists of those power sources, sensors, initiation circuits, logic matrices, bypasses, interlocks, racks, panels and control boards, and actuation and actuated devices that are required to initiate reactor shutdown; this includes circuit breakers, the control rods and control rod mechanisms. That portion of the RTS exclusive of the control rods and control rod mechanisms is here referred to as the scram system. ATWS accidents are a cause of concern because under certain postulated conditions they could lead to severe core damage and release of radioactivity to the environment. The ATWS question involves safe shutdown of the reactor during a transient, if there is a failure of the RTS. There have been precursors to an ATWS; the latest was a failure of the automatic portion of the RTS at the Salem 1 nuclear generating station on February 25, 1983. In that incident, manual shutdown was accomplished after 30 seconds, and no core damage or release of radioactivity occurred.

On November 24, 1981, the Commission invited comments on three alternative proposed rules relating to ATWS (46 FR 57521). Each of the three alternative proposed rules had the objective of reduction of risk from ATWS and each featured a different approach to achieve that objective. One alternative (the Staff Rule) emphasized individual reactor evaluation to identify needed improvements. The second alternative (the Hendrie Rule) emphasized reliability assurance and would have also required certain hardware modifications. The third alternative, proposed by the Utility Group on ATWS in petition for rulemaking PRM 50-29, prescribed specific changes that were keyed to the type of reactor and its manufacturer.

Thirty-nine public comments were received at or close to the April 23, 1982 deadline for submission of comments. An additional comment was received on June 24, 1982. Copies of the comments may be examined in the Commission's Public Document Room at 1717 H Street, NW., Washington, D.C. The following organizations and individuals provided comments:

1. F. I. Lewis, Philadelphia, Pennsylvania (private citizen)
2. S. L. Hiatt, Mentor, Ohio (private citizen)
3. Washington Public Power Supply System (WPPSS)
4. Standardized Nuclear Unit Power Plant System (SNUPPS)
5. South Carolina Electric and Gas Company (South Carolina)
6. General Electric Company (GE)
7. Duke Power Company (Duke)
8. Atomic Industrial Forum (AIF)
9. Detroit Edison (DE)
10. Mississippi Power and Light Company (MP&L)
11. Texas Utilities Generating Company (TUGC)
12. Commonwealth Edison Company
13. Combustion Engineering, Incorporated (CE)
14. The Utility Group on ATWS, representing 22 utilities
15. Combustion Engineering Owners Group
16. Houston Lighting and Power (HL&P)
17. Portland General Electric Company (PGEC)
18. GPU Nuclear (GPU)
19. Babcock and Wilcox Company (B&W)
20. Ebasco Services, Incorporated (Ebasco)
21. Public Service Electric and Gas Company (PSE&G)
22. Carolina Power and Light Company (CP&L), first comment
23. Stone and Webster Engineering Corporation (S&W)
24. Florida Power Corporation (FPL)
25. Gulf States Utilities Company (Gulf)
26. Duquesne Light Company
27. Wisconsin Public Service Corporation (WPSC)
28. Pacific Gas and Electric Company (PG&E)
29. Tennessee Valley Authority (TVA)
30. Pennsylvania Power and Light Company (PP&L)
31. Virginia Electric and Power Company (VEPCO)
32. Arkansas Power and Light Company (AP&L)
33. Alabama Power Company (Alabama)
34. Wisconsin Electric Power Company (WEPC)
35. Power Authority of the State of New York (PASNY)
36. Yankee Atomic Electric Company (Yankee)
37. Public Service Company of Indiana (Indiana)
38. Northeast Utilities Service Company (NUSCO)
39. Carolina Power and Light Company (CP&L), second comment
40. American Electric Power Service Corporation (received June 24, 1982)

Following are members of the Utility Group on ATWS, the petitioner in the PRM-50-29.

Arkansas Power and Light Company
Boston Edison Company
Connecticut Yankee Power Company
The Detroit Edison Company
Florida Power Corporation
Gulf States Utilities Company
Maine Yankee Atomic Power Company
Northeast Nuclear Energy Company
Pacific Gas and Electric Company
Public Service Electric and Gas Co.
Washington Public Power Supply System
Baltimore Gas and Electric Company
Commonwealth Edison Company
Consumers Power Company
Duke Power Company
Florida Power and Light Company
Long Island Lighting Company
Nebraska Public Power District
Omaha Public Power District
Pennsylvania Power and Light Company
Vermont Yankee Nuclear Power Corp.

The breakdown by preference among commenters for the three alternative proposed rule approaches is as follows:

Support "Utility Rule" (PRM-50-29)

WPPSS
DE
Commonwealth Edison
The Utility Group on ATWS
HL&P
Ebasco
PSE&G
FPL
Gulf
PP&L
Yankee

Support "Hendrie Rule" (Most support for this option is tentative with many reservations.)

South Carolina
Duquesne

ATTACHMENT 1

CP&L first comment (would also be considered a "No Rule" choice)
WPSC
VEPCO
S&W

Favor No Rule

SNUPPS
CE
Duke
AIF
MP&L
TUGC
CE
CE Owners Group
PGEC
GPU
B&W
PG&E
AP&L
Alabama
WEPC
Indiana
CP&L second comment
NUSCO
American Electric

The Staff Rule option was favored by Ms. S. L. Hiatt who commented that it was the most stringent of the three proposals, but that it would be better to return to the implementation of specific hardware changes than to require evaluation models. Commenters TVA and PASNY stated a preference for "Alternative 2A" of NUREG-0460[1], Vol. 4, which is very similar to the Utility Rule. The comments from Mr. M I. Lewis did not favor any of the alternatives, but he pointed out limitations of both NRC-proposed rules (limitations of modeling) and felt that the Commission was not fully addressing ATWS.

Most of the utility commenters preferred that the Commission promulgate no rule on ATWS. However, many commenters chose either the Utility Rule or the Hendrie Rule as the more favorable of the alternatives presented (including some commenters within the Utility Group). The No Rule category described above includes those who felt that the risks from ATWS are already sufficiently low, plus those who recommended combining the ATWS rulemaking with other Commission activities such as the Severe Accident Program or the development of a Safety Goal.

The comments provided by the Utility Group on ATWS consisted of a three volume technical report which includes a review and evaluation of past NRC and industry studies, a generic but substantial probabilistic risk assessment of the issue for each NSSS vendor, and a value-impact analysis of all three proposed rules. The conclusions are:

1. The Staff and Hendrie Rules fail the value-impact test.

2. Only the Utility Rule is consistent with current NRC policies.

3. The record and notice for the Staff and Hendrie Rules are inadequate.

In order to resolve the ATWS rule issue, it was necessary for the NRC staff to evaluate the Utility Group report. This was done by a technical assistance contract.

A report which provided a critique of the Utility Group comments was prepared by Energy Incorporated through Sandia National Laboratories and may be examined at the Commission's Public Document Room (PDR) at 1717 H Street, Washington, D.C. Also, a summary of 39 public comments, as well as a plan to resolve the ATWS rule, is available in SECY-82-275 at the PDR.

As proposed in SECY-82-275 and the Commission briefing on July 13, 1982, a Task Force and Steering Group of NRC personnel from several offices was formed to consider the following alternatives:

1. Promulgation of no ATWS rule or including ATWS under the Severe Accident Program;

2. Adoption of the proposed or a modified version of the Utility Group Rule (PRM-50-29);

3. Adoption of the Staff Rule or a modification of it; or

4. Adoption of those portions of the Hendrie Rule for which there exists a technical basis.

The Commission has given careful consideration to all the comments and is now publishing a final rule. This final rule uses, in part the same approach that is used in the Utility Group's petition for rulemaking. Prescribed changes, keyed to the reactor's type and manufacturer, are set out in the final rule. The costs and values of these changes and of other considered changes are discussed in a document on file in the Commission's Public Document Room, entitled "Recommendations of the ATWS Task Force."

Summary of Staff, Hendrie, and Utility Rules

The Staff Rule (46 FR 57521) would have resolved ATWS by establishing performance criteria (e.g., there would be analyses to verify that Service Level C of the ASME Boiler and Pressure Vessel Code would not be exceeded, fuel integrity would be maintained, there would be no excessive radioactivity release, the containment would not fail, and long-term shutdown and cooling would be assured). The Hendrie Rule (46 FR 57521), while using much of the same information base as the Staff Rule, proposed to resolve ATWS by establishing a reliability assurance program for systems that prevent or mitigate ATWS accidents and prescribing certain hardware modifications which would allow for: (1) Automatically tripping recirculation pump of a BWR under conditions indicative of an ATWS; (2) automatically actuating the standby liquid control system (SLCS) for BWRs; (3) providing a reliable scram discharge volume for BWRs; (4) providing for the prompt, automatic initiation of the auxiliary feedwater system for conditions indicative of an ATWS; and (5) assuring that the instruments necessary for the diagnosis of and recovery from ATWS accident sequences will not be disabled. Finally, the Utility Rule proposed specific design modifications for each reactor manufacturer. It contained proposals that: (a) all Westinghouse reactors have initiation of the auxiliary feedwater system and turbine trip diverse from the reactor protection system; (b) all Combustion Engineering and Babcock and Wilcox reactors have diverse initiation of auxiliary feedwater and turbine trip (similar to Westinghouse) and a diverse scram system; and (c) existing boiling water reactors manufactured by General Electric have (1) a means to trip the recirculation pumps upon receipt of a signal indicative of an ATWS, (2) a diverse scram system, and (3) a modification of the scram discharge volume. Also, new (three years after the rule becomes effective) General Electric plants would have a standby liquid control system increased to 86 gpm and all reactor licensees would institute training for operators.

Basis for Final Rule as Promulgated by the Commission

The vast majority of the commenters felt that the approach of the Staff Rule was too open-ended in terms of costs to resolve ATWS (e.g., the analyses could be very costly and time consuming). The Hendrie Rule was found difficult to interpret by most commenters. The ATWS Steering Group opted to evaluate generic plants, in a fashion similar to the Utility Group approach, and define the various fixes and estimate the reduction in probability for ATWS sequences as each additional requirement was added. This would then give a value (reduction in risk) that could be compared to the impact (cost in dollars) of each

[1] A free single copy of NUREG-0460, Vol. 4, to the extent of supply, may be requested for public comment by writing to the Publication Services Section, Document Management Branch, Division of Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

incremental requirement. There are large uncertainties in these analyses, and the detailed results of the analyses can be found in the report entitled "Recommendations of the ATWS Task Force" (discussed above). A brief discussion of the final rule's provisions, including value/impact evaluations, is given next:

Diverse and Independent Auxiliary Feedwater Initiation and Turbine Trip for PWRs: § 50.62(c)(1)

This was proposed by the Utility Group on ATWS. It consists of equipment to trip the turbine and initiate auxiliary feedwater independent of the reactor trip system. It has the acronym AMSAC, which stands for *A*uxiliary (or *A*TWS) *M*itigating *S*ystems *A*ctuation *C*ircuitry. It has a highly favorable value/impact for Westinghouse plants[3] and a marginally favorable value/impact for Combustion Engineering and Babcock and Wilcox plants. Since it has the potential for a spurious trip of the reactor which reduces its value/impact, it should be designed to minimize these trips.

Diverse Scram System: 50.62 (c)(2) and (c)(3)

This was proposed by the Utility Group on ATWS for General Electric, Combustion Engineering, and Babcock and Wilcox plants. It has a favorable value/impact from the Staff's analysis. However, the principal reasons for requiring the feature are to assure emphasis on accident prevention and to obtain the resultant decrease in potential common cause failure paths in the trip system. It also has the potential for a spurious trip of the reactor; therefore, it should be designed to minimize spurious trips. For General Electric plants, installation may extend by one or two days the downtime during a refueling outage.

A diverse scram system for Westinghouse plants was not a recommendation of the Utility Group on ATWS and was not a clear requirement of the Staff Rule or the Hendrie Rule, although the Utility Group on ATWS interpreted the Staff Rule to include it. The system does, however, have a marginally favorable value/impact for Westinghouse plants, assures emphasis on accident prevention, and results in a minimization of the potential for common cause failure paths. To assure full opportunity for public comment, the requirement for a diverse scram system for Westinghouse plants will be published separately as a proposed rule.

[3] The installation of a diverse scram system significantly affects the value/impact of AMSAC.

Increased Standby Liquid Control System (SLCS): § 50.62(c)(4)

The SLCS is a system for injecting borated water into the reactor primary coolant system. The neutron absorption by the boron causes shutdown of the reactor. Addition of this system was proposed by the Utility Group on ATWS for new plants (those receiving an operating license three years after the effective date of the final rule). The Commission believes that, with the use of the Emergency Procedure Guidelines proposed by the BWR Owners Group and General Electric that are being implemented at operating BWRs, increasing the SLCS capacity for operating plants may insure an intact containment for isolation transients, although there is uncertainty in containment failure modes. Because of the vulnerability of BWR containments to ATWS sequences, the Commission has determined that this enhanced mitigation feature is warranted. The high pressure portion of the ECCS of BWR/5 and BWR/6 licensees (HPSC) is injected into spray spargers in the core exit plenum. For these plants, the preferred location for the injection of the borated water from the SLCS is the HPCS line just external to the reactor vessel instead of the standpipe at the core inlet plenum. A similar location is preferred for those BWR/4 licensees with HPCI injection into spargers in the core exit plenum. This injection location provides significant improvement in mixing of the borated water, particularly under low vessel water level conditions such as encountered when the EPGs are followed. This injection location is also preferred, since it could prevent local power increases and possible power excursions during the recovery phase of an ATWS when cold unborated ECCS water could be added above the core. Some BWR/5 and BWR/6 licensees already have this injection location and have designed the SLCS accordingly.

Automatic Recirculation Pump Trip for BWRs: § 50.62(c)(5)

Recirculation pump trip (RPT) was proposed as a rule requirement by the Utility Group on ATWS. This safety feature will result in a reduction of reactor power from 100 percent to about 30 percent following a transient (and failure to scram) within a minute or so. This proposed requirement has already been implemented on all operational BWRs in response to a show cause order dated February 21, 1980. The BWR owners generally agree that this is a necessary requirement, and it is being included in the final rule for completeness.

Automatic Initiation of Standby Liquid Control System

One of the alternatives considered by the Task Force was an automatically initiated standby liquid control system with a capacity of greater than 86 gpm (such as 150–200 gpm). This would have resulted in a considerable risk reduction (about a factor of seven) after the ARI is installed for operating plants. Unfortunately, the cost to do this (based on information supplied by the Utility Group on ATWS) is on the order of $24 million per plant and is significantly impacted by the costs of downtime from an inadvertent trip which would inject boron into the reactor water and by the costs of downtime for installation in existing plants. The value/impact does not favor this alternative for existing plants.

New plants (those which will receive construction permits after the effective date of this rule) will be required to have equipment for automatic initiation of the SLCS. Most of those plants already have been designed for this feature. Also, other plants that have been designed and built to include this feature must utilize the feature. The equipment for automatic SLCS actuation should be designed to perform its function in a reliable manner and to provide high reliability against spurious actuation.

Adding Extra Safety Valves or Burnable Poisons

One of the alternatives considered by the Task Force was adding more safety valves to plants manufactured by Combustion Engineering (CE) and Babcock and Wilcox (B&W). This would reduce the peak pressure in the reactor vessel and yield a higher probability of the plant surviving an ATWS with no core damage. The peak overpressure could also be reduced by modifying the core behavior (the fraction of the time the moderator temperature coefficient is unfavorable) by adding burnable poisons. The Utility Group on ATWS estimated that installing larger valve capacity could cost up to $10 million per plant. A large fraction of this cost is the downtime for installation of the valves. While the probability of ATWS can be reduced about a factor of three or more, the value/impact is unfavorable for this alternative for existing plants. These plants all have large dry containments and will be most able to mitigate the radiological consequences from an ATWS. This rule does not cover enhanced pressure relief capacity for new CE and B&W plants. However, the Commission expects that this issue

would be addressed during the NRC's design review of any specific new plant or standard plant application.

Need for all Control Rods to be Inserted for PWRs

By using soluble boron for burnup and xenon control, PWRs normally operate at or near 100 percent power with control rods nearly out (except for some Babcock and Wilcox "rodded" reactors which keep one bank inserted for xenon control). Thus, nearly all rods are available to participate in a scram.

Insertion of only about 20 percent (approximately 10) of the control rods is needed to achieve hot, zero power provided that the inserted rods are suitably uniformly distributed. What is important is the uniform spacing of the rods. In installing a diverse scram system, the licensee can allow for partial scram failures if it is demonstrated that the rod insertion pattern is sufficiently uniformly spaced such that a hot, zero power is achieved.

Considerations Regarding Reliability Assurance

As a result of the failure of the Salem Unit 1 reactor to scram automatically on February 25, 1983, the NRC conducted an investigation of the events (see NUREG–0977, "NRC Fact-finding Task Force Report on the ATWS Events at Salem Nuclear Generating Station, Unit 1, on February 25, 1983" [3]). One of the principal findings was the lack of adequate attention being paid to the reliability of the reactor trip system. The Salem Generic Issues Task Force recommended to the Commission that a reliability assurance program be included in the final ATWS rule (NUREG–1000, Volume 1, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant" [3]). While this rule does not require such a program, the Commission urges the voluntary development of a reliability assurance program for the RTS.

The reliability assurance program should have the following elements:

1. An analysis of the challenges to and failure modes of the RTS system, considering independent failures quantitatively and common cause failures qualitatively. An estimate of the challenge rate and the reliability of the RTS should be a part of the analysis.

2. A numerical performance standard for the RTS challenges and the RTS unavailability to use as an aid in the initial and continuing evaluation of the adequacy of the system.

3. A process of evaluating plant-specific and industry-wide operating experience to provide feedback to assess whether the RTS is performing reliably enough.

4. Procedures within quality assurance programs to ensure that the RTS performs satisfactorily in service from a reliability perspective. The frequency of challenges to the RTS should be as low as practicable.

A pivotal aspect of the ATWS issue is the reliability of the reactor trip system (RTS), including the control rods, and the difficulty associated with assessing the impact of common cause failures on the availability of the system to function when required. All RTS systems are designed for high availability, yet ATWS precursors at Kahl and Browns Ferry 3, and the ATWS event at Salem 1 did occur and were the result of common cause failures of the RTS. The Kahl and Brown Ferry 3, incidents were described in the Federal Register notice containing the proposed rules which was published on November 24, 1981 (46 FR 57521). The Salem 1 incident occurred after the proposed rules were published.

An analysis of the RTS should be performed using existing methodologies for quantitative evaluation of system reliability (e.g., unavailability). A fault tree and qualitative common cause failure analysis should be performed to identify the potential important faults of the RTS. Examples of quantitative analysis for the RTS are: WASH–1400 (the Reactor Safety Study) [4], the Indian Point Probabilistic Safety Study [5], the Zion Probabilistic Safety Study [5], and other probabilistic safety studies performed by Industry at their own initiative or at the request of the Commission. There are an estimated 15–20 probabilistic studies of plants that have been performed or are being performed, although some of these do not include detailed RTS analyses.

Additional methodological guidance is given in the PRA Procedures Guide, NUREG/CR–2300 [6], January 1983. This Guide was developed jointly by the Commission, the American Nuclear Society and the Institute of Electrical and Electronic Engineers.

Each licensee should establish a goal or benchmark to assess the performance of the trip system. The Commission and the industry have had considerable disagreement about the "correct" or "appropriate" value of RTS unavailability. It would be more fruitful for each licensee to have a benchmark for comparison as the plant operates and generates new data. The treatment of common cause failures will be analyzed in a qualitative fashion to determine if there are any significant failure modes previously unidentified. The cost of doing this can be minimized by forming or using existing owners groups, since there is much commonality in RTS designs.

Each licensee, as part of the RTS unavailability analysis, should examine its maintenance, surveillance, and testing requirements. The testing frequency would be examined to determine if testing is done too often or not often enough. The type of testing, e.g., completeness and sequencing of component verification for operability, would be throughly reviewed. The nature and frequency of maintenance, e.g., lubrication, cleaning, calibration, dimensional verification, physical movement, would be reviewed. Recordkeeping procedures should be reviewed.

The Commission believes that a reliability assurance program for the reactor trip systems should be developed and implemented, with clear objective of providing additional assurance that the desired high reliability of the RTS is indeed achieved and maintained. Operating experience in the United States appears to demonstrate, in some instances, that implementation of Appendix A (particularly General Design Criterion 21) and Appendix B to 10 CFR Part 50, and other NRC regulatory requirements may not have yielded the degree of reliability that is possible to achieve with available technology in a cost-effective manner. One reason for this failure might be that a reliability standard has not been sufficiently developed nor quantitatively set down in procedures. Another reason might be a failure to understand fully the dominant role played by common cause failures.

[3] Copies of NUREG–0977 and 1000 may be purchased by calling (301) 492–9530 or by writing to the Publication Services Section, Document Management Branch, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, or purchased from the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161.

[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.

[5] These may be examined at the NRC Public Document Room, 1717 H Street, NW., Washington, D.C. 20555.

[6] Copies of this NUREG may be purchased by calling (301) 492–9530 or by writing to the Publication Services Section, Document Management Branch, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; or purchased from the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, VA 22161.

The techniques for a reliabilty assurance program are in existence. They have been applied in an orderly, structured fashion in defense and aerospace applications since at least the 1960s. However, details of its application to a commercial nuclear power plant have not been worked out. Therefore, it is strongly recommended that the development of a voluntary reliability assurance program, limited to the reactor trip system, be performed jointly by the NRC and Industry, appropriately coordinated with INPO, EPRI, and the various owners groups. If this program is not voluntarily implemented in an effective manner, the Commission will reconsider the question of rulemaking in this area.

The development of industry programs on a voluntary basis has precedence in the evaluation of operating data for commercial nuclear power plants. The industry has developed the Nuclear Plant Reliability Data (NPRD) System as a voluntary program for the reporting of reliability data. The NPRD system is now undergoing a program of substantial improvement under INPO direction with close NRC interest. Even while such improvement is underway, the NPRD system is a valuable element of a reliability assurance program.

Challenges to Safety Systems

This rule concerns itself with mitigating systems which are intended to reduce the challenge to plant safety systems due to a low probability ATWS event. However, the Commission has concluded that a reduction in the frequency of challenges to plant safety systems should be a prime goal of each licensee, and the Commission believes that ATWS risk reductions can also be achieved by reducing the much larger frequency of transients which call for the reactor protection system to operate. Challenges to the reactor protection system may arise from such things as: Unreliable components, inadequate post-trip reviews, testing, and tolerance of inadequate or degraded control systems. Operating experience in Japan indicates a transient frequency that is substantially less that in the United States. Utilities have categorized transients for over ten years but have not specifically instituted a program to reduce them. While not specifically required by this rule, the Commission urges licensees to analyze challenges to the plant safety systems, particularly the reactor trip system, so as to determined where improvements can be made.

Considerations Regarding System and Equipment Criteria

The Commission place a high premium on hardware, operating practices and maintenance practices which will reduce the frequency of challenges to plant safety systems. Therefore equipment required by this rule should be of sufficient quality and reliability so as to perform its intended function while at the same time minimizing the potential for transients, e.g., inadevertent scrams, which challenge other safety systems.

The additional equipment required by this amendment to implement diversity for auxiliary feedwater system initiation, turbine trip, recirculation pump trip, and reactor trip, while required to be reliable, will not have to meet all of the stringent requirements normally applied to safety-related equipment. The equipment required by this amendment is for the purpose of reducing the probability of unacceptable consequences following anticipated operational occurrences. Since the combination of an anticipated operational occurrence, failure of the existing reactor trip system, and a seismic event or an event which results in significant plant physical damage has a low probability, seismic qualification and physical separation criteria need not be applied to the equipment required by this rule. In view of the redundancy provided in existing reactor trip systems, the equipment required by this amendment does not have to be redundant within itself.

The amendment is to require diversity to those portions of existing reactor trip systems, where only minimal diversity is currently provided. The logic circuits and actuation devices (e.g., circuit breakers on pressurized water reactors) in existing reactor trip systems utilize redundant, but in general identical, components and thus are subject to potential common cause failures. Existing reactor trip systems, however, measure a variety of plant parameters and utilize a variety of sensor types. Common cause failures in the diverse sensors of existing reactor trip systems are considered sufficiently unlikely that additional sensor diversity is not necessary. Even though sensor diversity is not necessary, it is desirable that sensors in the existing reactor trip system not be used to provide the signals for the diverse equipment required by this amendment. Use of the same sensor for the existing reactor trip system and the diverse equipment would result in interconnections between the two systems that are difficult to analyze and which could increase the potential for common cause failures affecting both systems. Since the sensors for the equipment required by this amendment do not have to be safety related, there should be considerable flexibility for using existing sensors without using reactor trip system sensors. However, there may be some cases where the use of less than safety-related sensors would result in increased risk from frequent safety system challenges or where it would not be cost effective to use sensors separate from those in the existing reactor trip system. This is particularly the case where not using sensors in the existing reactor trip system would result in the need to install a new sensor connected to the reactor coolant system. This could result in significant radiation does to personnel making the modifications. Another case would be where installation of additional containment penetrations would be required. In cases where existing protection system sensors are used to provide signals to the diverse equipment, particular emphasis should be placed on the design of the method used to isolate the signal from the existing protection system to minimize the potential for adverse electrical interactions.

The equipment required by this amendment must be implemented such that it does not degrade the existing protection system. This is to be accomplished by making the diverse equipment electrically independent to the extent practicable from the existing protection system and by insuring that the existing protection system will continue to meet all applicable safety-related criteria after installation of the diverse equipment.

The following table illustrates the system specifications that the staff would find acceptable for the diverse scram and mitigating systems. The staff will publish this guidance in a Regulatory Guide or Standard Review Plan revision which will also cover

testing, maintenance, and surveillance. Additionally, the staff will issue explicit QA guidance for the non-safety related equipment in the form of a generic letter. The generic letter will specify which requirements of the following sections of Appendix B are to be applied to non-safety related equipment: (1) Instructions, procedures, and drawings, (2) document control, (3) inspection, (4) test control, (5) control of measuring and testing equipment, (6) inspection, test, and operating status, (7) corrective action, and (8) quality assurance records.

Exemptions

Some of the older operating nuclear power plants (e.g., those licensed to operate prior to August 22, 1969) may be granted an exemption from these amendments if they can demonstrate that their risk from ATWS is sufficiently low. Factors important to this demonstration could be power level, unique design features that could prevent or mitigate the consequences of an ATWS, remaining plant lifetime, or remote siting.

GUIDANCE REGARDING SYSTEM AND EQUIPMENT SPECIFICATIONS

System / lance	Diverse Reactor Trip System	Mitigating Systems (Recirculation Pump Trip and Automatic SLCS actuation for BWRs: Auxiliary Feedwater Actuation and Turbine Trip for PWRs)*
aty Related (E-279)	Not required, but the implementation must be such that the existing protection system continues to meet all applicable safety related criteria.	Not required, but the implementation must be such that the existing protection system continues to meet all applicable safety related criteria.
undancy	Not required.	Not required.

xisting recirculation pump trip equipment installed in BWRs in accordance with vious staff requirements for the mitigat'cn of anticipated transients without am need not be modified.

System / Guidance	Diverse Reactor Trip System	Mitigating Systems (Recirculation Pump Trip and Automatic SLCS actuation for BWRs: Auxiliary Feedwater Actuation and Turbine Trip for PWRs)*
Diversity from existing Reactor Trip System	Equipment diversity to the extent reasonable and practicable to minimize the potential for common cause failures is required from the sensors to and including the components used to interrupt control rod power or vent the scram air header. Circuit breakers from different manufacturers alone is not sufficient to provide the required diversity for interruption of control rod power. The sensors need not be of a diverse design or manufacturer. Existing protection system instrument-sensing lines may be used. Sensors and instruments-sensing lines should be selected such that adverse interactions with existing control systems are avoided.	Equipment diversity to the extent reasonable and practicable to minimize the potential for common cause failures is required from the sensors to, but not including, the final actuation device--e.g., existing circuit breakers may be used for auxiliary feedwater initiation. The sensors need not be of a diverse design or manufacturer. Existing protection system instrument-sensing lines may by used. Sensors and instrument-sensing lines should be selected such that adverse interactions with existing control systems are avoided.
Electrical Independence from existing Reactor Trip System	Required from sensor output to the final actuation device at which point non-safety related circuits must be isolated from safety related circuits.	Required from sensor output to the final actuation device at which point non-safety related circuits must be isolated from safety related circuits.

[7590-01]

Guidance \ System	Diverse Reactor Trip System	Mitigating Systems (Recirculation Pump Trip and Automatic SLCS actuation for BWRs: Auxiliary Feedwater Actuation and Turbine Trip for PWRs)*
Physical Separation from existing Reactor Trip System	Not required, unless redundant divisions and channels in the existing reactor trip system are not physically separated. The implementation must be such that separation criteria applied to the existing protection system are not violated.	Not required, unless redundant divisions and channels in the existing reactor trip system are not physically separated. The implementation must be such that separation criteria applied to the existing protection system are not violated.
Environmental Qualification	For anticipated operational occurrences only, not for accidents.	For anticipated operational occurrences only, not for accidents.
Seismic Qualification	Not required.	Not required.
Quality Assurance for Test, Maintenance, and Surveillance	Explicit guidance will be issued in a letter.	Explicit guidance will be issued in a letter.
Safety-Related (1E) Power Supply	Not required, but must be capable of performing safety functions with loss of offsite power. Logic and actuation device power must be from an instrument power supply independent from the power supplies for the existing reactor trip system. Existing RTS sensor and instrument channel power supplies may be used provided the possibility of common mode failure is prevented.	Not required, but must be capable of performing safety functions with loss of offsite power. Logic power must be from an instrument power supply independent from the power supplies for the existing reactor trip system. Existing RTS sensor and instrument channel power supplies may be used provided the possibility of common mode failure is prevented.
Testability at Power	Required.	Required.

[7590-01]

Guidance \ System	Diverse Reactor Trip System	Mitigating Systems (Recirculation Pump Trip and Automatic SLCS actuation for BWRs: Auxiliary Feedwater Actuation and Turbine Trip for PWRs)*
Inadvertent Actuation	The design should be such that the frequency of inadvertent reactor trips and challenges to other safety systems is minimized.	The design should be such that the frequency of inadvertent actuation and challenges to other safety systems is minimized.

With the promulgation of this final ATWS rule, the Commission has completed action on PRM-50-29. The petitioner's requests have been granted in part through the incorporation of requirements into the final rule which address the following issues: (1) (For GE BWRs) (a) recirculation pump trip following an event indicative of an ATWS, and (b) independent, redundant and diverse electrical initiation of scram following an event indicative of an ATWS; (2) (For CE and B&W PWRs) automatic initiation of auxiliary feedwater independent of the reactor protection system; and (3) (For Westinghouse PWRs) automatic initiation of turbine trip and auxiliary feedwater independent of the reactor protection system. The petitioner's request for promulgation of specific provisions *within the context of an ATWS rulemaking* for the following systems are hereby denied: (1) (For GE BWRs) a scram discharge volume system (this provision was not included in the final ATWS rule because licensees already have installed or are installing this system); and (2) (For CE and B&W PWRs) an alternate means to shut down the reactor that is diverse from and redundant to the electrical portion of the reactor protection system *up to but not including* the *trip breakers* (the final ATWS rule includes a requirement for the installation of an alternate shut-down system which *must include* the trip breakers).

Additional View of Commissioner Asselstine

While I approve this rule, I would have required automation of the Standby Liquid Control System (SLCS) for all boiling water reactors. In addition, while I approve the elements of the final rule dealing with future reactors, I am not satisfied that sufficient attention has been given to future reactors. It appears that significant additional reductions in the ATWS risk can be achieved without incurring insurmountable economic costs if such measures are considered during the design phase. I believe this rule should not be taken as a barrier to further consideration of measures for future reactors that can reduce ATWS risk below that achieved by this rule.

Additional Views of Commissioner Roberts

In addition to specifying measures to reduce the risk from ATWS events, the Statement of Considerations which accompanies this rule directs licensees to "volunteer" to implement a reliability assurance program for the Reactor Trip System.

The Reactor Trip System is one of the most important safety systems at commercial nuclear power plants. However, it is only one of many safety-related systems which must be closely monitored and carefully maintained to ensure a plant's safety and reliability. It is my view that a more logical approach to reliability assurance would be to consider such a program embracing those several safety systems which experience and analyses show could be significantly improved by such a program. This program should be reviewed separately from the ATWS rulemaking effort.

Furthermore, the Commission should not call upon the industry to implement complicated and costly reliability assurance programs until it more thoroughly analyzes the concept and until it provides specific guidance.

Regulatory Analysis

The Commission has prepared a regulatory analysis for this regulation. The analysis examines the costs and benefits of the rule as considered by the Commission. A copy of the regulatory analysis is available for inspection and copying for a fee at the NRC Public Document Room, 1717 H Street, NW., Washington, D.C. Single copies of the analysis may be obtained from David W. Pyatt, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Telephone (301) 443-7631.

Paperwork Reduction Act Statement

This final rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et. seq.). These requirements were approved by the Office of Management and Budget approval number 3150-0011.

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980, 5 U.S.C. 605(b), the Commission hereby certifies that the rule will not have a significant economic impact on a substantial number of small entities. This rule affects only licensees that own and operate nuclear utilization facilities licensed under sections 103 and 104 of the Atomic Energy Act of 1954, as amended. These licensees do not fall within the definition of small businesses set forth in section 3 of the Small Business Act, 15 U.S.C. 632, or within the Small Business Size Standards set forth in 13 CFR Part 121.

List of Subjects in 10 CFR Part 50

Antitrust, Classified information, Fire prevention, Intergovernmental relations, Incorporation by reference, Nuclear power plants and reactors, Penalty, Radiation protection, Reactor siting criteria, and Reporting and recordkeeping requirements.

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and sections 552 and 553 of Title 5 of the United States Code, the following amendment to 10 CFR Part 50 is published as a document subject to codification.

PART 50—DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

1. The authority citation for Part 50 continues to read as follows:

Authority: Secs. 103, 104, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2133, 2134, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, 202, 206, 88 Stat. 1242, 1244, 1246, as amended (42 U.S.C. 5841, 5842, 5846), unless otherwise noted.

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Sections 50.57(d), 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2071, 2073 (42 U.S.C. 2133, 2239). 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). Sections 50.100-50.102 also 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), §§ 50.10 (a), (b), and (c), 50.44, 50.46, 50.48, 50.54, and 50.80(a) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 50.10(b) and (c) and 50.54 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.55(e), 50.59(b), 50.70, 50.71, 50.72, 50.73, and 50.78 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. A new § 50.62 is added to read as follows:

§ 50.62 Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants.

(a) *Applicability.* The requirements of this section apply to all commercial light-water-cooled nuclear power plants

(b) *Definition.* For purposes of this section, "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 trip portion of the protection system specified in General Design Criterion 20 of Appendix A of this part.

(c) *Requirements.* (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

(2) Each pressurized water reactor manufactured by Combustion Engineering or by Babcock and Wilcox must have a diverse scram system from the sensor output to interruption of power to the control rods. This scram system must be designed to perform its function in a reliable manner and be independent from the existing reactor trip system (from sensor output to interruption of power to the control rods).

(3) Each boiling water reactor must have an alternate rod injection (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must have redundant scram air header exhaust valves. The ARI must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

(4) Each boiling water reactor must have a standby liquid control system (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution. The SLCS and its injection location must be designed to perform its function in a reliable manner. The SLCS initiation must be automatic and must be designed to perform its function in a reliable manner for plants granted a construction permit after July 26, 1984, and for plants granted a construction permit prior to July 26, 1984, that have already been designed and built to include this feature.

(5) Each boiling water reactor must have equipment to trip the reactor coolant recirculating pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

(6) Information sufficient to demonstrate to the Commission the adequacy of items in paragraphs (c)(1) through (c)(5) of this section shall be submitted to the Director, Office of Nuclear Reactor Regulation.

(d) *Implementation.* By 180 days after the issuance of the QA guidance for non-safety related components each licensee shall develop and submit to the Director of the Office of Nuclear Reactor Regulation a proposed schedule for meeting the requirements of paragraphs (c)(1) through (c)(5) of this section. Each shall include an explanation of the schedule along with a justification if the schedule calls for final implementation later than the second refueling outrage after July 26, 1984, or the date of issuance of a license authorizing operation above 5 percent of full power. A final schedule shall then be mutually agreed upon by the Commission and licensee.

Dated at Washington, DC, this day of 19th day of June 1984.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 84-16839 Filed 6-25-84; 8:45 am]

BILLING CODE 7590-01-M

DEPARTMENT OF TRANSPORTATION

Federal Aviation Administration

14 CFR Part 39

[Docket No. 84-NM-44-AD; Amdt. 39-4882]

Airworthiness Directives; Canadair Model CL-600 and CL-601 Airplanes

AGENCY: Federal Aviation Administration (FAA), DOT.

ACTION: Final rule.

SUMMARY: This document adds a new airworthiness directive which supersedes two existing airworthiness directives (AD) applicable to the Canadair Model CL-600 and CL-601 airplanes. These AD's require repetitive inspections of the outboard flap vane attachment structure. The manufacturer has modified the outboard flaps on all airplanes, making some inspection requirements unnecessary. This rule consolidates and revises the inspections contained in the existing AD's.

EFFECTIVE DATE: June 27, 1984.

ADDRESSES: The service information specified in this AD may be obtained upon request to Canadair Ltd, Commercial Aircraft Technical Services, Box 6087, Station A, Montreal, PQ H3C 3G9, Canada, or may be examined at the address shown below.

FOR FURTHER INFORMATION CONTACT: Mr. Lester Lipsius, Airframe Section, ANE-172, New York Aircraft Certification Office, FAA, New England Region, 181 S. Franklin Avenue, Room 202, Valley Stream, New York 11581, telephone (516) 791-6220.

SUPPLEMENTARY INFORMATION: AD 83-14-06, Amendment 39-4687 (48 FR 33245; July 21, 1983), and telegraphic AD T83-20-51, issued September 30, 1983, require inspection of the wing outboard flap vane support structure for cracks. The manufacturer has since modified the outboard flap design so that some of the inspections prescribed by these AD's are no longer required. The repetitive inspection intervals may also be increased. The FAA has been advised that all airplanes in the world fleet have been modified in accordance with the manufacturer's instructions. The Canadian Department of Transport has issued an AD which reflects the revised repetitive inspections. This amendment incorporates the revised inspections and intervals and supersedes AD's 83-14-06 and T83-20-51.

This airplane model is manufactured in Canada and type certificated in the United States under the provisions of § 21.29 of the Federal Aviation Regulations and the applicable airworthiness bilateral agreement.

This amendment combines the inspection requirements of two existing AD's and imposes no additional regulatory or economic burden on any person. Further, it deletes inspection requirements that now are superfluous due to modification of the affected aircraft, therefore, notice and public procedure hereon are unnecessary and contrary to the public interest, and good cause having been shown therefor, the amendment may be made effective in less than 30 days.

List of Subjects in 14 CFR Part 39

Aviation safety, Aircraft.

Adoption of the Amendment

Accordingly, pursuant to the authority delegated to me by the Administrator, § 39.13 of Part 39 of the Federal Aviation Regulations (14 CFR 39.13) is amended by adding the following new airworthiness directive:

Canadair: Applies to Model CL-600-1A11 (CL-600) and Model CL-600-2A12 (CL-601) airplanes, certificated in all categories. Compliance required as indicated.

A. To detect possible fatigue cracks in the outboard flap vane support structure, accomplish the following inspections for cracks on each side of the aircraft, initially within 100 hours time in service after the effective date of this AD, unless already accomplished, and thereafter at intervals not to exceed 100 hours time in service.

1. Visually inspect the following parts:

a. The flap vane support straps, P/N 600-10460-13 and -23, at the inboard and outboard ends of the outboard flap.

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Dalwyn R. Davidson
VICE PRESIDENT
SYSTEM ENGINEERING AND CONSTRUCTION

August 13, 1982

Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
ATWS Mitigation Design Features

Dear Mr. Schwencer:

In a meeting on July 20, 1982, between CEI and members of NRR, we discussed our plans for changing several systems associated with the mitigation of a postulated ATWS event. The Perry Nuclear Power Plant (PNPP) construction schedule and the current ATWS rulemaking schedule have made it necessary for us to anticipate potential future requirements. We believe it to be in our best interest to modify the current design and install these systems during our construction as opposed to waiting until the construction of these systems impacts our construction schedule or operations. The inclusion of these systems is based on the proposed rulemaking and is not based on a belief by CEI that these systems are needed to mitigate an ATWS event. As such, we maintain our support of industry comments on the proposed ATWS rule.

The basic changes to be made to the Perry plant include the following:

(1) An increased flow capacity for the Standby Liquid Control System from 43 gpm to 86 gpm. This will involve increasing the size of both pumps' suction lines as well as changing the reactor vessel injection point to the HPCS header. Although the design includes both manual and automatic initiation capability, only manual initiation will be functional. The existing pumps will be used.

(2) An upgrade to safety grade of the Recirculation Pump Trip initiation circuitry.

(3) A control grade feedwater runback feature.

(4) An Alternate Rod Insertion system which is redundant to the Reactor Protection System scram logic.

ATTACHMENT 2

The details of the above described design will be submitted as an amendment to the Perry FSAR by January 1983.

We believe that this design along with appropriate emergency operating procedures and training adequately address the ATWS issue for PNPP.

Very truly yours,

Dalwyn R. Davidson

Dalwyn R. Davidson
Vice President
System Engineering and Construction

DRD:WEC:mb

cc: Jay Silberg, Esq.
John Stefano
Max Gildner

CERTIFICATE OF SERVICE

DOCKETED
USNRC

This is to certify that copies of the foregoing were served by deposit in the U.S. Mail, first class, postage prepaid this 6th day of July, 1984 to those on the service list below.

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