ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO AMENDMENT 22 TO GENERAL ELECTRIC TOPICAL REPORT NEDE-24011-P-A "GENERAL ELECTRIC STANDARD APPLICATION FOR REACTOR FUEL"

1.0 INTRODUCTION

In order to support customer needs and remain competitive, the fuel vendors are continually improving their fuel designs. Generally, the changes in design are made with approved methodologies. The regulatory procedures to qualify and approve the new designs are standard. However, the review and approval of these new designs place a burden on the staff resources.

Recently, the staff proposed that a set of acceptance criteria, to be satisfied by new fuel designs, be established for each fuel vendor. Once the acceptance criteria are approved, the fuel designs or changes satisfying the criteria would not require explicit staff review. Satisfaction of the acceptance criteria would be sufficient for approval by reference to the acceptance criteria. Also, the staff requires that the acceptance criteria be entirely non-proprietary so that any interested party will have access to the acceptance criteria. The objective of this approach is to expedite the review process and reduce the staff resources needed for review of new fuel designs.

In response to the staff proposal, General Electric submitted a topical report, "Proposed Amendment 22 to GE Licensing Topical Report NEDE-24011-P-A," dated September 9, 1988, proposing the new fuel licensing acceptance criteria for staff review. The proposed acceptance criteria consider fuel thermal-mechanical, nuclear, and thermal-hydraulic aspects of design analyses. If a fuel design complies with the fuel acceptance criteria, it is acceptable for licensing applications without the explicit staff review. In a letter dated May 10, 1989 from J. S. Charnley (GE) to M. W. Hodges (USNRC), GE further stated that all the fuel design information in GESTAR II will be transferred

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to GE Fuel Bundle Designs Information Report (NEDE-31152P) and will be provided to the NRC. The staff's evaluation of the proposed criteria follows.

2.0 EVALUATION

2.1 General Criteria

GE has established five general criteria to deal with generic problems:

(1) "NRC-approved analytical models and analysis procedures will be applied."

This statement is consistent with the current and past practices for new fuel designs. Therefore, this is acceptable.

(2) "New design features will be included in lead test assemblies."

The staff requires that significant new design features be tested before full implementation. GE philosophy apparently is consistent with the staff approach. Therefore, this is acceptable.

(3) "The generic post-irradiation fuel examination program approved by the NRC will be maintained."

We have generically approved the GE post-irradiation examination program in a letter from L. S. Rubenstein (NRC) to R. L. Gridley (GE), dated June 27, 1984. GE will continue to use this approved program for new fuel designs. This is acceptable.

(4) "New-fuel-related licensing issues identified by the NRC will be evaluated to determine if the current criteria properly address the concern; if necessary, new criteria will be proposed to the NRC for approval."

If, for a future new-fuel design, the existing fuel licensing acceptance criteria cannot adequately address all concerns, GE will submit new criteria for the staff review. This approach is acceptable.

(5) "If any of the criteria in subsection 1.1 (of Amendment 22) are not met for a new fuel design, that aspect will be submitted for review by the NRC separately."

If for future new fuel design, any fuel licensing acceptance criteria are not satisfied, GE will submit that specific area for the staff review. However, the related parts or the whole parts associated with this specific area in the fuel design may have to be submitted in order to assist the staff review. This approach is acceptable.

For future new fuel designs which satisfy the licensing acceptance criteria, we require that GE notify NRC of the first application of each new fuel design based on the approved fuel licensing criteria, and that the fuel bundle design information report be submitted to NRC prior to loading of the new fuel into a reactor.

2.2 Thermal-Mechanical Fuel Licensing Acceptance Criteria

(1) "The fuel rod and fuel assembly component stresses, strains, and fatigue life usage shall not exceed the material ultimate stress or strain and the material fatigue capability."

GE uses a design ratio concept, that is, a ratio of effective stress/strain to ultimate stress/strain, to determine the adequacy of the stress and strain loading on fuel rod and fuel assembly. The design ratio must be less than unity. The design ratio concept was approved in Amendment 7 to GESTAR II. Therefore, the staff considers that the stress and strain criteria are acceptable. For strain fatigue analysis, GE requires that fatigue life usage, that is, the ratio of actual number of cycles to stress or strain to allowable cycles at stress or strain, must be less than unity. This fatigue criterion was also approved in Amendment 7 to GESTAR II; therefore, this criterion is acceptable.

(2) "Mechanical testing will be performed to ensure that loss of fuel rod and assembly component mechanical integrity will not occur due to fretting wear when operating in an environment free of foreign material."

The Standard Review Plan (SRP) (NUREG-0800) states that allowable fretting wear should be considered in the overall safety analyses. GE considers the effect of fretting wear in design analyses based on testing and experience in reactor operations. The proposed fretting wear criterion was approved in Amendment 7 to GESTAR II; thus, this criterion is acceptable.

(3) "The fuel rod and assembly component evaluations include consideration of metal thinning and any associated temperature increase due to oxidation and the buildup of corrosion products to the extent that these effects influence the material properties and structural strength of the components."

GE considers the effect of external corrosion and buildup of crud on the cladding surface in the design analysis. This approach is consistent with the GE past experience and was approved in Amendment 7 to GESTAR II; thus, this criterion is acceptable.

(4) "The fuel rod internal hydrogen content is controlled during manufacture of the fuel rod consistent with ASTM standards C776-83 and C934-85 to assure that loss of fuel rod mechanical integrity will not occur due to internal cladding hydriding."

Hydriding as a cladding failure mechanism is precluded by controlling the level of moisture and other hydrogenous impurities during fabrication. GE employed ASTM [American Society for Testing and Materials] standards to control the hydrogen content. Since the use of ASTM standards is consistent with the SRP, this criterion is acceptable.

(5) "The fuel rod is evaluated to ensure that fuel rod or channel bowing does not result in loss of fuel rod mechanical integrity due to boiling transition."

Fuel assembly components such as fuel rods and channel boxes may undergo various types of dimensional changes such as rod bowing, irradiation growth, and channel box deflection. The result of irradiation growth is rod bowing or channel box deflection. Such phenomena are related to neutron fluence, fuel burnup, and core residence time. Rod bowing can affect local nuclear power peak and heat transfer to the coolant. Channel box deflection can also affect fuel assembly performance with respect to boiling transition. GE-proposed acceptance criteria of rod bowing and channel box deflection are consistent with the previously approved design criteria; thus, this criterion is acceptable.

(6) "Loss of fuel rod mechanical integrity will not occur due to excessive cladding pressure loading."

In the SRP, the staff stated that fuel and burnable poison rod internal gas pressures should remain below the nominal system pressure during normal operation, unless otherwise justified. GE proposed an alternative to rod pressure criterion in Amendment 7 to GESTAR II. The staff approved the GE proposal of rod pressure; therefore, this GE-proposed criterion is acceptable.

(7) "The fuel assembly (including channel box) control rod and control rod drive are evaluated to assure control rods can be inserted when required."

The control rod insertability is required during combined seismic and loss-of-coolant accident (LOCA) loading. GE described these analyses in the approved report NEDE-21175-3-P; which also dealt with the assembly vertical liftoff analyses as part of control rod insertability requirement. This control rod insertability criterion is acceptable.

(8) "Loss of fuel rod mechanical integrity will not occur due to cladding collapse into a fuel column axial gap."

If axial gaps in the fuel column were to occur as a result of densification, the cladding would have the potential of collapsing into a gap. The GE-proposed cladding structural design criterion to preclude collapse was approved in Amendment 7 to GESTAR II; thus, this criterion is acceptable.

(9) "Loss of fuel rod mechanical integrity will not occur due to fuel melting."

The GE design basis for fuel pellet overheating is that the fuel rod is evaluated to ensure that fuel rod failure due to fuel melting will not occur during normal operation and corewide anticipated operational occurrences (AOOs). This criterion was approved in Amendment 7 to GESTAR II; thus, it is acceptable.

(10) "Loss of fuel rod mechanical integrity will not occur due to pellet-cladding mechanical interaction."

Fuel failures due to pellet-cladding interaction have been encountered in operating boiling-water-reactor (BWR) fuel. The SRP stated that to preclude pellet-cladding-interaction (PCI) failures, two criteria should be observed, although they may not be sufficient: (a) the cladding uniform strain should not exceed 1 percent and (b) fuel melting should be avoided. In the safety evaluation of Amendment 7 to GESTAR II, the staff concluded that the GE design criteria are consistent with the SRP thus, the PCI criterion is acceptable.

2.3 Nuclear Licensing Acceptance Criteria

(1) "A negative Doppler reactivity coefficient shall be maintained for any operating conditions."

A negative Doppler coefficient guarantees instantaneous negative reactivity feedback to any rise in fuel temperature, thus providing an inherent self-control feature of BWR fuel. This criterion, was previously approved in GESTAR II; thus, it is an acceptable fuel design limitation.

(2) "A negative core moderator void reactivity coefficient resulting from boiling in the active flow channels shall be maintained for any operating conditions."

A negative core moderator void coefficient in the active flow channels flattens the radial power distribution and provides ease of reactor control due to negative void feedback. This criterion was previously approved in GESTAR II; thus, it is acceptable.

(3) "A negative moderator temperature coefficient shall be maintained above hot standby."

The moderator temperature coefficient is associated with the change in the water moderating capability. A negative moderator temperature coefficient during power operation provides inherent protection against power excursion. Since this criterion is consistent with the SRP, it is acceptable.

(4) "For a super prompt critical reactivity insertion accident (e.g., control rod drop accident) originating from any operating condition, the net prompt reactivity feedback due to prompt heating of the moderator and fuel shall be negative."

To mitigate the effects of a superpromp[illegible]itical reactivity insertion accident such as a control rod drop accident, the mechanical and nuclear fuel design shall be such that the prompt reactivity feedback provides an automatic shutdown mechanism. The negative prompt reactivity feedback criterion is consistent with the SRP requirement and is, thus, acceptable.

(5) "A negative power coefficient, as determined by calculating the reactivity change, due to an incremental power change from a steady-state base power level, shall be maintained for all operating power levels above hot standby."

A power coefficient like the Doppler coefficient or moderator temperature coefficient is maintained negative for reactivity control. Since this power coefficient criterion is consistent with the SRP requirement, it is acceptable.

(6) "The plant shall be calculated to meet the cold shutdown margin requirement for each plant cycle specific analysis."

The core must be designed to remain subcritical with adequate margin with the most reactive control rod in its fully out position and all other rods fully inserted. Since this criterion satisfies the SRP requirement, it is acceptable.

(7) "The effective multiplication factor for new fuel designs stored under normal and abnormal conditions shall be shown to meet fuel storage limits by demonstrating that the peak uncontrolled lattice k-infinity calculated in a normal reactor core configuration meets the limits provided in Section 3 (of GESTAR-II) for GE-designed regular or high density storage racks."

For fuel storage racks, the design criterion [illegible] k-infinity is 0.90 for regular racks, and 0.95 for high-density racks, and has been previously approved in GESTAR II. Since this acceptance criterion is consistent with the GESTAR II requirements and the SRP, it is acceptable.

2.4 <u>Thermal-Hydraulic Licensing Evaluation</u>

"Flow pressure drop characteristics shall be included in plant cycle specific analyses for calculation of the Operating Limit MCPR."

GE stated that a coupled thermal-hydraulic-nuclear analyses will be performed to determine fuel bundle flow and power distribution using the various bundle pressure loss coefficients applicable to a plant specific cycle analyses. The margin to the thermal limits, for example, the maximum critical power ratio (MCPR), of each bundle is determined using the set of calculated bundle flow and power. GE will explicitly model these pressure drop characteristics in the analysis. Since the GE approach is consistent with [illegible] SRP, this evaluation approach is acceptable.

2.5 <u>Safety Limit MCPR Licensing Evaluation</u>

(1) "Safety Limit MCPR shall be recalculated following steps in Subsection 1.2.5.B (of Amendment 22) or reconfirmed when a new fuel design or new critical power [illegible]rrelation is introduced."

The safety lim[illegible]CPR [illegible]nsitive to bundle design and critical power correlations, for exampl[illegible] GEXL or GEXL-PLUS. The bundle design depends on rod diameter, [illegible]erm[illegible] [illegible]me constant, spacer, and bundle R-factor. Any change in fuel de[illegible] [illegible] critical power correlation will affect the safety limit MCPR. T[illegible]refore, recalculation or reconfirmation of MCPR is necessary. Th[illegible] evaluation commitment is acceptable.

(2) GE has establ[illegible] si[illegible] [illegible]ons to be assumed when calculating the safety limit M[illegible] [illegible]ditions are consistent with the current procedures de[illegible] [illegible] approved GESTAR II. Therefore, these six conditions for [illegible] MCPR analysis are acceptable.

2.6 <u>Operating Limit MCPR Lic[illegible]nsing Evaluation</u>

(1) "Plant Operating Limit [illegible]R is established by considering the limiting anticipated operational occurrences for each operating cycle."

The operating limit MCPR is determined by adding the change in the MCPR for the limiting AOO to the safety limit MCPR. The limiting AOO events are described in the approved GESTAR II. GE stated that the operating limit MCPR is cycle dependent, and is calculated prior to the cycle operation. This procedure is consistent with the approved GESTAR II approach and is, therefore, acceptable.

(2) "For each new fuel design the applicability of generic MCPR analyses described in Section 4 (of GESTAR II) or in the country-specific supplement to this base document shall be confirmed for each operating cycle or a plant specific analysis will be performed."

The applicability of new fuel design to the generic MCPR analyses needs to be examined. GE will document its applicability to the rod withdrawal error in the fuel design information report. This acceptance criterion is consistent with the approved approaches in GESTAR II; thus, it is acceptable.

2.7 Critical Power Correlation Licensing Evaluation

(1) "The currently approved critical power correlations will be confirmed or a new correlation will be established when there is a change in wetted parameters of the flow geometry; this specifically includes fuel and water rod diameter, channel sizing and spacer design."

The coefficients for the critical power correlation are dependent on fuel assembly design parameters. Any change in fuel design will affect the critical power correlation. Thus, the existing correlation must be confirmed or a new correlation must be developed. The proposed acceptance criterion is consistent with the staff position, but should be revised to clarify that the correlation will be confirmed or a new correlation established if any of the parameters impacting the correlation, e.g., rod-to-rod peaking factors, are significantly outside of the range tested. With this understanding, the criterion is acceptable.

(2) "A new correlation may be established if significant new data exist for a fuel design."

When significant new data have been generated for a fuel design, a better fit to the data may be achieved by adjusting the coefficients in the critical power correlation. This acceptance criterion is consistent with the staff position; thus, it is acceptable.

(3) GE has established seven conditions to be assumed when calculating a new critical power correlation. These conditions are consistent with the current procedures described in the approved GESTAR II. Therefore, these seven conditions for establishing critical power correlation are acceptable.

2.8 Stability Licensing Acceptance Criteria

The new fuel design must meet either of the two criteria described below.

(1) "The stability behavior, as indicated by core and limiting channel decay ratios, must be equal to or better than a previously approved GE BWR fuel design."

The GE thermal-hydraulic stability licensing criteria set forth in Amendment 8 to GESTAR II allow for the application of the generic bounding stability approach to new fuel designs provided that the stability performance of the new design is bounded by that of current fuel design. GE describes six steps for evaluating the new fuel stability performance against the currently approved fuel design. The staff has approved the GE stability licensing criteria; thus, this stability criterion is acceptable.

(2) "If the core and limiting channel decay ratios are not equal to or better than a previously approved GE fuel design, it must be demonstrated that there is no change to the exclusion zone."

If the new fuel design cannot meet the preceding stability acceptance criterion, then GE will demonstrate that there is no change to the exclusion zone as an alternate method for stability acceptance evaluation. The GE proposed criterion is consistent with the staff position; thus, it is acceptable.

2.9 Overpressure Protection Analysis Licensing Evaluation

(1) "Adherence to the ASME overpressure protection criteria shall be demonstrated on plant cycle specific analysis."

GE will demonstrate the adequacy of the plant overpressure protection system on cycle-specific analysis based on core loading pattern. This approach is consistent with the current procedures described in the approved GESTAR II; therefore, this approach is acceptable.

2.10 Loss-of-Coolant Accident Analysis Methods Licensing Evaluation

(1) "The criteria in 10 CFR 50.46 shall be met on plant-specific or bounding analyses."

The emergency core cooling system (ECCS) criteria in 10 CFR 50.46 are met by the exposure-dependent maximum average planar linear heat-generation rate (MAPLHGR) limit in plant-specific or bounding analyses. GE will continue to evaluate these ECCS criteria for any new fuel design; thus, this approach is acceptable.

(2) "Plant MAPLHGR adjustment factors must be confirmed when a new fuel design is introduced."

Plant MAPLHGR limit is sometimes adjusted for a special operational configuration or region. GE will confirm the revised MAPLHGR limit for new fuel design before each cycle operation; thus, this approach is acceptable.

2.11 Rod Drop Accident Analysis Licensing Evaluation

(1) "Plant cycle specific analysis results shall not exceed the licensing limit in GESTAR-II."

The current licensing limit of control rod drop accident analysis is 280 cal/gm. GE will perform the rod drop analysis each cycle to ensure compliance with the 280 cal/gm licensing limit. The GE acceptance criterion is consistent with the licensing criterion; therefore, it is acceptable.

(2) "Applicability of the bounding BPWS analysis must be confirmed."

The bounding rod drop accident analysis for plants with bank position withdrawal sequence (BPWS) procedure is dependent on the fuel design and should be confirmed for each new fuel design. The staff agrees with the GE assessment; therefore, this approach is acceptable.

2.12 Refueling Accident Analysis Licensing Evaluation

"The consequence of a refueling accident as presented in the country-specific supplement GESTAR-II or the plant FSAR shall be confirmed as bounding or a new analysis shall be performed."

The consequence of refueling accident is mainly dependent on the amount of fuel rods in a bundle. If there is a change in the number of fuel rods or a new fuel design is proposed, the effect on the refueling accident must be reconfirmed or reanalyzed; therefore, this approach is acceptable.

2.13 Anticipated Transient Without Scram Licensing Acceptance Criteria

The new fuel must meet either of the acceptance criteria described below:

(1) "A negative core moderator void reactivity coefficient, consistent with the analyzed range of void coefficients, shall be maintained for any operating conditions above the startup critical condition."

For core response to an anticipated-transient-without-scram (ATWS) event, the core moderator void reactivity coefficient is the key parameter. If the coefficient remains within the range of void coefficients used in the ATWS point kinetics analyses, the conclusion of BWR mitigation of an ATWS event is still valid for new fuel designs. The GE-proposed ATWS criterion is consistent with the staff guidelines; therefore it is acceptable.

(2) "If the preceding criterion is not satisfied, the limiting events will be evaluated to demonstrate that the plant response is within the ATWS criterion."

For new fuel designs that have core moderator void reactivity coefficients outside the range of point model void coefficients, a specific evaluation of ATWS will be performed for limiting cases to comply with the ATWS acceptance criteria. The GE-proposed criterion is consistent with the staff guidelines; therefore, it is acceptable.

3.0 CONCLUSION

The staff has reviewed the GE submittal, Amendment 22 to GESTAR II, and concludes that the submittal describing a set of licensing acceptance criteria and methods for new fuel design is acceptable for future licensing applications. For future reload application, we require that GE notify NRC of the first application of a new fuel design based on the approved fuel licensing criteria, and that the fuel bundle design information report be submitted to NRC prior to loading of the new fuel into a reactor. However, should NRC criteria or regulations change so that staff conclusions as to the acceptability of this submittal are invalidated, or should circumstances arise causing some criteria to be invalid, GE will be expected to revise or resubmit its documentation for further review by the staff.

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

SEP 18 1989

MEMORANDUM FOR: Edward L. Jordan, Director
Office for Analysis and Evaluation
of Operational Data

FROM: James M. Taylor
Acting Executive Director
for Operations

SUBJECT: CRGR REVIEW OF BACKFITTING APPEALS

(1) Backfitting Appeal Regarding System Hydrostatic and Leakage testing, Letter to EDO from D. Stenger, Counsel to the Nuclear Utility Backfitting and Reform Group, March 16, 1989.

(2) Appeal from Staff Decision Requiring Total Equipment Diversity under ATWS Rule (10 CFR §50.62), Letter to Acting EDO from S. Floyd, Chairman of the BWR Owners Group, August 11, 1989.

I have received two different backfit appeals both generic in nature but each originally stemming from plant-specific appeals denied pursuant to Manual Chapter 0514. Copies of these appeals are enclosed. I request a thorough CRGR review on the merit of these appeals, the underlying technical/legal issues involved, and recommendations to me on their proper disposition. I request that you also take the lead in preparing a final response for my signature for each of these appeals. I would appreciate timely CRGR review to the extent permitted by the current CRGR agenda. I have asked a member of my technical staff (M. Taylor) to give supplemental assistance on these matters should you or your CRGR staff require it. By copy of this memorandum, I am also directing NRR and RES to provide the CRGR with any and all background information to include briefings that may be required to complete its review of these appeals.

As to the substance of the appeals, each relates to BWR's and each raises questions as to the correctness of staff interpretations and positions being taken pursuant to existing applicable regulations (i.e., Appendix G and ATWS, §50.62). Questions also include the degree of liberty being taken by the staff under §50.109 which calls for a disciplined, documented analysis on new or differing staff positions. Each appeal relates to complex design and operational issues of some economic substance, but in terms of safety I would preliminarily view each issue to be relatively quite small on the overall scale of public risk. In addition to questions raised about the staff's denial process (which has basically involved reliance on the compliance exception under §50.109), each appeal raises an underlying concern as to whether the applicable regulations (Appendix G and ATWS, §50.62), in their historical development or final form, suffer through a lack of clarity or too much ambiguity. If so, then corrections should perhaps be initiated via rulemaking. I request specific CRGR comments and recommendations on these important questions.

As a future practice on all generic backfitting appeals to the EDO, I intend to refer these to the CRGR for review and recommended disposition. Please advise if you have further questions on this assignment.

James M. Taylor
Acting Executive Director
for Operations

Enclosures:
As stated

cc: T. Murley, NRR
E. Beckjord, RES
J. Heltemes, AEOD

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March 16, 1989

Read this and let's meet

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555

Attn: Victor Stello, Jr.

3/20/89

Re: Backfitting Appeal Regarding System <u>Hydrostatic and Leakage Testing</u>

Dear Mr. Stello:

Pursuant to Section 044 of Staff Manual Chapter 0514, the Nuclear Utility Backfitting and Reform Group (NUBARG) appeals a Staff denial of a claim of backfit under 10 C.F.R. § 50.109. The claim concerns a Staff interpretation of system hydrostatic and leakage testing requirements under ASME Code Section XI. NUBARG presented its claim in a letter to the Director of Nuclear Reactor Regulation on April 25, 1988. The Staff denied the claim by letter dated August 17, 1988 from the Director, Division of Engineering and Systems Technology.

BACKGROUND

A. Factual Background

This appeal is concerned with a new Staff position on the acceptability of "nuclear" hydrostatic and leakage testing by BWRs, *i.e.*, the use of nuclear power during normal startup, as opposed to heat generated by recirculation pumps, to heat up and pressurize the reactor coolant system for performance of the tests. As discussed below, this testing method is clearly permitted by Section IV.A.5 of 10 C.F.R. Part 50, Appendix G, and the Staff has recognized that there is minimal difference in safety between testing with nuclear heat rather than pump heat. In accordance with Section IV.A.3 of Appendix G, testing with nuclear heat is conducted at low power and <u>with the vessel water level within the normal range for power operation</u>.

The flexibility to use this method is important because control of reactor coolant temperature is more difficult with the use of pump heat and because testing with the reactor at low

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power permits higher system temperatures than are possible with pump heat alone. The elevated system temperatures are particularly advantageous in that testing can be performed in a region where brittle fracture is not a concern. This testing method is likely to be of increasing importance in the future as Revision 2 of Regulatory Guide 1.99 is implemented and most BWRs have to perform leakage and hydrostatic tests at higher temperatures.[1]

It should also be noted that conformance with technical and safety limits during a nuclear heat up permits a more orderly and natural sequence of events than is possible using pump heat. In addition, the pump-heat testing method could contribute to operational unreliability of the affected systems, primarily the plant's main recirculation pumps, since running the pumps outside their normal design conditions could jeopardize pump performance characteristics or at a minimum increase pump maintenance (e.g., for seal wear).

There are also substantial costs associated with the pump-heat testing method. The use of pump heat for testing may add up to three days to the duration of refueling outages due to the much longer heatup times, and in fact may not be a viable option for some plants without significant plant modifications. At current costs of replacement power, the potential delay in startup could cost well in excess of $1 million per reactor per operating cycle. Mandating the pump-heat testing method thus would carry substantial costs without any comparable safety benefit.

B. New Staff Position

The Staff has taken the position that hydrostatic and leakage tests must be conducted with the reactor in a noncritical state. This position was referred to in a Staff letter of April 10, 1987 denying a technical appeal by one licensee on this issue.[2] The April 10, 1987 letter dealt solely with the technical merits of the issue rather than backfitting implications; it concluded that in view of the environmental conditions (higher temperatures and the plant at low power) in which inspection personnel would have to work, nuclear pressure testing was "not conducive to a thorough and deliberate visual inspection."

1/ See Regulatory Analysis, Revision 2 to Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, dated November 20, 1987, at 24.

2/ Letter from J.H. Sniezek, NRC, to J.P. O'Reilly, Georgia Power Company, dated April 10, 1987.

The April 10, 1987 letter was based on a Staff interpretation set forth in a letter to the licensee dated May 5, 1986. It was in the May 5, 1986 correspondence that the Staff first set forth its position on the issue. In that letter, the Staff noted that it had permitted the licensee to perform nuclear pressure testing "for a number of years" and recognized that the Staff position was new. In particular, the Staff, in concluding that the licensee could continue such testing on a one time basis, cited the following reasons, among others: (1) "the past history of this activity where the Staff has permitted [the licensee] to perform these tests using nuclear heat"; (2) "the late arrival of the staff position"; (3) the Staff's "conclusion that there is minimal difference in the safety afforded by the testing as performed by [the licensee] using nuclear heat versus testing in accordance with the staff position on the code requirements"; and (4) the Staff's "desire to consider new information that may have a bearing on this recent staff position . . . ."

NUBARG believes that the Staff position prohibiting nuclear hydrostatic and leakage testing should be reconsidered. Accordingly, we respectfully request that the EDO review and modify the Staff's denial.

DISCUSSION

NUBARG's position is based on (1) the provisions of the Commission's regulations in 10 C.F.R. Part 50, Appendix G, and (2) the provisions of ASME Code Section XI.

A. Appendix G

System pressure tests (leakage and hydrostatic tests)[3] are conducted in accordance with Section XI of the ASME Code. Relevant portions of Section XI have been incorporated by reference into 10 C.F.R. § 50.55a as part of the NRC's inservice inspection requirements. Criteria for conducting the tests, including temperature and pressure limits, are prescribed by Appendix G to 10 C.F.R. Part 50.

Appendix G clearly permits hydrostatic and leakage testing with the core critical. Section IV.A.5 of Appendix G expressly states (emphasis added):

3/ Throughout this appeal we refer to "hydrostatic" and "leakage" testing together. For the sake of simplicity, we treat requirements for leakage testing as essentially the same as those for hydrostatic testing, even though different test pressures are required and other differences exist.

> If there is fuel in the reactor during system hydrostatic pressure tests or leak tests, the requirements of paragraphs 2 or 3 of this section apply, depending on whether the core is critical during the test.[4]

This provision was adopted as part of the revisions to Appendix G adopted in 1983. 48 Fed. Reg. 24008 (May 17, 1983). Those revisions included an explicit exception from the temperature limits of Section IV.A.3 for BWRs. Significantly, the NRC revised Section IV.A.3 to permit BWRs to conduct core critical operations at vessel temperatures below the minimum permissible temperature for hydrostatic testing, provided the vessel water level is within the normal range for power operation. Section IV.A.3 thus provides:

> When the core is critical (other than for the purpose of low-level physics tests), the temperature of the reactor vessel must not be lower than 40°F (22°C) above the minimum permissible temperature of paragraph 2 of this section nor lower than the minimum permissible temperature for the inservice system hydrostatic pressure test. An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure.

This change was made on the basis of the 1978 GE Topical Report NEDO-21778-A. The purpose of that Topical Report was to justify changing the minimum temperature limits for core criticality for BWRs. In accepting the Topical Report and agreeing that the requested revision to Appendix G was desirable, the Staff was fully aware that this would permit hydrostatic testing after core criticality, a procedure which the Staff apparently was not immediately ready to accept due to concerns about the possibility of a control rod drop accident ("CRDA") with the vessel water solid, which was the bounding accident scenario. In its 1978 Evaluation of the GE Topical Report, the Staff stated:

> If the criticality limit is modified as requested, it is possible that the reactor could be taken critical to warm up the

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4/ Paragraph 2 defines acceptable temperature limits when the core is not critical, and paragraph 3 defines these limits when the core is critical.

> vessel for a hydrotest. To further reduce the possibility of a CRDA while the vessel is water solid for a hydrotest, it will be necessary to add a requirement to Appendix G that all control rods must be fully inserted during hydrotest.[5]

The Staff indicated that it was considering amending Section IV of Appendix G to state explicitly: "All control rods shall be fully inserted during hydrotests."

Significantly, however, no such restriction was contained in either the 1980 proposed revisions to Appendix G[6] or the final revisions published in 1983.[7] Instead, the NRC adopted the provision in Section IV.A.3 that requires the "water level [to be] within the normal range for power operation . . . ." This restriction would have alleviated the Staff concern with a control rod drop accident while the vessel was water solid. With the vessel water level within the normal range, the effect of a pressure spike from a control rod drop accident would not present a significant concern, due to the presence of steam and noncondensibles in the system.

At the same time the NRC adopted the provision of Section IV.A.5 that permits core criticality during hydrostatic and leak tests. It can be inferred, therefore, that the NRC concluded that hydrostatic and leakage tests could properly be conducted with the core critical so long as the vessel water level was in the normal range.

In its August 17, 1988 denial, the Staff dismissed the plain language of Section IV.A.5 with the following statement (at page 3): "The phrase 'depending on whether the core is critical during the test' in Appendix G does acknowledge the use of nuclear heat, but only when special circumstances arise, such as for Hatch 1 on a one-time basis." No support is cited for this limitation on the otherwise clear language of Section IV.A.5. This interpretation should therefore not be relied upon as a basis for denial of the claim.[8]

5/ Topical Report Evaluation, dated January 1978 (emphasis added).

6/ 45 Fed. Reg. 75536 (1980).

7/ 48 Fed. Reg. 24008 (1983).

8/ In its denial, the Staff also stated (emphasis added): "SRP Section 5.3.2 [which implements Appendix G] permits lower safety margins during hydrostatic and leakage testing than during core critical operations, thereby implying that the

[Footnote 8 continued on following page.]

B. ASME Code Section XI

The use of nuclear pressure testing is consistent with Section XI of the ASME Code. On at least two occasions the ASME Boiler and Pressure Vessel Committee has issued written interpretations of Code requirements in this area. By letter dated February 11, 1987, the ASME Committee responded to an inquiry as to whether Section XI requires the reactor to be in a noncritical state during pressure tests (hydrostatic and leak tests).[9] The ASME Committee stated that "[c]ore criticality during pressure testing is not addressed by Section XI, Division 1." In an earlier response, dated September 18, 1986, the ASME Committee agreed that the Code permits the use as a pressurizing medium of "a mixture of steam, water, and non-condensible gases in a proportion no greater than that present during normal startup . . . ."[10] This was consistent with an earlier interpretation,[11] and indicates that the use of nuclear heat for testing while the vessel water level is in the normal range for power operation is not prohibited by the Code.

In its August 17, 1983 denial, the Staff indicated that because Table IWB-2500-1 provides that system leakage tests are to be conducted "prior to plant startup following each reactor refueling outage," and because typical BWR Technical Specifications and FSAR provisions define the plant condition prior to startup as hot shutdown with all control rods inserted, the testing may not be done with the core critical. Given the ASME's ruling that Section XI does not address core criticality, we believe the Staff has given too narrow an interpretation to the provision in IWB-2500-1.

In view of the ASME's ruling, it appears that the Code's use of the term "startup" was not intended to coincide with

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[Footnote 8 continued from previous page.]
core will not be critical during the testing." It appears, however, that the implication drawn by the Staff does not necessarily hold. Section 5.3.2 does not by its terms preclude testing with the plant at low power if the higher safety margins are used. In fact, Appendix G does just that. Under Section IV.A.5, hydrostatic and leakage testing may be conducted when the core is critical, provided that the higher safety margins of Section IV.A.3 of Appendix G are used.

9/ Letter from S. Wienman, ASME, to L.T. Gucwa, Georgia Power Company, dated February 11, 1987.

10/ Letter from S. Weinman, ASME, to L.T. Gucwa, Georgia Power Company, dated September 18, 1986.

11/ Interpretation XI-1-83-25, dated October 27, 1983.

definitions used in plant Technical Specifications. Terminology used by the American Society of Mechanical Engineers cannot and should not necessarily be equated with specific terms contained in Technical Specifications. A more reasonable view of the intent of IWB-2500-1 is that it is meant simply to prescribe those tests to be performed to verify system integrity before the system is placed in an unrestricted mode of operation within its designed temperature and pressure limits.

It should also be noted that conducting system pressure tests with the plant at low power is consistent with other testing requirements. The environmental conditions that the Staff has cited as not being conducive to a thorough and deliberate visual inspection also exist for other types of inspections. Numerous administrative and personnel safety measures have been implemented to deal with these conditions. In addition, leak detection instrumentation may also be available. Any concerns with inspection requirements that could expose personnel to adverse environmental conditions should be addressed directly and in a comprehensive manner.

CONCLUSION

For the foregoing reasons, NUBARG believes the Staff's position prohibiting nuclear hydrostatic and leakage testing should be withdrawn. Given the provisions of the NRC's own regulations and the fact that the NRC itself has recognized that there is a minimal difference in safety between testing with nuclear heat and pump heat, it appears that such a position is not justified. Furthermore, as a matter of policy, the Staff should not preclude a testing method that could allow testing to be carried out at higher system temperatures capable of meeting the PTS limits of Revision 2 to Regulatory Guide 1.99 without the substantial plant modifications that otherwise might be necessary.

We would be pleased to discuss this subject with you should you have any questions.

Sincerely,

Daniel F. Stenger

Nicholas S. Reynolds
Daniel F. Stenger
Counsel to Nuclear Utility
Backfitting and Reform Group

cc: Lawrence C. Shao

BWR OWNERS' GROUP

Stephen D. Floyd, Chairman
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BWROG-8962
August 11, 1989

Mr. James Taylor
Acting Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subj: Appeal From Staff Decision Requiring Total Equipment Diversity Under ATWS Rule (10 C.F.R. § 50.62)

Dear Mr. Taylor:

Attached is the appeal of the BWR Owners' Group (BWROG) regarding Staff action on the use of Rosemount transmitter trip units as they relate to the ATWS Rule. There exists a difference of opinion with the Staff on the subject of what constitutes a sensor and what kind of diversity, if any, should be applied to the trip unit portions of the alternate rod injection system level and pressure sensors.

The Rule requires alternate rod injection system diversity, from sensor output to the final actuation device. The currently installed alternate rod injection and reactor trip system level and pressure sensors each comprise a transmitter plus a remotely located trip unit. Were it not for the separation of the trip unit from the transmitter, the transmitter/trip unit would be a sensor (within the meaning of the Rule) according to the Staff, and would be exempt from the diversity requirement of the Rule. Because of the perception that a transmitter/trip unit is not a sensor, the Staff is requiring the level and pressure trip units of the alternate rod injection system to be manufactured by an alternate supplier, i.e., they are requiring equipment diversity. However, this is inconsistent; the portion of the ATWS Rule in question focuses on the potential for common cause failure. The trip unit and transmitter are connected by a passive device (wiring) which is not a common cause failure concern. Moreover, even if the remote location of the trip unit were a source of common mode failure,

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equipment diversity of the trip unit would not address location-based concerns.

Regardless, if the ATWS Rule is applied to the trip units, the Rule itself calls for diversity (the Rule does not indicate any specific type of diversity). There are many ways to provide diversity, including, among others, equipment, functional and application (energization state) diversity. BWR owners have provided such diversity in all active components of their alternate rod injection systems, including the level and pressure trip units. Functional diversity and diversity by application are provided for the level and pressure trip units. We think the outright rejection of these acceptable forms of diversity is again inconsistent with the intent of the Rule. A common mode failure of the trip units must result in a reactor trip because the trip units are identical but have opposite energization states during operation. (Alternate rod injection would trip the reactor if a common mode failure caused the output of the trip unit to energize. However, if a common mode failure caused the output of the trip unit to deenergize, the reactor protective system would trip the reactor.) The Staff has determined quite the opposite--that a common mode failure must result in a failure to trip because the units are identical. This conclusion is erroneous.

The Staff's position of equipment diversity stems from <u>guidance</u> in the Statement of Considerations published with the Rule which states that the preferred form of diversity is equipment diversity which is to be provided <u>where reasonable and practicable</u>. Aside from requiring diversity where none is required, the current Staff position requiring equipment diversity in this case is unreasonable. The maximum possible benefit to be gained by installing diverse trip units is negligible, but the cost is substantial and carries with it the unmeasured but significant risk of increasing maintenance-related common mode failures. We are concerned that the Staff is requiring equipment diversity only for the sake of diversity, in spite of the lack of safety benefit. The ACRS shares this concern. See Attachment 1.

We have raised the diversity issue on two occasions with Dr. Murley, who after due consideration, rejected our position. While we have great respect for Dr. Murley's technical expertise, we think his conclusions on this issue are inconsistent with the Rule and the prior Staff positions supporting the Rule. In fact, the current Staff position has the potential to increase common cause failures, thereby defeating the purpose of the Rule. The ACRS also shares this concern.

We have attached a detailed analysis and history of this issue (Attachment 2) for your consideration and we request that you review our position. Staff acceptance of the diversity currently provided would allow the BWR owners to avoid unnecessary modifications to the alternate rod injection system. We also believe that the technical input of the ACRS is extremely useful and we encourage you to study their letter. See Attachment 1.

We further request that the Commission be asked to send the question of transmitter trip unit diversity to the ACRS for resolution. Specifically:

1. Are the ARI and RTS Rosemount transmitter/trip units sensors within the meaning of the ATWS Rule, and if so, are they subject to the diversity requirement of the Rule?

2. If the ARI and RTS Rosemount trip units are subject to the diversity requirement of the ATWS Rule, is the use of diversity of application (energization state diversity) in the trip units sufficient, when combined with the equipment and functional diversity of ARI and RTS systems, to meet the diversity requirement of the Rule?

We believe the Staff has reached an inappropriate conclusion on this issue. The NRC has already addressed the technical and safety questions involved with sensor diversity and has provided very clear guidance in the ATWS Rule, the Statement of Considerations, and the Policy Statement on Diversity. We do not question that guidance. At issue here is whether the current Staff position on sensor diversity conforms to that guidance.

The comments/positions provided in this letter have been endorsed by a substantial number of the members of the BWROG. However, this letter should not be interpreted as a commitment of any individual member to a specific course of action. Each member must formally endorse the BWROG position for that position to become the member's position.

Thank you for your attention to this matter. We look forward to your response.

Sincerely yours,

Stephen D. Floyd

Stephen D. Floyd, Chairman
BWR Owners' Group

SDF/

Attachment

cc: BWROG Executive Oversight Committee
BWROG Primary Representatives
G.J. Beck, BWROG Vice Chairman
R.F. Janecek, RRG Chairman
S.J. Stark (GE)
G. Samstad (GE)
Dr. T. Murley (USNRC)
F.J. Miraglia (USNRC)
T. Price (NUMARC)

ATTACHMENT 2

Appeal Of Staff Decision Concerning the Diversity Requirement of the ATWS Rule (10 C.F.R. § 50.62)

I. INTRODUCTION

This letter is an appeal of a Staff decision regarding the extent to which Rosemount level and pressure transmitter/trip units installed in the alternate rod injection (ARI) system and the reactor trip system (RTS) need to be diverse pursuant to the ATWS Rule (10 C.F.R. § 50.62).

The issue initially was joined on the Carolina Power & Light (CP&L) docket when it requested reconsideration of a Staff decision[1] requiring complete equipment diversity of the water level transmitter/trip units installed in the ARI system and the RTS.[2] The BWR Owners' Group (BWROG) supported CP&L's appeal;[3]

1/ Safety Evaluation By The Office Of Nuclear Reactor Regulation Related To Amendment No. 150 To Facility Operating License No. DPR. 62 Carolina Power & Light Co. et al. Brunswick Steam Electric Plant, Unit 2 Docket No. 50-324 (Apr. 8, 1988).

2/ Letter from L.W. Eury to Thomas E. Murley (May 11, 1988).

3/ Letter from D.N. Grace to Thomas E. Murley (Jun. 28, 1988).

however, on August 8, 1988, the appeal was denied.[4] Subsequently, the BWROG again appealed the Staff decision on the basis of further information, but this appeal also was denied.[5]

In each denial, the Staff maintained the position that the water level and pressure transmitter/trip units in the ARI system and the RTS required equipment diversity. The BWR Owners' Group finds this answer completely inconsistent with the ATWS Rule and its associated guidance. In summary, the ATWS Rule does not require water level or pressure transmitter/trip units to be diverse. These types of units were recognized by the Staff during the ATWS rulemaking as being sufficiently reliable as to be excluded from the Rule. Moreover, if diversity is required, the ATWS Rule does not specify any particular type of diversity; rather, the various types of diversity recognized by the Staff to be present in the ARI system and the RTS suffice. Lastly, to reach a "requirement" for equipment diversity the Staff must resort to the Statement of Considerations accompanying the ATWS Rule. That Statement, in its "Guidance" section, refers to equipment diversity "to the extent reasonable and practicable." In view of the insignificant decrease in risk resulting from equipment diversity and in light of the cost involved, the Staff

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4/ Letter from Thomas E. Murley to Lynn W. Eury (Aug. 8, 1988).

5/ Letter from Ashok C. Thadani to Donald N. Grace (Mar. 17, 1989).

decision requiring equipment diversity in the water level and pressure transmitter/trip units is not reasonable.

II. BACKGROUND

The ATWS Rule and BWR Compliance.

The ATWS Rule, in 10 C.F.R. part 50.62(b)(3) requires that:

> 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 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.

In compliance with the above Rule, BWR licensees have installed diverse and independent ARI systems. Diversity from the RTS is achieved throughout the ARI system by combinations of allowable methods of diversity such as functional diversity, diverse hardware and by diversity of application (energize to trip versus deenergize to trip). Equipment diversity is provided where reasonable in the ARI by using components fabricated by different manufacturers. Functional diversity is provided within the RPS by having several different parameters, i.e., level, pressure, valve position, and neutron flux for the most likely conditions that could lead to a scram. Diversity by application is provided by designing the ARI to generate a scram signal when the level or pressure bistable is energized, whereas the RTS generates a scram signal when the level or pressure bistable is deenergized.

Most BWRs installed sensors utilizing trip units from a single manufacturer (Rosemount transmitters with either Rosemount or Foxboro trip units)[6] in both the RTS and ARI system.[7] The Staff seeks to have circuit boards manufactured by another entity inserted in the pressure and level sensors of either the ARI System or the RTS thereby achieving equipment diversity.[8]

6/ The issue addressed by this appeal is not affected by whether Rosemount or Foxboro trip units were selected for the two systems, therefore, only Rosemount trip units will be mentioned unless the issue is different for the Foxboro trip units.

7/ Rosemount or Foxboro trip units were used exclusively in both the RTS and ARI system because of (1) the operational advantages of the sensors over the Barton sensors, (2) encouragement from the Staff to implement the ARI design in spite of the diversity question because of the "clear safety benefit even with the Rosemount ATTUs," (Letter from Thomas E. Murley to Lynn W. Eury, Aug. 8, 1988), (3) the initial acceptance by the Staff of the same sensor configuration in the RTS and ARI system at Monticello (which was assumed to be the BWR precedent), (4) the statement in the Rule excluding sensors from diversity, and (5) the benefits to be derived from standardization of similar (highly reliable) components, not the least of which is reduction of common cause failures.

8/ Rosemount transmitter/trip units employ a pressure transmitter hydraulically connected to the primary system. Pressure action on the transmitter's transducer generates an electrical signal proportional to pressure (or differential pressure for a level transmitter) which is coupled to a remotely located trip unit circuit board. The circuit board generates a bistable signal as a function of the magnitude of the transmitter electrical signal. The output of the trip unit is the pressure or level input signal to the RTS or ARI system. The ATWS Task Force, when recommending excluding sensors from the reach of the Rule, analyzed sensors that

(Footnote 8 continued on next page.)

Installation of standard equipment can, in cases where the equipment is highly reliable, reduce the probability of commmon cause failures. When equipment is standardized, technicians are more skillful at calibration and repair. Non-standard trip units require additional, similar spare parts, hence, the probability of installing incorrect parts increases.[9] When diverse components are similar or identical in appearance, the probability of following the wrong calibration and maintenance procedures increases. These drawbacks can lead to increased common cause failures. Furthermore, when standard equipment is installed, training, spare parts and administrative costs can be minimized.

In the present case, the NRC-required diverse trip units will be produced in one batch or a small number of batches. While the BWROG believes these trip units will perform reliably, insufficient production time will exist to develop a closed feedback loop of quality improvements driven by field proven

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(Footnote 8 from previous page.)
differed from the Rosemount sensor by combining the transmitter (usually a Bourdon tube) and bistable (an iron core transformer device) into a single housing.

9/ "If you take a look at the proposed GE ATTU cards, one in your left hand and one in your right hand, they will be identical cards." Staff comment on the differences in physical appearance between the existing Rosemount trip unit and the trip unit being required by the current Staff decision, transcript pp. 32 of the ACRS Subcommittee meeting on Instrumentation and Control Systems (Apr. 21, 1989).

performance. Standardization is, in our opinion, both safer and more economical when applied to such highly reliable and frequently tested equipment such as Rosemount or Foxboro trip units.

III. ARGUMENT

A. The ATWS Rule does not apply to the Rosemount Transmitter/Trip Units

The ATWS Rule clearly acknowledges that devices upstream of the sensor output are excluded from the reach of the Rule. The subject circuit boards in the Rosemount transmitter/trip units are upstream of the sensor output and accordingly, the Staff's decision to require equipment diversity (or for that matter, any diversity) is inconsistent with the Rule.

To explain, the sensor portion of the RTS or ARI system consists of a transmitter unit and a bistable trip. The Staff has concluded that the sensor portion of these systems is sufficiently reliable and subject to such intense surveillance as to not require diversity.

> The trip portion of the sensor system consists of bistables that signal an out-of-tolerance condition. This portion of the system is vulnerable to bistable calibration errors and like component common cause failures. However, continuous monitoring of the sensor output, and the frequent testing of the trip values provide a good chance of discovery of such common cause problems. . . . Though differences exist in the level of redundancy and logic structure, these only influence the

> independent failure contribution which does not contribute significantly to the overall RPS unavailability. Therefore, for the purposes of this analysis, the sensor portion of the RTS will be ignored.[10]

The high bistable reliability (10^{-7} per demand) coupled with "the large number of sensed parameters [leads one to judge] that the controlling common cause problems would probably not reside in the sensor subsystem . . . ."[11]

In the Rosemount trip unit the circuit board is the bistable electronic element in the level or pressure sensor.[12] It is recognized by the Staff to be reliable; it has a proven history of good performance and is thus excluded from the reach of the Rule. To overcome this exclusion, the Staff focuses on the fact that in the Rosemount design, the trip unit (solid state bistable) is located remotel from the transmitter. The bistable of the sensor analyzed by the ATWS Task Force was mounted in the same enclosure as the pressure transmitter. The Staff considers the separation of the Rosemount transmitter and trip unit to

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10/ Memorandum from William J. Dirks to the Commissioners, SECY-83-293 (Jul. 19, 1983), Enclosure "D," Recommendation of the ATWS Task Force at A10-A11.

11/ *Id*. at A11.

12/ See Staff Statement, Transcript pp. 117 of the ACRS Subcommittee meeting on Instrumentation and Control Systems (Apr. 21, 1989). See also pp. 38, lines 6-7. The statement should be corrected from "is stable" to "which is the bistable."

disqualify the Rosemount sensor from the exemption provided by the Rule from diversity for sensors.

The separation does not alter the characteristics or the reliability of the device, nor does the introduction of wires increase common cause failure probability.[13] Wires are passive devices, and only active devices must be diverse.[14] In fact, the solid state trip units are more reliable than the older electromagnetic, sliding core-type, and are subject to monitoring and surveillance at least as often as the bistable devices analyzed by the ATWS Task Force. From a reliability and testing viewpoint, the Rosemount transmitter/trip units meet the conditions that were the basis for the exemption from sensor diversity set forth in the Rule and should, therefore, be exempt from diversity. Reinforcing this position is the Staff statement that the Rosemount transmitter/trip units would not need to be diverse if the trip unit were integrally mounted with the transmitter.[15] Based on this statement, we conclude that the Staff believes that the location of the trip unit remote from the transmitter must be a source of potential common cause failures, in which case it makes no sense to reduce these remote-location

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13/ The Staff agrees that distance is not an ATWS diversity concern. Id. at 117.

14/ Id. at 40.

15/ Id. at pp. 132 to 134.

common cause failures by installing diverse trip unit boards in ARI.

B. Even if it is determined that the ATWS Rule applies to the Rosemount transmitter/trip units, these units meet the Rule.

As noted, the Rule requires diversity "from sensor output to the final actuation device." Contrary to the position of the Staff, the Rule does not specify the type of diversity; it simply requires diversity. As noted in Section II, diversity from the RTS is achieved throughout the ARI system by combinations of allowable methods of diversity.[16] The ARI system employs equipment, functional and application state diversity and thus complies with the ATWS Rule.

To explain, equipment diversity is provided, for example, by diverse logic relays, contactors and scram air header vent valve designs. Functional diversity is provided within the RTS by employing diverse trip channels, including for example, turbine stop valve closure, high neutron flux and low water level/high pressure.[17] At least two paths to provide a scram

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16/ See IEEE Standard 352-1987, General Principles of Reliability Analysis of Nuclear Power Generating Station Safety Systems, Table A8, which provides a number of diversity alternatives, including functional and energization state diversity, which are used in the ARI system.

17/ For a list of channels, see letter from D.N. Grace to Thomas E. Murley (Jun. 28, 1988).

signal are available to respond to all expected ATWS events. The Staff has rejected the diversity by application provided for the loss of feedwater event; however, diverse level indication and sufficient time exists for the operator to initiate a manual scram, should six trip units fail simultaneously. Diversity by application is provided in the Rosemount trip units by designing the ARI to sense a level trip condition when the trip unit energizes, versus the RTS which senses a level trip condition when the trip unit deenergizes. Active components are the only components that need to be diverse, therefore, the trip units employ full diversity by application because the bistable element is the only active element on the trip unit during normal system operation.[18], [19] Other active components in the trip units are only used during calibration and testing.

18/ The Staff agrees that if all active components on the card are in a different state, diversity is achieved. See Transcript pp. 40 of the ACRS Subcommittee meeting on Instrumentation and Control Systems (Apr. 21, 1989).

19/ A letter from Thomas E. Murley to Lynn W. Eury, Aug. 8, 1988, states, "Since both boards [ARI and RTS] are made by Rosemount and are virtually identical, we conclude that the application of different energization states is not sufficient to minimize the potential for common cause failures in the ARI and RTS ATTUs." This statement is a non sequitur. If the trip units are virtually identical and the only active component (the bistable element) fails by a common mode, the trip units will either fail energized or fail deenergized. Regardless of which failure mode occurs, one of the scram systems will receive a trip signal. Furthermore, after stating that the potential for common cause failures is not minimized, the letter goes on to state that "the ARI system . . . provides a diverse logic design which addresses the major contributors to common cause failure in the RTS . . . ."

C. If the term "diversity" is more broadly construed to require "equipment diversity", such construction should be read as "equipment diversity to the extent reasonable and practicable"

As noted in Section III (B) above, the Rule itself does not impose a limitation on diversity so as to require that all diversity be equipment diversity. Rather, the Staff's support for equipment diversity comes from "Guidance" set forth in the Statement of Considerations.[20]

We recognize that language which is not incorporated in the Rule per se can be viewed as part of the Rule. In Automotive Parts & Accessories Ass'n v. Boyd, 407 F.2d 330, 338 (D.C. Cir. 1968), the Court considered "the statement in the text of the promulgation of the [s]tandard" (in this case, a rule issued by the Department of Transportation) to be "'a concise general statement' which passes muster under Section 4 of the APA." Further, in Home Box Office, Inc. v. F.C.C., 567 F.2d 9, 35 (D.C. Cir.), cert. denied, 434 U.S. 829, reh'g denied, 434 U.S. 988 (1977), the Court stated that the concise general statement of the basis and purpose of the rule ultimately adopted is "intended to assist judicial review as well as to provide fair treatment for persons affected by a rule."

20/ See 49 F.R. 26042 (1984).

It is not clear whether specifically delineated "Guidance" qualifies under this case law as inclusion to and thus part of the Rule. Rather "Guidance" could well be viewed simply as that -- guidance. The Atomic Energy and Administrative Procedures Acts empower the NRC to impose requirements on licensees by means of rules and orders. The Statement of Considerations is neither rule nor order, and as stated in Home Box Office, Inc., it is intended to assist judicial review and provide fair treatment. The Statement is in the same class of guidance as, for example, Regulatory Guides and Generic Communications. Therefore, it is well-recognized that guidance is non-binding on a licensee.

However, we need not reach this point. The language of interest regarding "equipment diversity" is not unqualified. Rather, the full statement, in the context of the matter at hand, is:

> 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."[21]

Simply put, if the "equipment diversity" language is viewed as a requirement, the "reasonable and practicable" language must also apply such that the requirement would be "equipment

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21/ Id.

diversity to the extent reasonable and practicable." If on the other hand, the language is viewed as guidance, it is non-binding, and other means of diversity are appropriate. A "guidance only" conclusion would dispose of the issue (since, as noted, the BWR's have diversity), therefore, the remaining focus of this argument is on the "requirement" conclusion.

The Staff acknowledges that "equipment diversity" must be read in the light "to the extent reasonabla and practicable." In the initial determination of this matter on the Carolina Power & Light Company Brunswick Plant docket, the Staff provided in Appendix 1 its "Position on Diversity Requirements" which provides:

> It is recognized that total/absolute component/hardware diversity can be difficult and sometimes impossible to achieve. For these instances, [an] acceptable level of component/hardware diversity can be achieved in accordance with combinations of allowable methods such as energization states, AC versus DC power, functional capability, and the use of components from different manufacturers.[22]

22/ The Staff Position On Diversity Requirements, Appendix 1 to the Safety Evaluation By The Office of Nuclear Reactor Regulation Related To Amendment No. 150 To Facility Operating License No. DPR-62, Carolina Power & Light Company et al., Brunswick Steam Electric Plant, Unit 2, Docket No. 50-324 (Apr. 8, 1988).

Several factors underscore our position. First, precedent; and second, the imperceptible reduction in risk associated with a not imperceptible increase in cost.

Northern States Power was the first utility to equip a BWR (Monticello) with a diverse alternate rod injection system. The Monticello ARI and RTS both used Rosemount-supplied analog transmitter/trip units for detecting reactor water level and relied on diversity of application of the trip units--one system energized to trip, the other system deenergized to trip. In keeping with published Staff policy, the initial Monticello SER acknowledged the reasonable and practicable basis for equipment diversity, stating:

> The NRC guidance on the ATWS Rule states that 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.[23]

and on the subject of ARI functional diversity, further adds:

> "that the diversity between the ATWS logic and the reactor trip system (RTS) logic [at Monticello] has been achieved primarily through the functional

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23/ Safety Evaluation By The Office of Nuclear Reactor Regulation, Northern States Power Company, Monticello Nuclear Generating Plant, Docket No. 50-263, Compliance With ATWS Rule, 10 C.F.R. part 50.62, Relating To Alternate Rod Injection And Recirculating Pumps Trip Systems (Dec. 21, 1987) (Emphasis added).

> application of the logic elements and the location of the logic elements. The ARI system will be energized-to-function instead of deenergized-to-function for the RTS. The ARI system will use DC power instead of AC power for the RTS. The ARI initiation logic (two-out-of-two) will be diverse from the RTS logic (one-out-of-two-twice). To the extent reasonable and practicable for ATWS Rule implementation, the staff finds the degree of diversity within the Monticello design acceptable."[24]

Our position is that the initial Monticello logic applies with equal force to our appeal. The reasonable and practicable requirement in the Statement of Considerations is part of the ATWS Rule and empowers the Staff to exercise discretion to accept "combinations of allowable methods" of diversity when total diversity is "difficult . . . to achieve."

With regard to the cost/benefit equation, there are no alternative qualified and highly reliable pressure and level transmitter/trip units available for installation in the RTS and ARI system other than Foxboro units which would cost approximatel $800,000 per plant to install.[25] The only alternate trip unit for Rosemount trip units is an essentially identical unit supplied by General Electric that is not readily

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24/ Id.

25/ Plants with Foxboro instruments would need to install Rosemount sensors at a cost of about $800,000 because no diverse supplier of Foxboro transmitter/trip units is obtainable.

available, but must be manufactured in batches. Procurement of these diverse trip unit circuit cards would cost about $8,000 to $12,000 per trip unit.[26] The cost per plant would be roughly $170,000 (including design modification reviews, drawing changes, maintenance training, procedures, etc.) to install diverse trip unit cards in ARI. In addition, environmental qualification of the diverse cards (required in some cases) will cost approximately $200,000 more.

The Staff has adhered to an "equipment diversity at any cost" policy, and has thus far not considered any discussion of the remoteness of a common cause failure in Rosemount transmitter/trip units employing energization state diversity. Thus, the Staff is not considering the cost in relation to the resulting risk reduction, which is almost negligible.

According to studies conducted by the Staff, the probability of core melt from an ATWS event, assuming no installed ARI, recirculation pump trip (RPT) or automatic (86 gallons per minute) standby liquid control system (SLC) is 5.3×10^{-5} per

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26/ Diverse analog transmitter/trip units were stated to cost $8,000 to $12,000/unit in a letter from Ashok Thadani to Donald N. Grace (Mar. 17, 1989). The unit cost referred to is per circuit card, not per power station which is also commonly referred to as a unit.

reactor year.[27] By addition of an ARI and an SLC $3.5 million,[28] the ATWS contribution to core mel reduced to 1.2×10^{-5}. This equates to a cost of each 10^{-5} reduction in ATWS probability, compared $3 million for each 10^{-5} reduction, giving a favo Value/Impact ratio of 3.5.[29]

Based on a study of the Brunswick plant by C Light, the ATWS contribution to core damage proba plant having an ARI, conservatively assuming 20% are common mode between ARI and RTS, is 1.02×10 contribution to core damage probability, assuming between ARI and RTS (0% common mode failures), is Thus, total diversity reduces the ATWS contributi melt probability by about 2%, at a cost of $8.5 m 10^{-5} reduction.[30] Approaching the problem in th ATWS Task Force, the cost of an ATWS is assumed t

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27/ Memorandum from William J. Dirks to the Commis 83-293 (Jul. 19, 1983), Enclosure "D," Recomme ATWS Task Force, at 22.

28/ Id. at 38.

29/ Id. at 32, 46.

30/ A 0% probability of common mode failures is ar approached and represents the best possible reality, the actual reduction in common mod probability resulting from diverse trip uni approach this goal. Moreover, because of t similarity of the diverse trip units, poten failures resulting from inadvertent mainten further detract from this ideal assumption. calculation, therefore, represents a best c and the plant will operate for 30 years.[31] The maximum potential value of the change in ATWS probability from the added diversity in Rosemount transmitters is $(1.02 \times 10^{-5} - 1.00 \times 10^{-5}) \times 10^{10} \times 30$ = \$60,000. The impact is \$170,000, therefore, the value to impact ratio is only about 0.35. This is well below the level considered to be cost effective by the Staff.

By either measure, the cost per increment of probability reduction is far in excess of the overall cost of complying with the ATWS Rule. Requiring a licensee to make this large expenditure, in light of the extremely small maximum potential reduction (2×10^{-7}) of risk, is unreasonable. In fact, the CP&L study is conservative because it calculated the risk reduction in achieving complete diversity of all components. The General Electric Company performed a more specific assessment of the likelihood of having the required six trip unit failures result in a failure to cause a scram signal from either the RPS or ARI. The study calculates a failure to scram of 1×10^{-8}.[32]

In sum, in view of the small benefit to be derived from equipment diversity at a comparatively high cost, it is

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31/ Memorandum from William J. Dirks to the Commissioners, SECY-83-293 (Jul. 19, 1983), Enclosure "D," Recommendation of the ATWS Task Force, at 31.

32/ General Electric Report No. EAS 90-1288, "Reliability Assessment of Anticipated Transients Without Scram For Loss of Feedwater Event" (Dec. 2, 1988).

unreasonable for the Staff to insist on complete equipment diversity. Rather, other types of diversity are appropriate.

Alternatively, if it can be shown that total equipment diversity would be difficult, then combinations of diversity would be permissible. The Staff has acknowledged this point.

In a letter to the BWR Owners' Group denying the appeals, Dr. Murley stated:

> "The diversity required by the ATWS rule is intended to ensure that common mode failures which disable the electrical portion of the existing reactor trip system will not affect the capability of the ARI system to perform its design functions. It is recognized that total component/hardware diversity can be difficult to achieve, however licensees are encouraged to provide a maximum effort to satisfy the diversity requirements.[33]

Dr. Murley recognizes that total Rosemount transmitter/trip unit component/hardware diversity "can be difficult to achieve."[34] Under such a circumstance, the Staff Position on Diversity Requirements becomes important, expressly its allowance of "combinations of allowable methods" of diversity when total diversity is "difficult . . . to achieve." As discussed in

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33/ Letter from Thomas E. Murley to Donald N. Grace (Aug. 8, 1988).

34/ Id. [emphasis added]

Sections II and III (B), the ARI system possesses such combinations of diversity.[35] The Rosemount transmitter/trip unit, as a sensor, is exempted from the ATWS Rule. Alternatively, the ARI system possesses adequate diversity to satisfy the Rule's reference to "diversity." Lastly, equipment diversity, (if diversity is required at all) is required to the extent reasonable and practicable. The combinations of diversity satisfy this Staff "guidance" position.

IV. CONCLUSION

The Staff has summarily rejected all attempts by the BWR owners to justify combinations of diversity by means of unfavorable cost/benefit analyses or negligible risk reduction arguments. Of itself, this Staff decision is unreasonable in light of the Staff's policy of rejecting proposed rules having

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35/ It is interesting to note that the Staff in stating that it required diversity for active, i.e., components having different states, it also stated that energization state diversity was provided only for the trip bistable on the Rosemount trip unit which, therefore, ignored diversity for the remaining components on the trip unit. Thus, in the opinion of the Staff, energization state diversity as applied to the trip unit as a whole did not meet the requirements of the Rule. However, the trip bistable is the only active component on the trip unit (other than components used only during calibration and testing), and by the Staff's position, only the trip bistable needs to be diverse, which it is, using one of the Staff's allowable methods of diversity: energization state diversity (diversity of application).

cost/benefits of substantially less than 1.[36] Moreover, the Staff has expressly recommended that functional and energization state diversity of sensors is acceptable in exactly these circumstances and initially accepted this level of diversity in the past, at Monticello.

We believe a meeting with you at your convenience would be beneficial for us to convey our concerns with, and to help us understand, the Staff's current position on diversity under the Rule. However, if our appeal is not persuasive, we believe an exemption request is appropriate under the circumstances, even though the Staff does not recommend this option.[37] Under 10 C.F.R. Section 50.12(a)(2) an exemption is proper when:

1. Application of the regulation under these circumstances is not necessary to achieve the underlying purpose of the Rule. In the present instance there is no credible common mode of failure of transmitter/trip units in the RTS and ARI when diversity of application is utilized. Even when common failure modes are assumed to exist, only an extremely small maximum potential reduction in risk (2×10^{-7} per demand) conservatively results from eliminating these common failure modes. The Staff

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36/ See, e.g., SECY 83-293, pages 2, 31, and 48 where the Commission rejects a requirement for additional safety valves.

37/ Letter from Thomas E. Murley to Lynn W. Eury (Aug. 8, 1988).

"continue[s] to believe that such numerical estimates of common mode failure likelihood are questionable,"[38] yet the Rule is justified by the Staff based on similar numerical estimates. Therefore, we assume that risk estimates are not per se invalid. Thus, the Staff should express specific concerns with our estimates rather than dismiss such numerical estimates as questionable; and

2. Compliance would result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted. The Commission recognized that the equipment diversity requirement would be unreasonable in some circumstances or even unnecessary. Sensor diversity is excluded from the Rule, and the Staff position is that a sensor includes the trip bistable.[39] To now require sensor equipment diversity results in a cost substantially in excess of the cost contemplated when the Rule was adopted. Notwithstanding the exemption for sensor diversity, if some diversity is required, the published Staff Position and precedent point toward the acceptability of diversity by application where equipment diversity is difficult to achieve. Therefore, the requirement for equipment diversity results in a substantial excess cost.

38/ Letter from Ashok C. Thadani to Donald N. Grace (Mar. 17, 1989).

39/ Id.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

June 14, 1989

The Honorable Lando W. Zech, Jr.
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Zech:

SUBJECT: RELIABILITY AND DIVERSITY

During the 349th and 350th meetings of the Advisory Committee on Reactor Safeguards, May 3-6, 1989 and June 8-10, 1989, respectively, we discussed the implementation status of the anticipated transients without scram (ATWS) rule. Our Subcommittee on Instrumentation and Control Systems also met with representatives of the staff and the industry on April 21, 1989 to review the progress being made regarding this matter.

It appears that reasonable progress is being made, especially in light of some of the difficulties that have arisen in the interpretation and application of the rule. However, during the course of our discussions of compliance with the rule, two issues arose that we consider to have enough general significance to deserve further attention.

The first of these is the significance and application of diversity in systems that use redundancy to achieve high levels of reliability. The ATWS rule requires that diversity be used in an effort to further improve reliability. The staff interprets the rule to require diversity even if, in a particular application, there is no evidence that its use increases reliability. It appears, indeed, that this interpretation would be used even in situations in which, by virtue of commercial availability of components, maintenance considerations, or other relevant factors, diversity might reduce the reliability of a particular system. This seems to us to be contrary to the spirit of the ATWS rule which is aimed at increasing the overall reliability of the rapid shutdown system. Furthermore, we believe that in any situation in which diversity is considered as a means to increase reliability, it should be kept in mind that reliability is the objective, and not diversity per se. Thus, if diversity is to be required, effort should be made to ensure that it will contribute to increased reliability rather than making the system less reliable.

The second issue, which also came up during the discussion of the use of diversity, has to do with the possible influence of aging on the occurrence of common mode failures. The staff reasoned that even if diversity were not important during the first forty years of plant life, it might avoid development of common mode failures from "wear out," that might occur if operation beyond the original forty-year license is approved. We believe such concern may arise from a misunderstanding. While it is true that "wear out" of components does cluster around some "mean-time-to-wear-out," this time should be well known from test or experience, and components should be replaced or overhauled early enough to avoid it. Time-in-service for components that have not been replaced should be far enough removed from "wear out" that failure due to wear out (i.e., "aging") should not be a contributor to common mode failures.

We believe some further consideration of these two issues by the staff is merited, not only as they may bear on the application of the ATWS rule, but because of their significance generally.

Sincerely,

Forrest J. Remick
Chairman

D Allison

ENCLOSURE 3 OMITTED

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

May 30, 1990

MEMORANDUM FOR: Edward L. Jordan, Chairman
Committee to Review Generic Requirements

FROM: Frank Miraglia, Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: REQUEST FOR CRGR REVIEW OF THE BWROG APPEAL OF THE STAFF POSITION REGARDING DIVERSITY OF ROSEMOUNT TRIP UNITS

REFERENCE: BWROG "Appeal from Staff Position Requiring Total Equipment Diversity Under ATWS Rule (10 CFR 50.62)"

A briefing of the CRGR regarding the BWROG appeal on ATWS diversity requirements is requested at your earliest possible convenience. As you are aware, this appeal was submitted to James Taylor, Executive Director for Operations (EDO), on August 11, 1989, and the EDO subsequently assigned CRGR to take the lead to review this issue. The NRR staff was directed by the EDO to perform a thorough review of this appeal and provide to the CRGR its recommendation with any and all background information that may be required to complete the CRGR review.

In essence, the dispute involves use of the same type of Rosemount trip units in both the Alternate Rod Injection (ARI) system and the Reactor Trip System (RTS). The guidance published with the ATWS Rule states: "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 headers." The ATWS Rule itself, 10 CFR 50.62, states that each BWR must have an ARI system that is diverse (from the RTS) from sensor output to the final actuation device. The NRR staff does not agree with the Owners Group contention that the subject trip unit is part of the sensor and, therefore, the diversity requirement set forth in the ATWS Rule does not apply because the Rule allows the use of the same sensor for output to both the ARI and the RTS. Other disagreements between the staff and the BWROG center on the degree of diversity as it relates to the subject trip unit application. The BWROG maintains that pursuing ARI/RTS diversity is both unreasonable and impracticable and little if any risk reduction is achieved by using trip units in the ARI that are diverse from the trip units being used in the RTS. In contrast to these BWROG positions, the staff continues to believe that an increase in scram reliability can be achieved by using diverse trip units in the ARI systems at BWR power plants. Since there are different trip units that can be used in the ARI system which are available at a reasonable cost, the BWROG's assertion that the staff's position on this issue is both, "unreasonable and impracticable" is without support. After reviewing all information submitted relating to this appeal, it is still the staff's position that the health and safety of the public will be enhanced by employing diverse trip units in the ARI system as stated above.

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CONTACT:
V. Thomas, (SICB:DST)
492-0786

The staff has completed its review of all pertinent facts mentioned in this latest BWROG appeal, has determined that its initial position on the issues is unchanged, and recommends that the appeal be denied.

The three enclosures relate to the diversity issue. Enclosure 1 is a draft letter from the EDO to the BWROG containing the decision on the appeal. Enclosure ? contains the staff review findings of the BWROG appeal. Enclosure 3 is the NRR Contractor's Study Report on the BWROG appeal.

This information is submitted per discussion with the CRGR staff (D. Allison). We are prepared to discuss our recommendation on this appeal with the CRGR at the earliest opportunity.

Frank Miraglia

Frank Miraglia, Deputy Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Letter to BWROG
2. Staff Review Findings
3. A Review of Diversity in Trip Units, Feb. 1990

cc: D. Allison

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

ENCLOSURE 1

Mr. Stephen Floyd, Chairman
BWR Owners' Group
Carolina Power and Light Company
411 Fayetteville Street
Raleigh, North Carolina 27602

Dear Mr. Floyd:

This correspondence is the followup response to my previous letter dated August 31, 1989. At that time, I committed to notify the BWROG of my decision on the latest appeal of a staff position regarding the use of Rosemount trip units. The BWROG appeal addresses the issue of the degree of diversity required when implementing hardware on a boiling water reactor (BWR) to comply with the requirements of the ATWS Rule (10 CFR 50.62).

Following an intensive review of all the pertinent facts mentioned in the appeal by a panel of selected staff members [i.e., Committee to Review Generic Requirements (CRGR)] and my review of its findings and recommendation, I have concluded that the information submitted in support of the BWROG appeal does not present a sufficient basis to support your position that the present ARI design meets the diversity requirements as set forth in the ATWS Rule. Further, I do not agree with your assertion that the staff is requiring equipment diversity only for the sake of diversity, in spite of the lack of safety benefit. The primary conclusion I reach in review of this appeal is that the staff position is a proper interpretation of the Rule and that it is in the interest of improving the reliability of the scram function. Therefore, the subject appeal of the Owners' Group is hereby denied. I expect that each licensee will propose a schedule to the NRC for modifying its plant.

If you wish to discuss this decision or any issue you believe to be germane, please contact Scott Newberry, Chief of the Instrumentation and Control Systems Branch, at (301) 492-0782.

Sincerely,

James M. Taylor
Executive Director
for Operations

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

ENCLOSURE 2

LISTING OF MAIN APPEAL POINTS AND STAFF RESPONSES

Appeal Position Number 1

Page 6, Section III, Item A:

Item A: "The ATWS RULE Does not apply to The Rosemount Transmitter/trip Units."

The BWR owners argue: "The ATWS Rule clearly acknowledges that devices upstream of the sensor output are excluded from the reach of the Rule. The subject circuit boards in the Rosemount/trip units are upstream of the sensor output and, accordingly, the staff's decision to require equipment diversity (or for that matter, any diversity) is inconsistent with the rule."

Staff Response to Appeal Position Number 1

The staff agrees with the first part of the appeal statement above regarding devices upstream of the sensor output; but disagrees with the second part regarding the subject circuit boards.

The ATWS Rule clearly states that those devices which are located upstream of the sensor output are beyond the scope of the diversity requirement. It has been and continues to be the staff's position that the phrase "upstream of the sensor output" includes only the sensor and its associated process sensing lines and valves which make up the front-end of a typical measuring system. The staff does not consider, and has never considered to our knowledge, such devices as signal conditioning equipment, analog trip units, or indicating/recorders which are part of the receiving or back end of a typical measuring system to be "upstream" of the sensor output. Process measuring systems do not always employ an analog trip unit with the sensor; such is the case of certain monitors installed pursuant to the guidance in Regulatory Guide 1.97 "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." In those applications, the sensor outputs can be fed directly to an indicator/recorder or data logger without the need for a trip unit.

The staff position regarding what constitutes a sensor is supported by the General Electric (GE) Report, NEDC-31336, "Instrument Setpoint Methodology," dated October 1986; the Rosemount Controls Inc. Product Data Sheet No. 2302; and several industry standards.

GE treats the sensor and analog trip unit as two separate components when they are used as part of an instrument channel (Page I-4, Items 9 and 10, in NEDC-31336). General Electric defines a sensor as: "The portion of the instrument channel which converts the process parameter value to an electrical signal." The trip unit is defined as: "The portion of the instrument channel which compares the converted process value of the sensor to the trip [desired] value, and provides the output "trip" signal when the trip value is reached." Another example of GE's approach to considering these components as separate components is shown on Pages I-12 and I-13 of the same report. On page I-12, the sensor transmitter and analog trip unit are treated as separate components in GE's discussion of the methodology for establishing instrument channel accuracy. The sensor transmitter component is represented as one term, A_T (A_T is equal to transmitter accuracy) and the trip unit is represented by a different term A_{TU} (A_{TU} is equal to trip unit accuracy). On Page I-13, in discussing instrument channel drift, GE assigns separate values of drift for the transmitter and the trip unit (i.e., D_T and D_{TU} respectively).

Another example of this approach by industry regarding the separate nature of the sensors and the trip units is demonstrated by Rosemount in their Product Data Sheet #2302. The electrical block diagram in this example shows the sensor as only one portion of the sensor/transmitter assembly. The sensor portion includes the capacitive element (plates) which sense a change in the sensing capsule oil pressure which in turn is affected by the changes in the process parameter value; the changes in the electrical characteristics of the plates are then converted to a proportional electrical signal. The remaining portion of the sensor transmitter is referred to as the transmitter section and includes the demodulator, current detector, oscillator, current control amplifier, and voltage regulator. The block diagram does not show the analog trip unit but does clearly show the converted process parameter output signal. As stated above, this output signal is sent "downstream" to indicators, trip units and data loggers as desired.

Additionally, <u>all</u> industry standards that have been reviewed by the staff define and treat the sensor and analog trip unit (sometimes referred to as a bistable or an alarm unit) as separate devices. These standards or guidelines include:

° IEEE Standard 603-1980: "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"

° ANSI/ISA S 51.1-1979 "Process Instrumentation Terminology"

° SAMA Standard PMC 20.1-1973 "Process Measurement and Control Terminology"

° ISA-RP67.04 Part II-1989-Draft "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation"

Early vintage BWR type power plants such as Oyster Creek, Dresden, Millstone, and the like originally used a local indicating pressure or differential pressure switches manufactured by Barton to initiate the scram function or actuate the engineered safety features system(s) when abnormal plant conditions were reached. However, after issuance of IE Bulletin 79-01B, "Environmental Qualification of Class 1E Electrical Equipment," many of these licensees opted to replace the local indicating type switch with an analog type measuring system consisting of the sensor/transmitter (described above) and an analog trip unit to perform the same functions. The sensors of each system sense the plant process in the same manner. The indicating switch, which is located in the body of the sensor, operates from physical movement of the sensor's sensing element (e.g., bourdon tube, diaphragm, bellows, etc.) whereas its counterpart, the trip unit, needs an electrical conversion (after the sensing element movement) and then transmission (signal conditioning) of the resultant signal to the trip unit to provide the same scram trip or actuation functions as the indicating switch. Replacing the switches in the RTS or ARI, which are outside the scope of the ATWS Rule, with the analog transmitter and trip unit adds a component (the trip unit) which the staff views not to be part of the sensor and within the diversity requirements of the Rule. The BWROG disagrees.

On page 6 of the Appeal, the BWROG presents an excerpt taken from SECY 83-293 as support for its contention that the sensor/trip unit should be treated as one device. This excerpt is taken from an appendix to the ATWS Task Force recommendations regarding an ATWS Rule. The excerpt from SECY 83-293 reads:

> "The trip portion of the sensor system consists of bistables that signal an out-of-tolerance condition. This portion of the system is vulnerable to bistable calibration errors and like component common cause failures. However, continuous monitoring of the sensor output, and the frequent testing of the trip values provide a good chance of discovery of such common cause problems.... Though differences exist in the level of redundancy and logic structure, these only influence the independent failure contribution which does not contribute significantly to the overall RPS unavailability. Therefore, for the purposes of this analysis, the sensor portion of the RTS will be ignored."

This discussion can be interpreted in a manner that reflects the view of the BWROG or interpreted in another manner to support the staff's position on this issue. Review of all of the Task Force Report, however, contradicts the BWROG interpretation of the above excerpt. The following excerpt taken from the same report states that the transmitters, amplifiers, logic matrices and relays are part of the measuring systems logic subsystem. In this statement even the transmitters are said to lack diversity, and the sensor is the only device that is not considered to be part of the logic subsystem. The excerpt reads:

> "The transmitters, amplifiers, logic matrices, and relays that make up the logic subsystems do have redundancy to some degree, but generally lack diversity. The PRA's conducted to date generally have not quantified the contribution to unavailability caused by the possible common cause influences on the logic subsystems. The failure rates for these components are low and multiple failures are rare, although multiple failures caused by such influences as temperature degradation for certain logic components have been reported. Failures in these components are generally not announced at once and must await surveillance testing. In addition, comparator adjustments and calibrations can introduce human error."

We conclude that this report is ambiguous with respect to defining the scope of the Rule.

Finally, all PWR power plants are also required by the ATWS Rule to install new systems. They employ the analog type measuring systems similar to those measuring systems in use at many BWRs to actuate a diverse scram system and/or diverse auxiliary feedwater/turbine trip systems. To date, the staff is not aware of any utility interpretation of the Rule that led to non-diverse trip units or bistables. On the contrary, all plants, to our knowledge, have designed and are installing systems that use different bistables/trip units in the RTS and ATWS systems.

We conclude that the background information on sensor channels and logic subsystems in SECY 83-293 is ambiguous and does not support the BWROG. We conclude that the definition of sensor in the literature and in practice is clear and that the ATWS Rule does apply to the trip units.

Appeal Position Number 2

Page 9, Section III, Item B:

Item B: "Even if it is determined that the ATWS Rule applies to the Rosemount/ trip units, these units meet the Rule."

The BWROG acknowledges the need for the Commission's diversity requirement "from sensor output to the final actuation device." However, they maintain that the Rule does not specify the type of diversity, but simply requires diversity. Because the alternate rod injection (ARI) system employs combinations of methods of diversity such as equipment, functional, and application state diversity, the BWROG reasons that the system complies with the ATWS Rule.

Staff Response to Appeal Position Number 2

The Statement of Considerations published with the ATWS Rule defines what is meant by the term "diversity" as required in the ATWS Rule. The Statement of Considerations states that "equipment diversity" is the primary objective of the general term "diversity" in the Rule. The staff has always interpreted equipment diversity to mean unlike or different equipment.

During staff reviews of various utility ATWS designs, equipment diversity has always played a significant role when assessing the acceptability of a given functionally diverse application, as in the case of the ARI system. For example, two instrument channels that are measuring different plant parameters such as level and flow and are part of the same logic matrix, are sufficiently diverse only if the components in each channel are different from sensor output up to and including the final actuation devices that vent the air header. In addition, past experiences and the studies conducted jointly by industry and the NRC that led to the ATWS Rule and the associated Statement of Considerations leave no doubt that the intent of "diversity" set forth in the Rule is to improve the reliability of the scram function by minimizing the potential for common mode failures. The staff believes that this increase in reliability is achieved through equipment diversity so long as the potential drawbacks of diversity (such as unreliable equipment or additional failure modes) are adequately addressed.

The need for equipment diversity can be illustrated by reviewing events involving equipment used in the reactor trip systems to achieve a reactor scram. For example, the Salem event resulted largely from inadequate equipment diversity. Two identical undervoltage trip attachments, located one in each of two reactor trip circuit breakers, simultaneously failed to perform their intended functions following a demand to scram, thereby causing the ATWS event.

An example of a component failure that has a potential to lead to common mode failure recently occurred when a defective component [1/] was used in the Rosemount 710 Master and Slave trip unit circuitry. These are the trip units in question. The deficiency was caused by a change in the manufacturing process. Specifically, under certain environmental and operating conditions, the trip unit may fail to actuate as intended even when in different energized states. The vendor has notified end-users of the potential problem and has offered a replacement unit considered more suitable for the intended service. In addition, our recent search of the Nuclear Plants Reliability Data System (NPRDS) uncovered other failures involving the Rosemount trip units which bring into question the perception that they are highly reliable and not vulnerable to common mode failure. The following are "Failure Descriptive Narratives" submitted by just one licensee about faulty Rosemount trip units:

- Grand Gulf personnel while conducting an 18-month surveillance test noted that an analog trip unit indicated a trip condition, but no reactor protection system response occurred. Subsequent investigation of the cause for failure revealed that a defective Rosemount trip unit was determined to contain two faulty operational amplifiers, a faulty potentiometer, one faulty timer and one faulty diode.

- Grand Gulf personnel experienced another failure of a Rosemount trip unit and in the Cause of Failure Narrative they state in part that "... the input diode failure is considered a normal electrical failure." The diode was replaced, a retest was performed satisfactorily on the trip unit, and it was returned to service.

The examples cited above are intended to illustrate the purpose of the diverse equipment in the ARI system which is to improve scram reliability by minimizing the potential for common mode failures and to enhance the confidence level that <u>all</u> power reactor plants will automatically scram on demand.

1/ (Part 21 notifications on Rosemount model 710 Trip/Calibration units and 414 E/F resistance bridges, dated August 17 and October 10, 1989)

This is not to say that the staff has always required completely different equipment in all instances during licensees' proposals to provide a diverse or alternate trip system. In the past, the staff has exercised engineering judgement and will continue to do so as questions on equipment diversity and the degree of design difference arise. The staff's decisions on these diversity issues are based on the reasonableness and practicableness of the given application coupled with a judgement regarding fundamental design differences. These are the bases the staff has used in arriving at the present decision to require licensees to use trip units in the ARI system diverse from similar functional trip units being used in the reactor trip system.

The BWROG argues against the use of diverse trip units and maintains that diversity from the RTS is already achieved throughout the ARI by combinations of allowable methods of diversity. It states the ARI system employs equipment, functional, and application state (i.e., de-energized versus energized) diversity from the RTS and thus complies with the Rule.

The staff agrees that combinations of methods such as energization states, the use of AC power versus DC power, functional diversity, components from different manufacturers, and different components from the same manufacturer are used when assessing the diversity issue. In addition to these methods, other factors that may influence the assessment include the history of successful operation and the ability to demonstrate reliability through periodic surveillance tests.

With respect to the BWROG contention that the present ARI system complies with the Rule, the staff has carefully reviewed the scenario presented on pages 9 and 10 of the appeal and disagrees with BWROG position for the following reasons:

° Functional diversity using different components is an acceptable means to meet the diversity requirement of the ATWS Rule. However, for the BWROG Loss of Feedwater event (LOF) mentioned above, there is no functionally diverse trip that uses diverse equipment to automatically initiate scram and mitigate the LOF event. For a LOF, the only RPS signal is low reactor water level. [This issue is discussed in detail in the attached contractor report dated February 1990, Enclosure 3.]

° Very little trip unit diversity is provided by different energization states. The bistable element (as stated on Page 10 of the appeal) is not the only active component on the trip unit during normal operation. The staff maintains that active components are not just components that have a physical movement such as relays or switches. Active components that could fail due to common cause are also those components that change their electrical states such as logic networks, ze[illegible] iodes, and transistors. Examples of components that don't continually change electrical state are resistors, capacitors, terminal strips and potentiometers.

- The issue of reasonableness is not violated because there are trip units available that have diverse active components as defined above.

- The practicable aspect of this issue is not violated because the cost to replace or use diverse trip units is not prohibitive if the trip unit card manufactured by GE is used.

- Other trip units that are available for replacement have proven histories of successful operation in similar service applications at many nuclear power plants.

- The use of other available diverse trip units will improve reliability and will minimize the potential for common mode failures in the ARI systems at BWR type power plants.

The BWROG has argued that the drawbacks of diversity outweigh the safety benefits in this case. In an effort to assist us in the assessment of the safety benefit of replacing the trip units in the ARI with different trip units, we have, with the assistance of our contractor, reviewed in detail the quantitative reliability and risk assessments performed by the BWR Owners' Group and CP&L which were referenced in the BWROG appeal.

Current PRAs are not helpful in resolving this issue because common mode failures between the RPS and the ARI are not modeled at all or in very little detail. For example, prior to the ATWS Rule, the Utility Group on ATWS did not explicitly include common mode failures involving the RPS and ARI in its analysis. The values used in its analysis suggest that common mode failures are not considered at all. The Brunswick PRA referenced in the CP&L appeal also provides no models sufficiently detailed to aid in this evaluation. The simplified analysis provided by CP&L does provide a common mode failure analysis but also introduces considerable benefit from manual scram by the operator. The General Electric analysis includes common cause failures within each trip function but does not include any consideration of common cause failure of identical trip units that exist in all of these functions. Even the staff ATWS models which provided a basis for the recommended ATWS rule did not model components such as trip units separately. A more detailed review and description of these analyses is contained in Enclosure 3.

The improvement in overall system reliability provided by diversity is difficult to estimate quantitatively. However, also contained in Enclosure 3 is a quantitative estimate of this improvement using the same event trees used by the staff in recommending the ATWS Rule. While the uncertainties in such estimates are large, we believe that the estimates in Enclosure 3 are reasonable and that they provide an improved methodology for evaluating the safety benefit. In addition to concluding that replacing the ARI trip units would be cost beneficial, these models point out systematically that, contrary to our previous understanding that equipment outside the scope of the ATWS Rule (sensors) was diverse to a very large extent in the BWR design, identical trip units exist in all instrumentation channels that automatically trip the plant in response to a loss of feedwater event. We conclude that installation of reliable trip units that are different will improve safety.

With respect to the "drawbacks-of-diversity" that the BWROG noted in its letter to J. Taylor, NRC, dated August 11, 1989, and in the subsequent meeting with the staff (same subject) on November 15, 1989, little new or substantive information was offered in response to the EDO's request for information. Enclosure 3, on pages 15 through 19, discusses in detail the events surrounding the three drawbacks of diversity highlighted by BWROG. We conclude that there are no significant drawbacks to installing different trip units.

Appeal Position Number 3

Page 11, Section III, Item C:

Item C: If the term "diversity" is more broadly construed to require "equipment diversity," such construction should be read as "equipment diversity, to the extent reasonable and practicable."

The BWROG maintains that, as stated in its Appeal Position Number 2, the Rule itself does not impose a limitation on diversity so as to require that all diversity be achieved through diversity of equipment. Rather, the staff's support for equipment diversity comes from guidance set forth in the Statement of Considerations.

Staff Response to Appeal Number 3

As noted in the staff responses to Appeal Position Number 2, the staff's position regarding functional and equipment diversity are influenced by the aspects of both reasonableness and practicableness, risk reduction/benefit gained, and engineering judgement. Additionally, these staff positions have been and continue to be strongly influenced by the guidance set forth in the Statement of Considerations as the Owners' Group indicated above.

Responses to the many concerns and assertions that the BWROG raised throughout this appeal position are addressed in the staff responses to Appeal Positions 1 and 2 herein and/or in Enclosure 3.

Conclusion

We conclude that the original NRR position is the proper one. The definition of a sensor in the literature and in practice is clear, and the diversity statement in the ATWS Rule applies to the analog trip units. The language found in an appendix to the ATWS Task Force Report attached to SECY 83-293 recommending a rule is ambiguous. We conclude that in the affected plants no diverse equipment to the RTS analog trip units exists for automatically scramming the reactor following a loss of feedwater. The BWROG provided insufficient information to support their assertions regarding the drawbacks of diversity. Our review indicates that these suggested drawbacks are non-existent or are not significant. Finally, we conclude that replacement of the Rosemount trip units will improve safety, is cost beneficial, and should proceed. It is our judgement that such action is reasonable and practicable and is consistent with the guidance issued with the ATWS Rule.