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February 7, 1990

[redacted]

Mr. Carl H. Berlinger
Chief, Generic Communications Branch
United States Nuclear Regulatory Commission
Washington, DC 20555

Subj: Comments On Draft NRC Bulletin On Loss Of Fill-Oil In Rosemount Transmitters

Dear Mr. Berlinger:

The Nuclear Utility Backfitting and Reform Group (NUBARG) and the Nuclear Utility Group On Equipment Qualification (NUGEQ) provide the following comments on the above-referenced draft Bulletin. Generally, we agree with the purpose of the subject draft Bulletin; however, our members are concerned that the current scope of the draft Bulletin and the absence of proper backfitting justification will cause this draft Bulletin to dilute a meaningful effort on the part of the industry to effectively address this concern.

Rather than providing you with redundant observations, we begin by stating that our members wholly endorse the comments already provided by NUMARC, especially with regard to the inclusion of Model 1151 and 1152 transmitters. These models utilize elastomeric o-rings, rather than metal process o-rings, which do not create the additional stresses needed to cause the glass to metal seal failure mechanism.

In addition, our members believe that the Staff should reconsider the tremendous amount of research data compiled by Rosemount which indicates that transmitter failures can be correlated to certain "suspect" manufacturing lots having a failure fraction substantially higher than that of all other manufacturing lots. Rosemount has also compiled data which demonstrates that the failure rate for transmitters is highest when the transmitters are subjected to high static pressure over a certain period of time. The time in service required to reach the peak failure rate decreases with static pressure. Failure rates drop exponentially with time in service, reaching acceptably low levels at times that can be predicted by a product of pressure and time. We bring this to your attention because many of our members do not believe that transmitters which have been in service over a long period of time at full system pressure should be required to be replaced unless there is an indication of fill-oil leakage.

Nevertheless, the draft Bulletin would request licensees to take the following actions:

1. Identify all Rosemount Model 1151, 1152, 1153, and 1154 transmitters (except Model 1153 and 1154 transmitters manufactured after July 11, 1989) which are in safety related systems or ATWS systems;

2. Review plant records to determine whether any of the identified transmitters exhibit signs of fill-oil leakage;

3. Develop an enhanced surveillance program to monitor transmitters for symptoms of loss of fill-oil;

4. Identify whether any Model 1153 and 1154 transmitters from the high failure rate lots are present;

5. Justify continued operation until high failure rate lot transmitters used in the reactor protection system or ESFAS can be replaced; and

6. File requested reports.

The NRC Staff indicates that the above actions constitute a backfit under 10 C.F.R. § 50.109, but that a full backfit analysis is not required because the backfit is necessary to bring facilities into compliance with existing requirements. The requirements cited include General Design Criterion 21, which requires that protection systems be designed for high functional reliability and with sufficient capability to allow periodic testing of their functioning when the reactor is in operation, and IEEE-279 which requires that means be provided for checking operational availability of input sensors during reactor operation.

In general, facilities already comply with such system design and surveillance requirements. At issue is whether facilities should develop new programs to address high-failure-rate transmitters installed within those systems. We concur that transmitters known to be susceptible to unacceptable failure rates should be identified and replaced. Therefore, compliance is appropriate for actions 1-6 above, but only for transmitter Models 1153 and 1154 from the known high failure rate lots. Transmitter Models 1153 and 1154 from lots not having high failure rates should be excluded from the requested actions, as should transmitter Models 1151 and 1152. We do not believe an adequate basis has been shown to conclude that these transmitters are susceptible to unacceptably high failure rates. Therefore, there is no justification for indicating that protection systems containing these transmitters are not in compliance with existing regulations. Accordingly, a backfit analysis under 10 C.F.R. § 50.109 is appropriate prior to imposing on licensees the above-described requirements for these transmitters.

Use of the compliance exception to the backfitting rule, Section 50.109(a)(4)(i), is not justified because of the demonstrated high reliability of Rosemount transmitters, except for the high-failure-rate manufacturing lots. An appropriate backfit analysis would likely indicate that the cost of including other transmitters in the actions required by this draft Bulletin far exceed the benefit which may result from the actions.

Accordingly, we recommend that only the known high-failure-rate transmitters be included in the scope of the Bulletin. In the alternative, a backfit analysis should be performed prior to including transmitters other than the high-failure-rate manufacturing lot transmitters, Models 1153 and 1154, in the actions required. We understand that the NRC would like to identify failure modes of all Rosemount transmitters, i.e., those having acceptably low failure rates, however, a less manpower intensive and costly program would be more appropriate than the program of the draft Bulletin. To this end, we recommend a bulletin notifying licensees to be aware of failures of Model 1151 and 1152 transmitters, and requesting licensees, for example, to return failed transmitters to Rosemount for testing.

Sincerely,

John MacEvoy

Nicholas S. Reynolds
John A. MacEvoy

Counsel to the Nuclear Utility Backfitting And Reform Group and the Nuclear Utility Group On Equipment Qualification

cc: Dennis P. Allison

Enclosure 3 to the Minutes of CRGR Meeting No. 179
Proposed Generic Letter on BWR Channel Box Bow Problem
February 7, 1990

TOPIC

A. Thadani (NRR) and D. Fieno (NRR) presented for CRGR review a proposed generic letter on actions to be taken by BWR licensees to address the problem of loss of thermal margin due to excessive fuel channel box bow. Briefing slides used by the staff to guide their presentation and discussion with the Committee at this meeting are enclosed (see Attachment).

BACKGROUND

The documents submitted to CRGR for review in this matter were transmitted by memorandum dated January 10, 1990, J.H. Sniezek to E.L. Jordan; that review package included the following documents:

1. Proposed Generic Letter (undated), "Correction of Deficiency in BWR Critical Power Ratio Calculation Due To Channel Box Bow", and attachment:

   a. NRC Information Notice 89-69, dated September 29, 1989, "Loss of Thermal Margin Cuased by Channel Box Bow"

2. "Response to Requirements for Content of Package Submitted for CRGR Review"

CONCLUSIONS/RECOMMENDATIONS

As a result of their review of this matter, including the discussions with the staff at this meeting, CRGR recommended in favor of the issuance of guidance to BWR licensees in connection with the channel box bow problem; but the Committee recommended a number of changes, with respect to both the format and the content of the guidance to be issued, as follows:

1. The vehicle for implementing this regulatory action should be an NRC Bulletin instead of a Generic Letter; reference to 10 CFR 50.54(f) should be deleted from the bulletin. The bulletin can be addressed to CP holders (for their information); but no "Actions" or "Reporting Requirements" should be directed to CP holders at this time.

2. The scope of "Requested Actions" should be narrowed to apply only to BWR licensees that currently use channel boxes for a second bundle lifetime; the only action requested of such licensees should be to verify that current technical specifciation CPR limits are met.

3. The scope of "Reporting Requirements" should be similarly narrowed to apply only BWR licensees that currently use channel boxes for a second bundle lifetime; affected licensees' responses shall include the following:

a. Advise the NRC of the number and disposition of such channel boxes in the core.

b. Describe the methods and the associated data base used to account of channel box bow during their second bundle lifetime use, to conformance with the CPR technical specification operating and safety limits.

4. The staff acknowledges that the possibility of any fuel failures as a result of channel box bow in the first bundle life time is remote for U.S. BWRS; so no "Actions" or Reporting Requirements" should be directed to BWR licensees in those circumstances. The discussion regarding new vendor methodologies (currently under review by NRC), that properly account for channel box bow for first bundle lifetime, does apply to all BWR licensees, and should be retained in the bulletin. The staff should also indicate in that discussion when the NRC review will be completed and the new methodologies approved for use by the licensees in core reload calculations, and how the channel box bow question will be handled in core reload applications in the interim.

5. The staff should coordinate all changes resulting from CRGR recommendations with the CRGR staff, and resubmit a revised package to the CRGR for final review on a negative consent basis.

It was noted that this action was considered to be justified as a compliance backfit.

File
Mtg 179
Channel Box Bow

PRESENTATION TO CRGR ON

PROPOSED GENERIC LETTER ON

DEFICIENCY IN BWR THERMAL LIMITS

CALCULATIONS CAUSED BY CHANNEL BOX BOW

FEBRUARY 7, 1990

Attachment to
Enclosure 3

BACKGROUND

- ISSUE DISCUSSED WITH NRC BY ABB ATOM, INC. (ABB) AT MEETING ON JULY 20, 1989 ON OSKARSHAMN 2 FUEL FAILURES OBSERVED DURING AUGUST 1988 REFUELING OUTAGE

- MEETINGS WITH US BWR FUEL VENDORS

  - GE - SEPTEMBER 12, 1989

  - ANF - SEPTEMBER 13, 1989

- NRC INFORMATION NOTICE 89-69 (SEPTEMBER 29, 1989)

OSKARSHAMN 2 FUEL FAILURE CONCERNS

- EXTENSIVE INVESTIGATIONS LED TO THE CONCLUSION THAT THE FUEL FAILURES WERE CAUSED BY DRYOUT (CLADDING OVERHEATING) CONDITIONS WITH SECONDARY FAILURES CAUSED BY HYDRIDING

- DRYOUT CONDITION ON FAILED RODS CAUSED BY TWO EFFECTS
  - CHANNEL BOX BOW GREATER THAN EXPECTED
  - INCORRECT MODELLING IN PROCESS COMPUTER OF SVEA-64 FUEL BUNDLES (THIS IS NOT A CONCERN FOR US BWRs)

axial Representation of Channel Bow

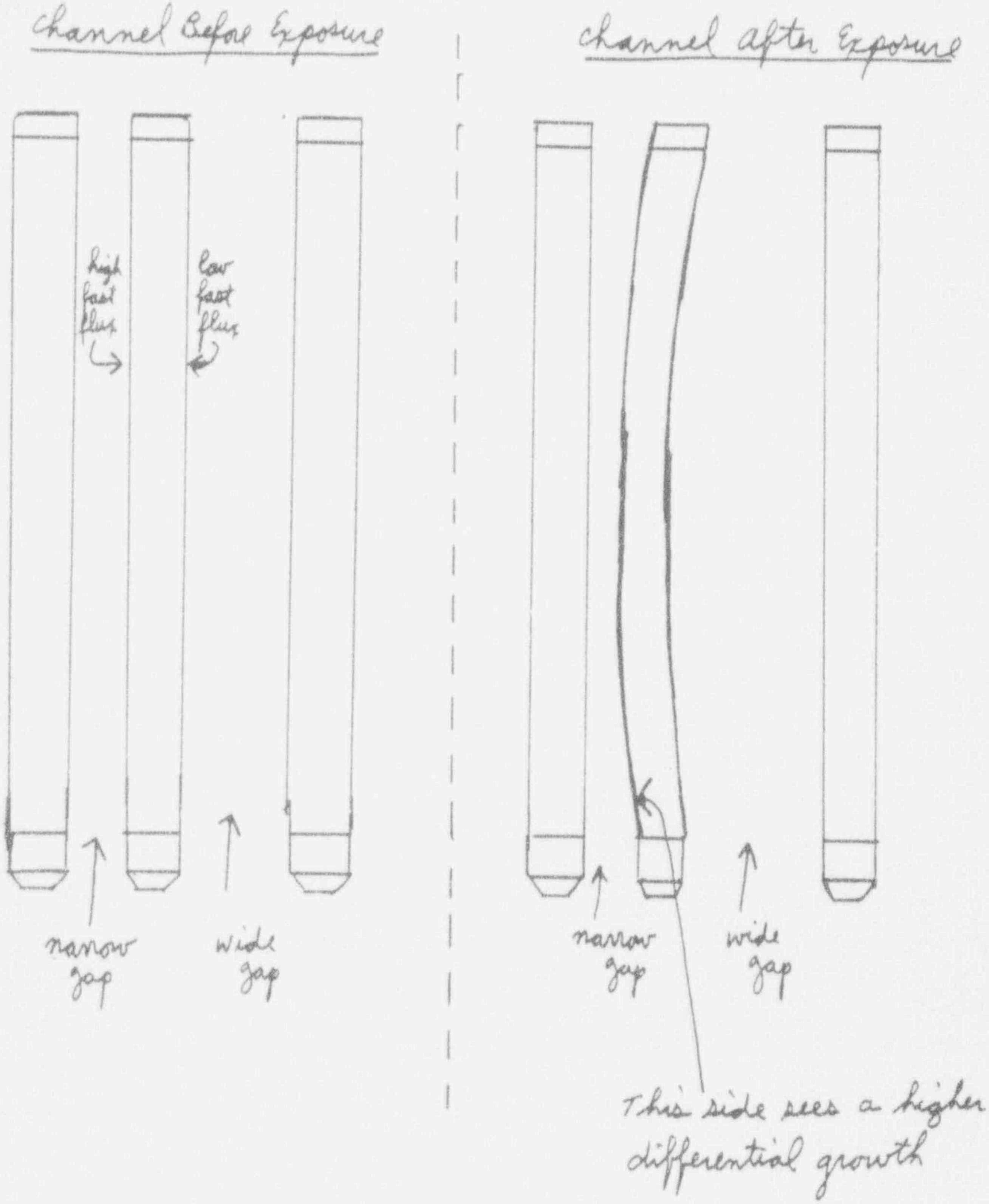
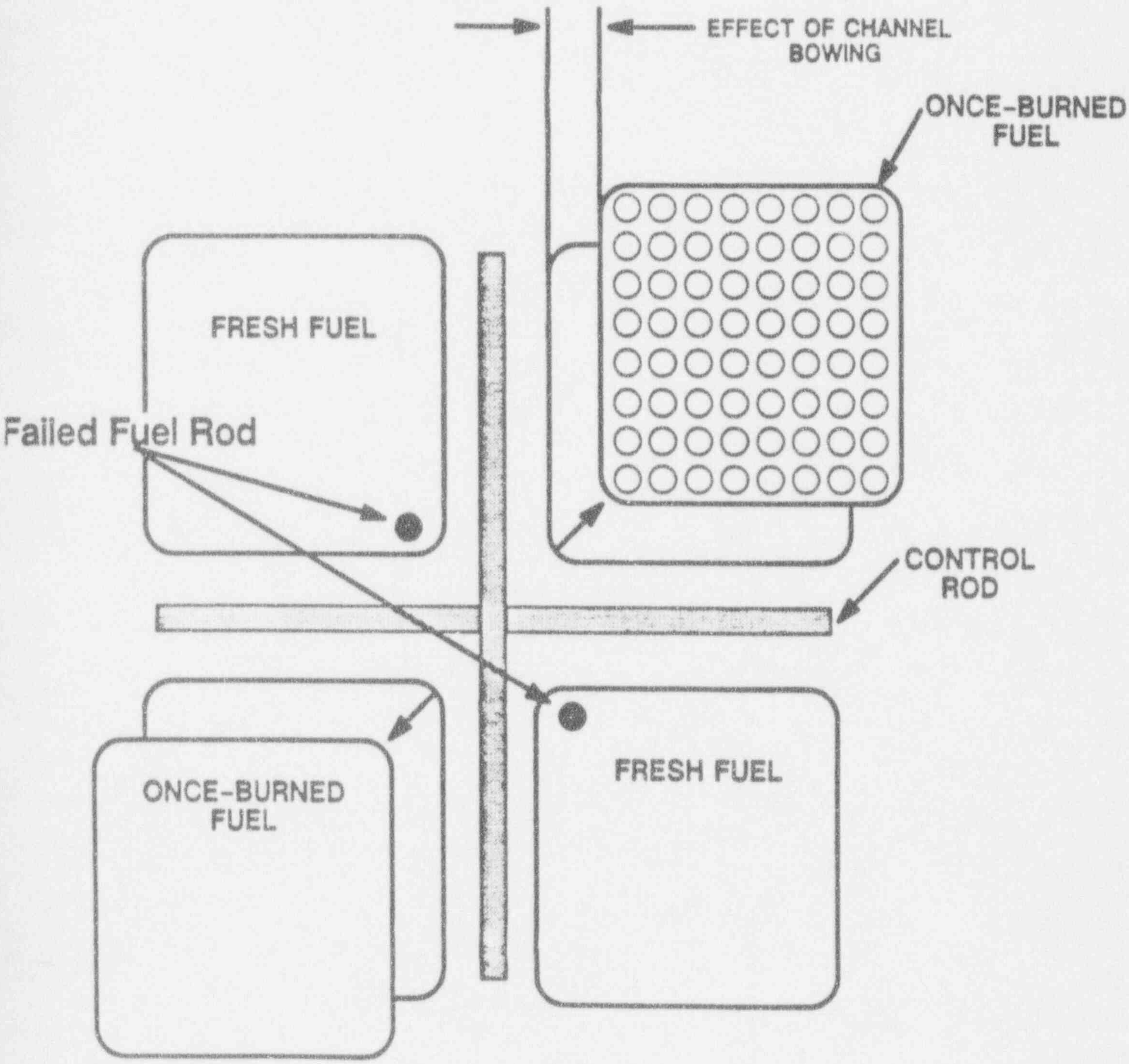


Figure 1
Channel Bow



APPLICABLE REGULATIONS

- GENERAL DESIGN CRITERION 10

  - THE REACTOR CORE AND .... SHALL BE DESIGNED WITH APPROPRIATE MARGIN TO ASSURE THAT SPECIFIED ACCEPTABLE FUEL DESIGN LIMITS ARE NOT EXCEEDED DURING ANY CONDITION OF NORMAL OPERATION, ...

- 10 CFR 50.36(c)(1)(i)A

  - REQUIRES SAFETY LIMITS TO BE SET FOR IMPORTANT PROCESS VARIABLES WHICH ARE FOUND TO BE NECESSARY TO REASONABLY PROTECT THE INTEGRITY OF THE PHYSICAL BARRIERS THAT GUARD AGAINST THE UNCONTROLLED RELEASE OF RADIOACTIVITY

IMPACT OF BWR CHANNEL BOX BOW

- POTENTIAL VIOLATION OF THE SAFETY LIMIT MINIMUM CRITICAL POWER RATIO (SLMCPR) AND POTENTIAL FOR FUEL FAILURES IN BWRs

- IMPACT ON LINEAR HEAT GENERATION RATE (LHGR) AND FUEL PELLET OVERHEATING

- IMPACT ON TRAVERSING INCORE PROBE (TIP) READINGS AND EFFECT ON CORE MONITORING

IMPACT OF CHANNEL BOW ON CPR MARGINS

- GE
  - D-LATTICE PLANTS DELTA-CPR/ICPR UP TO -.03
  - C-LATTICE PLANTS DELTA-CPR/ICPR UP TO -.02
  - ABOVE ESTIMATES ARE FOR SINGLE BUNDLE LIFETIME CHANNELS; IMPACT IS MUCH GREATER FOR SECOND BUNDLE LIFETIME CHANNELS DUE TO ACCELERATED RATE OF BOW

- ANF
  - PRELIMINARY RESULTS COMPARABLE TO GE RESULTS FOR NEW CORE ANALYSIS METHODS
  - CORES ANALYZED BY CURRENTLY APPROVED METHODS CONTAIN SUFFICIENT CONSERVATISM TO ACCOUNT FOR CHANNEL BOW EFFECTS

IMPACT OF CHANNEL BOW ON LHGR MARGINS

- IMPACT LESS IN BOTTOM OF CORE WHERE MARGIN TO LHGR IS LEAST
- IMPACT GREATER IN TOP OF CORE WHERE MARGIN TO LHGR IS GREATER
- OVERALL IMPACT ON LHGR NOT JUDGED TO BE SIGNIFICANT

IMPACT OF CHANNEL BOW ON TIP READINGS

- NEGLIGIBLE IMPACT ON GAMMA - TIPs
- IMPACT ON THERMAL-TIPs - HOWEVER UNCERTAINTY USED IS LARGE ENOUGH TO COVER ANY INCREASE IN TIP UNCERTAINTY

CORRECTIVE ACTIONS REQUESTED BY PROPOSED GENERIC LETTER

- ENSURING THAT THE PROCEDURES FOR MONITORING THERMAL LIMITS DURING REACTOR OPERATION WILL PREVENT VIOLATION OF THE MINIMUM CPR OPERATING LIMIT BY THE USE OF AN NRC APPROVED METHODOLOGY THAT TAKES INTO ACCOUNT THE NEW DATA ON FUEL CHANNEL BOWING, OR

- IMPOSING ON THE OPERATING CPR LIMIT (OR LIMITING CURVE) A CHANNEL BOW PENALTY OF EITHER 0.03 DELTA-CPR FOR D-LATTICE PLANTS OR 0.02 DELTA-CPR FOR C-LATTICE (OR S-LATTICE) PLANTS AND BY REMOVING ANY CHANNEL BOXES THAT ARE BEING REUSED AFTER THEIR FIRST BUNDLE LIFETIME

RESPONSE REQUESTED BY PROPOSED GENERIC LETTER

- WHETHER UTILITY WILL IMPLEMENT THE CORRECTIVE ACTIONS FOR THE FIRST RELOAD SCHEDULED AFTER APRIL 30, 1990, INCLUDING THE DATE OF THE RELOAD

- WHETHER OR NOT CHANNEL BOXES ARE BEING USED FOR A SECOND BUNDLE LIFETIME

JUSTIFICATION FOR CONTINUED OPERATION

- CONSERVATISM IN SAFETY LIMIT CPR

- LOW PROBABILITY OF EXCEEDING THE SAFETY LIMIT CPR DURING TIME FOR IMPLEMENTING CORRECTIVE ACTION (LOW PROBABILITY OF HAVING LIMITING TRANSIENT)

- LOW PROBABILITY OF A ROD GOING INTO BOILING TRANSITION IF THE SAFETY LIMIT CPR IS EXCEEDED

- LOW PROBABILITY THAT ANY SUCH ROD WILL BE IN BOILING TRANSITION FOR A SIGNIFICANT PERIOD OF TIME

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

JAN 29 1990

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

FROM: James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: REQUEST FOR REVIEW OF A DRAFT BULLETIN ON LOSS OF FILL-OIL IN TRANSMITTERS MANUFACTURED BY ROSEMOUNT

On April 21, 1989 the NRC staff issued Information Notice No. 89-42 "Failure of Rosemount Models 1153 and 1154 Transmitters." This information notice alerted addressees to potential problems relating to Rosemount Model 1153 and 1154 transmitters that may be susceptible to failure due to loss of fill-oil from the sensor's sealed sensing module. These transmitters are widely used throughout industry in plant safety systems, including the reactor protection and engineered safety features actuation systems. The performance of a transmitter that is leaking fill-oil gradually deteriorates and may eventually lead to failure. Although some failed transmitters have shown symptoms of loss of fill-oil prior to failure, it has been reported that in some cases the failure of a transmitter that is leaking fill-oil may not be detectable during operation. Loss of fill-oil may result in a transmitter not performing its intended safety function.

Rosemount has recently indicated that, in addition to Model 1153 and 1154 transmitters, they have also identified confirmed failures of both Model 1151 and 1152 transmitters due to loss of fill-oil. Rosemount has also indicated that they manufacture both complete transmitters and transmitter parts (including sensing modules) for other manufacturers who supply equipment for use in nuclear power plants. In addition, Rosemount has indicated that unauthorized remanufacturers and refurbishers exist for Model 1151 and possibly Model 11[illegible], 1153, and 1154 transmitters.

General Design Criterion (GDC) 21 "Protection System Reliability and Testability" of 10 CFR Part 50, Appendix A requires the protection system to be designed for high functional reliability and with sufficient capability to allow periodic testing of its functioning when the reactor is in operation in order to readily detect failures of subcomponents and subsystems within the protection system as well as loss of the required protection system redundancy as they occur. 10 CFR 50.55a(h) requires that protection systems meet the Institute of Electrical and Electronics Engineers Standard: "Criteria for Protection Systems for Nuclear Power Generating Stations" (IEEE-279). IEEE-279 states that means shall be provided for checking, with a high degree

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492-1167

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of confidence, the operational availability of each system input sensor during reactor operation. Thus, we conclude that facilities that utilize transmitters that may be susceptible to loss of fill-oil may not be in full compliance with these regulations because undetected transmitter failure could occur.

Enclosed are the proposed bulletin and appropriate supporting background information. We request that CRGR review of this matter be scheduled as quickly as possible. The bulletin is sponsored by Charles E. Rossi, Director, Division of Operational Events Assessment, and by Ashok Thadani, Director, Division of Systems Technology.

James H. Sniezek

James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

Enclosures:
1. NRC Bulletin No. 90-XX
2. CRGR Item IV.B.

OMB No.: 3150-0011
NRCB 90-XX

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

January xx, 1990

NRC BULLETIN NO. 90-XX: LOSS OF FILL-OIL IN TRANSMITTERS MANUFACTURED BY ROSEMOUNT

Addressees:

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose:

This bulletin is being provided to request that addressees take actions to promptly identify and replace transmitters manufactured by Rosemount that may be leaking fill-oil.

Description of Circumstances:

NRC Information Notice No. 89-42 "Failure of Rosemount Models 1153 and 1154 Transmitters," dated April 21, 1989, was issued to alert industry to a series of reported failures of Rosemount Models 1153 and 1154 pressure and differential pressure transmitters. The reported failures occurred at Northeast Utilities' Millstone Unit 3 between March and October 1987. Subsequent investigation into the cause of the failures by Rosemount confirmed that the failure mode was a gradual loss of fill-oil from the transmitter's sealed sensing module.

Discussion of Safety Significance:

The performance of a transmitter that is leaking fill-oil gradually deteriorates and may eventually lead to failure. Although some failed transmitters have shown symptoms of loss of fill-oil prior to failure, it has been reported that in some cases the failure of a transmitter that is leaking fill-oil is not detectable during operation. Loss of fill-oil may result in a transmitter not performing its intended safety function.

Discussion:

Model 1151, 1152, 1153, and 1154 Rosemount transmitters are utilized extensively in nuclear power plants. Model 1153 and 1154 transmitters are supplied by Rosemount as both seismically and environmentally qualified equipment. Model 1152 transmitters are supplied by Rosemount only as seismically qualified equipment. Model 1151 transmitters are supplied by Rosemount as commercial-grade equipment.

Rosemount has indicated, to date, that failure of approximately 91 Model 1153 and 1154 transmitters due to loss of fill-oil from a glass to metal seal failure have been confirmed. Since the sensing module is sealed, loss of fill-oil usually cannot be confirmed without destructive analysis of the sensing module. NRC staff review of this issue has identified additional failed Model 1153 and 1154 transmitters with symptoms indicative of loss of fill-oil that may not have been brought to Rosemount's attention. Thus, the number of failed Model 1153 and 1154 transmitters that have experienced a loss of fill-oil may be greater than that confirmed by Rosemount.

Rosemount has indicated that similar sensing modules are utilized in Model 1151, 1152, 1153, and 1154 transmitters and that failures of both Model 1151 and 1152 transmitters due to loss of fill-oil from a glass to metal seal failure have been confirmed. The NRC staff believes that, while Model 1153 and 1154 transmitters have a greater susceptibility to loss of fill-oil, Model 1151 and 1152 transmitters may also be susceptible to loss of fill-oil. Thus, loss of fill-oil may be generically applicable to Rosemount manufactured sensing modules. Accordingly, for the purposes of the actions requested in this bulletin, Model 1151 and 1152 transmitters utilized in safety-related systems should be addressed in a manner comparable to that of Model 1153 and 1154 transmitters. In addition, Rosemount has indicated that they have instituted additional quality control and quality assurance steps in the manufacturing process that they believe will minimize the potential for Model 1153 and 1154 transmitter failures due to loss of fill-oil. As a result, Rosemount has indicated that Model 1153 and 1154 transmitters manufactured after July 11, 1989 are not subject to their May 1989 10 CFR Part 21 notification. The NRC staff has not, to date, received indications that Model 1153 and 1154 transmitters manufactured by Rosemount subsequent to July 11, 1989 are susceptible to loss of fill-oil; therefore, the NRC staff concludes that Model 1153 and 1154 transmitters manufactured by Rosemount subsequent to July 11, 1989 are not subject to the actions requested in this bulletin. The NRC staff has not, to date, received sufficient information to address the applicability of these manufacturing process modifications to Model 1151 and 1152 transmitters.

Rosemount had previously indicated that Model 1153 and 1154 transmitters that were experiencing a loss of fill-oil should fail within approximately 36 months of in-service time. Recent information indicates that the rate at which fill-oil leaks is application and pressure dependant. Therefore, while transmitters that are experiencing a loss of fill-oil that are subject to continuous high-pressure (e.g. reactor operating pressures) may fail within this timeframe, transmitters utilized in low-pressure systems or not subject to continuous high-pressure may take longer to fail.

Rosemount has indicated that they manufacture both complete transmitters and transmitter parts (including sensing modules) for other manufacturers. At least one vendor purchases complete transmitters from Rosemount and then provides these transmitters for use in nuclear power plants under a different brandname. At least one other instrument manufacturer purchased Rosemount manufactured sensing modules and incorporated these sensing modules into transmitters supplied to nuclear power plants. Thus, equipment supplied for use in nuclear power plants by other manufacturers may also be susceptible to loss of fill-oil. In addition, Rosemount has indicated that unauthorized remanufacturers and refurbishers exist for Model 1151 and possibly Model 1152, 1153, and 1154 transmitters.

The symptoms a Model 1153 or 1154 transmitter may exhibit during normal operation if it is leaking fill-oil include:

- a slow setpoint drift of 1/4 of 1 percent per month
- deviation from the normal system signal fluctuation that is consistent in only the increasing or decreasing direction ("one-sided-noise")
- slow response to or inability to follow planned or unplanned plant transients
- a decrease in noise amplitude
- an output that deviates from that of redundant transmitters

The symptoms a Model 1153 or 1154 transmitter may exhibit during calibration activities if it is leaking fill-oil include:

- inability to respond over the entire design range
- slow response to either an increasing or decreasing test pressure
- drift of greater than 1 percent from the previous calibration

The NRC staff believes these symptoms can also be utilized to detect other transmitter models that may be experiencing a loss of fill-oil. In addition, addressees may wish to consult References 1, 2, 3, and 4 to obtain additional detailed technical information concerning loss of fill-oil. However, addressees are cautioned that the NRC staff has reviewed Reference 4 and concludes that, while Rosemount has provided sufficient bases to support their proposed diagnostic procedures (trending calibration data, trending operational data, sluggish transient response, and process noise analysis) for detecting whether a transmitter may be leaking fill-oil, Rosemount has not provided sufficient bases to support their proposed methodology for identifying which transmitters should be put into the enhanced surveillance program (pressure versus time-in-service and only Model 1153 and 1154 transmitters).

Certain manufacturing lots of Model 1153 and 1154 transmitters have been previously identified by Rosemount as having had a high failure fraction (on the order of 6%) due to loss of fill-oil. Specific information needed to identify transmitters that are from these suspect lots has been provided to industry by Rosemount. The NRC staff believes that transmitters from these suspect lots have an unacceptably high susceptibility to failure from loss of fill-oil and should not be utilized in the reactor protection or engineered safety features actuation systems.

General Design Criterion (GDC) 21 "Protection System Reliability and Testability" of 10 CFR 50, Appendix A requires the protection system to be designed for high functional reliability and with sufficient capability to allow periodic testing of its functioning when the reactor is in operation in order to readily detect failures of subcomponents and subsystems within the protection system as well as loss of the required protection system redundancy as they occur. 10 CFR 50.55a(h) requires that protection systems meet the Institute of Electrical and Electronics Engineers Standard: "Criteria for Protection Systems for Nuclear Power Generating Stations" (IEEE-279). IEEE-279 states that means shall be provided for checking, with a high degree of confidence, the operational availability of each system input sensor during reactor operation. Thus, the NRC staff concludes that facilities that utilize transmitters that may be susceptible to loss of fill-oil may not be in full compliance with these regulations because undetected transmitter failure could occur. Accordingly, the NRC staff requests that addressees take the actions requested below.

Requested Actions:

Operating Reactors

1. Identify, within 60 days after the receipt of this bulletin, all pressure or differential pressure transmitters, including Model 1151, 1152, 1153, and 1154 transmitters but excluding Model 1153 and 1154 transmitters manufactured by Rosemount subsequent to July 11, 1989, that were manufactured by Rosemount or that contain Rosemount manufactured sensing modules and are utilized in either safety-related systems or systems installed in accordance with 10 CFR 50.62 (the ATWS rule). Addressees may find it necessary to perform, in addition to document reviews, system walkdowns to complete this action. In addition, the following information is provided to facilitate addressee's activities in this area.

   All Model 1153 and 1154 transmitters, whether obtained directly from Rosemount, obtained through intermediary suppliers, or provided as an integral part of another component (such as an emergency diesel generator), should a) indicate manufacture by Rosemount, b) have a distinctive Rosemount model and serial number, c) have the physical profile characteristics of a Rosemount transmitter, and d) have a blue or stainless steel housing. Rosemount has indicated that Model 1153 and 1154 transmitters are not provided to other manufacturers for resale under a different brandname. In addition, a simplified diagram that describes the typical physical characteristics of a Rosemount transmitter is provided by Attachment 1.

Model 1152 transmitters, except as noted below, should a) indicate manufacture by Rosemount, b) have a distinctive Rosemount model and serial number, c) have the physical profile characteristics of a Rosemount transmitter, and d) have a blue or stainless steel housing. Rosemount has indicated that they have supplied Model 1152 transmitter sensing modules to Bailey Controls (formerly Bailey Meter). Bailey manufactured transmitters that contain Rosemount manufactured Model 1152 sensing modules have gray housings that appear slightly different (more rounded) than Rosemount housings.

Model 1151 transmitters, except as noted below, should a) indicate manufacture by Rosemount, b) have a distinctive Rosemount model and serial number, c) have the physical profile characteristics of a Rosemount transmitter, and d) have a blue housing. Model 1151 transmitters manufactured by Rosemount may have been supplied for use in nuclear power plants by other original equipment manufacturers (OEM's). The OEM's identified in Attachment 2 may offer for resale under their own brandname Model 1151 transmitters purchased from Rosemount. These transmitters should have the physical profile characteristics of a Rosemount transmitter and have a blue housing. Fisher Controls may also offer for resale under their own brandname Model 1151 transmitters purchased from Rosemount. These transmitters should have the physical profile characteristics of a Rosemount transmitter, but have a green housing. In addition, Rosemount has indicated that they have supplied Model 1151 transmitter sensing modules to Bailey Controls. Bailey manufactured transmitters that contain Rosemount manufactured Model 1151 sensing modules have gray housings that appear slightly different (more rounded) than Rosemount housings.

2. Review, within 90 days after receipt of this bulletin, plant records (for example, calibration records) associated with the transmitters identified in Item 1 above to determine whether any of these transmitters may have already exhibited symptoms indicative of loss of fill-oil. Appropriate operability acceptance criteria should be developed and applied to transmitters identified as having exhibited symptoms indicative of loss of fill-oil from this plant record review. Transmitters identified as having exhibited symptoms indicative of loss of fill-oil that do not conform to the operability acceptance criteria should be addressed in accordance with the applicable technical specification. Transmitters identified as having exhibited symptoms indicative of loss of fill-oil that do not conform to the operability acceptance criteria and are not addressed in the technical specifications should be replaced at the earliest appropriate opportunity.

3. Develop and implement, within 120 days after receipt of this bulletin, an enhanced surveillance program to monitor transmitters identified in Item 1 for symptoms of loss of fill-oil. This enhanced surveillance program should consider the following or equally effective actions:

a) Ensuring appropriate licensee personnel are aware of the symptoms that a transmitter, both during operation and during calibration activities, may exhibit if it is experiencing a loss of fill-oil and the need for prompt identification of transmitters that may exhibit these symptoms;

b) Enhanced transmitter monitoring to identify excessive transmitter drift;

c) Review of transmitter output data following planned or unplanned plant transients or tests to identify sluggish transmitter response;

d) Inclusion of sensor response time testing into routine channel calibration activities;

e) Development and implementation of a program to detect a decrease in transmitter noise level amplitude; and

f) Development and application to transmitters identified as having exhibited symptoms indicative of loss of fill-oil of an appropriate operability acceptance criteria. Transmitters identified as having exhibited symptoms indicative of loss of fill-oil that do not conform to the operability acceptance criteria should be addressed in accordance with the applicable technical specification. Transmitters identified as having exhibited symptoms indicative of loss of fill-oil that do not conform to the operability acceptance criteria and are not addressed in the technical specifications should be replaced at the earliest appropriate opportunity.

4. Determine, within 60 days after receipt of this bulletin, whether any Model 1153 or 1154 transmitters identified in Item 1 are from the manufacturing lots that have been identified by Rosemount as having a high failure fraction due to loss of fill-oil. (Information concerning these transmitters was provided to industry by Rosemount concurrent with Reference 4). Addressees are requested not to utilize transmitters from these suspect lots in the reactor protection or engineered safety features actuation systems; therefore, transmitters from these suspect lots in use in the reactor protection or engineered safety features actuation systems should be replaced at the earliest appropriate opportunity.

5. Document, within 60 days after receipt of this bulletin, and maintain in accordance with plant procedures a basis for continued plant operation covering the time period from the present until such time that the Model 1153 and 1154 transmitters from the manufacturing lots that have been

identified by Rosemount as having a high failure fraction due to loss of fill-oil in use in the reactor protection or engineered safety features actuation systems can be replaced. In addition, while performing the actions requested above, addressees may identify transmitters exhibiting symptoms indicative of loss of fill-oil that do not conform to the established operability acceptance criteria and are not addressed in the technical specifications. As these transmitters are identified, this basis for continued plant operation should be updated to address these transmitters covering the time period from the time these transmitters are identified until such time that these transmitters can be replaced. When developing and updating this basis for continued plant operation, addressees may wish to consider transmitter diversity and redundancy, diverse trip functions (a separate trip function that may also provide a corresponding trip signal), special system and/or component tests, or (if necessary) immediate replacement of certain suspect transmitters.

Construction Permit Holders

1. All construction permit holders are requested to complete Items 1 and 3 of Requested Actions for Operating Reactors prior to the date scheduled for fuel loading or in accordance with the timeframes specified for Operating Reactors, whichever is later.

2. All construction permit holders that are completing Items 1 and 3 of Requested Actions for Operating Reactors in accordance with the timeframes specified for Operating Reactors are requested to complete Items 4 and 5 of Requested Actions for Operating Reactors in accordance with the timeframes specified for Operating Reactors.

3. All construction permit holders that are completing Items 1 and 3 of Requested Actions for Operating Reactors prior to the date scheduled for fuel loading are requested to address the intent of Items 4 and 5 of Requested Actions for Operating Reactors by performing the following actions:

   a) Identify and replace, prior to the date scheduled for fuel loading, any Model 1153 or 1154 transmitters from the manufacturing lots that have been identified by Rosemount as having a high failure fraction due to loss of fill-oil that are installed in the reactor protection or engineered safety features actuation systems; and

   b) Document and maintain in accordance with plant procedures a basis for continued plant operation that addresses transmitters that, subsequent to fuel loading, are identified as exhibiting symptoms indicative of loss of fill-oil that do not conform to the established operability acceptance criteria and are not addressed in the technical specifications covering the time period from the time these transmitters are identified until such time that these transmitters can be replaced. When developing and updating this basis for continued plant operation, addressees may wish to consider transmitter diversity and redundancy, diverse trip functions (a separate trip function that may also provide a corresponding trip signal), special system and/or component tests, or (if necessary) immediate replacement of certain suspect transmitters.

Reporting Requirements:

Operating Reactors

1. Provide, within 120 days after receipt of this bulletin, a response that:

   a) Confirms that those Requested Actions for Operating Reactors in Items 1, 2, 3, 4, and 5 that are to be completed within 120 days after receipt of this bulletin have been completed and that programs are in place to perform the remaining requested actions;

   b) Identifies the indicated manufacturer; the model number; the safety-related system the transmitter was utilized in; the approximate amount of time in service; the corrective actions taken; and the disposition (e.g., returned to vendor for analysis) of transmitters, including those identified while performing Item 2 of Requested Actions for Operating Reactors above, that are believed to have exhibited symptoms indicative of loss of fill-oil or have been confirmed to have experienced a loss of fill-oil; and

   c) Identifies the safety-related system in which the Model 1153 or 1154 transmitters from the manufacturing lots that have been identified by Rosemount as having a high failure fraction due to loss of fill-oil are utilized and provides a schedule for replacement of these transmitters which are in use in the reactor protection or engineered safety features actuation systems.

2. Transmitters that, subsequent to providing the response required by Item 1 above, exhibit symptoms of loss of fill-oil or are confirmed to have experienced a loss of fill-oil should be reviewed for reportability under existing NRC regulations. If determined not to be reportable, addressees are requested to document and maintain, in accordance with plant procedures, information consistent with that requested in Item 1 b) above for each suspect transmitter identified.

Construction Permit Holders

1. All holders of construction permits that perform Items 1 and 3 of Requested Actions for Operating Reactors in accordance with the timeframes specified for Operating Reactors should provide, within 120 days after receipt of this bulletin, a response that:

   a) Confirms that those Requested Actions for Operating Reactors in Items 1, 3, 4, and 5 that are to be completed within 120 days after receipt of this bulletin have been completed and that programs are in place to perform the remaining requested actions; and

   b) Identifies the safety-related system in which the Model 1153 or 1154 transmitters from the manufacturing lots that have been identified by Rosemount as having a high failure fraction due to loss of fill-oil are utilized and provides a schedule for replacement of these transmitters which are in use in the reactor protection or engineered safety features actuation systems.

2. All holders of construction permits that perform Items 1 and 3 of Requested Actions for Operating Reactors prior to the date scheduled for fuel loading should provide, prior to the date scheduled for fuel loading, a response that:

   a) Confirms that all actions in Items 1 and 3 of Requested Actions for Operating Reactors have been completed; and

   b) Confirms that Model 1153 or 1154 transmitters from the manufacturing lots that have been identified by Rosemount as having a high failure fraction due to loss of fill-oil are not utilized in the reactor protection or engineered safety features actuation systems.

3. Transmitters that, subsequent to providing the response required by Item 1 or 2 above, exhibit symptoms of loss of fill-oil or are confirmed to have experienced a loss of fill-oil should be reviewed for reportability under existing NRC regulations. If determined not to be reportable, addressees are requested to document and maintain, in accordance with plant procedures, information consistent with that requested in Item 1 b) of the Reporting Requirements for Operating Reactors above for each suspect transmitter identified.

The written reports required above shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, and shall be submitted under oath or affirmation pursuant to the provisions of Section 182a, Atomic Energy Act of 1954, as amended and 10 CFR 50.54(f). In addition, a copy shall be submitted to the appropriate Regional Administrator.

Backfit Discussion

The objective of the actions requested in this bulletin are to ensure that transmitter failures due to loss of fill-oil are promptly detected. Loss of fill-oil may result in a transmitter not performing its intended safety function.

The actions requested in this bulletin represent new staff positions and thus, this request is considered a backfit in accordance with NRC procedures. Because established regulatory requirements exist but were not satisfied, this backfit is to bring facilities into compliance with existing requirements. Therefore, a full backfit analysis was not performed. An evaluation of the type discussed in 10 CFR 50.109(a)(6) was performed, including a statement of the objectives of and reasons for the modification and the basis for invoking the compliance exception. It will be made available in the Public Document Room with the minutes of the _____ meeting of the Committee to Review Generic Requirements.

This request is covered by Office of Management and Budget Clearance Number 3150-0011 which expires January 31, 1991. The estimated average burden hours are 100 person-hours per licensee response, including assessment of the new requested actions, searching data sources, gathering and analyzing the data, and preparing the required letters. These estimated average burden hours pertain only to these identified response-related matters and do not include the time for actual implementation of the requested actions. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch, Division of Information Support Services, Office of Information Resources Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Paperwork Reduction Project (3150-0011), Office of Management and Budget, Washington D.C. 20503.

If you have any questions about this matter, please contact one of the technical contacts listed below or the appropriate NRR project manager.

Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Jack Ramsey, NRR
(301) 492-1167

Vince Thomas, NRR
(301) 492-0786

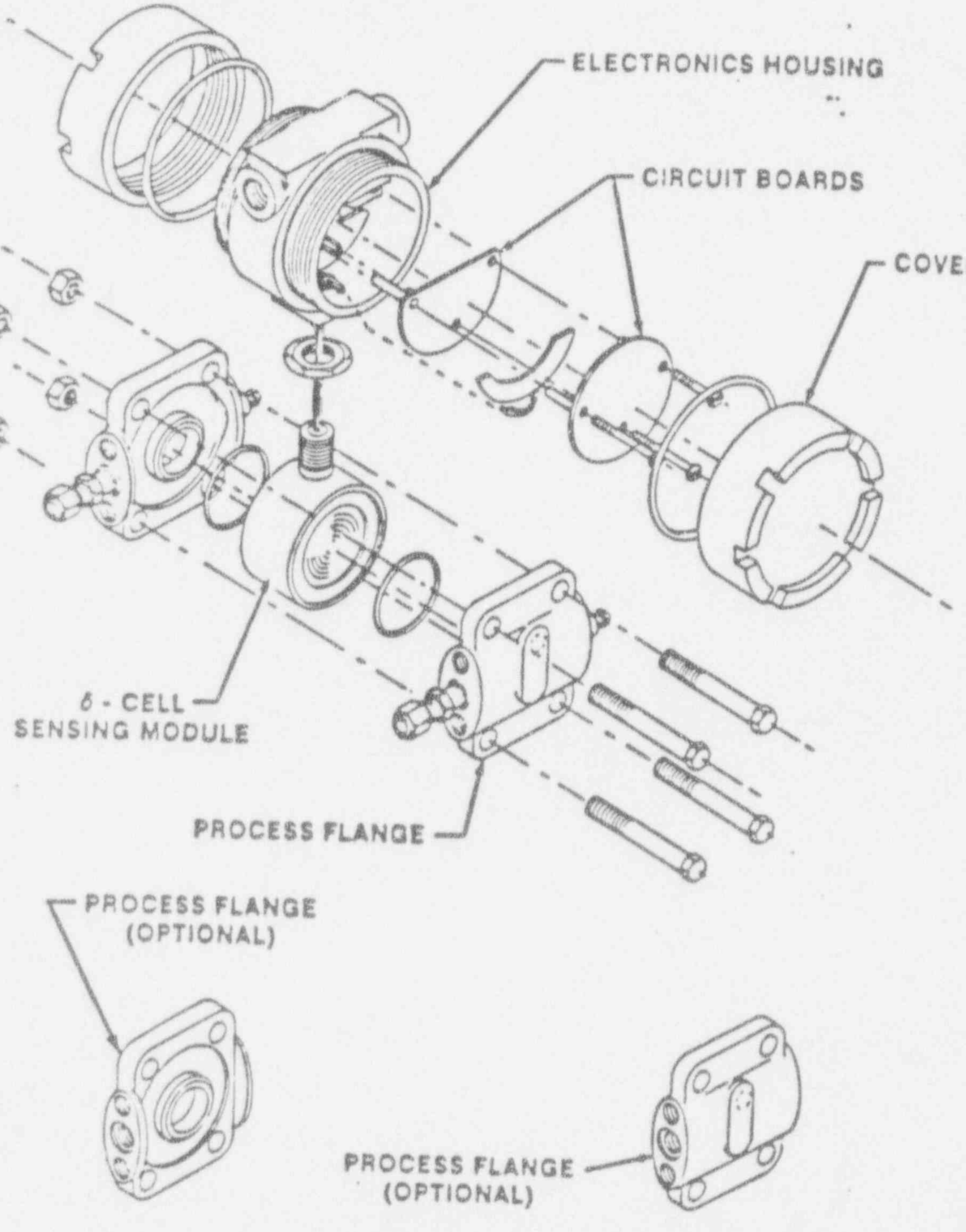
References:

1. Rosemount Technical Bulletin No. 1 dated May 10, 1989
2. Rosemount Technical Bulletin No. 2 dated July 12, 1989
3. Rosemount Technical Bulletin No. 3 dated October 23, 1989
4. Rosemount Technical Bulletin No. 4 dated December 22, 1989

Attachment 1: Typical Physical Characteristics of a Rosemount Transmitter
Attachment 2: Original Equipment Manufacturers That May Resell Rosemount Manufactured Model 1151 Transmitters Under Their Own Brandname
Attachment 3: List of Recently Issued NRC Bulletins

Attachment 1
NRCB 90-XX
January xx, 1990

TYPICAL PHYSICAL CHARACTERISTICS OF A ROSEMOUNT TRANSMITTER



ORIGINAL EQUIPMENT MANUFACTURERS THAT MAY RESELL
ROSEMOUNT MANUFACTURED MODEL 1151 TRANSMITTERS
UNDER THEIR OWN BRANDNAME

1. FISHER CONTROLS
2. BAILEY CONTROLS (FORMERLY BAILEY METER)
3. DIETRICH STANDARD
4. DANIEL INDUSTRIES
5. CLEVELAND CONTROLS
6. F. B. LEOPOLD
7. HAYS REPUBLIC
8. MOORE PRODUCTS
9. NORTH AMERICAN MANUFACTURING
10. OMEGA ENGINEERING
11. LEEDS & NORTHRUP

CRGR Item IV.B. Contents of Packages Submitted to CRGR (Rev. 4, Stello to List 042387, dcs 41860 342 ff)

The following requirements apply for proposals to reduce existing requirements or (regulatory) positions as well as proposals to increase requirements or (regulatory) positions. Each package submitted to the CRGR for review shall include twenty (20) copies of the following information:

SUBJECT: PROPOSED BULLETIN ON LOSS OF FILL-OIL IN TRANSMITTERS MANUFACTURED BY ROSEMOUNT

Question:

I. The proposed generic requirement or staff position as it is proposed to be sent out to licensees.

Response:

The proposed staff position is set forth in the bulletin (Enclosure 1).

Question:

II. Draft staff papers or other underlying staff documents supporting the requirements or staff position. (A copy of all materials referenced in the document shall be made available upon request to the CRGR staff. Any committee member may request CRGR staff to obtain a copy of any referenced material for his or her use.)

Response:

1. NRC Information Notice 89-42 "Failure of Rosemount Models 1153 and 1154 Transmitters," April 21, 1989.

2. Memorandum from C.H. Berlinger to C.E. Rossi dated January 11, 1990 (Meeting minutes of December 1989 meeting with Rosemount).

3. Memorandum from C.H. Berlinger to C.E. Rossi dated September 12, 1989 (Meeting minutes of August 1989 meeting with Rosemount).

4. Memorandum from C.H. Berlinger to C.E. Rossi dated May 10, 1989 (Meeting minutes of April 1989 meeting with Rosemount).

5. Rosemount 10 CFR Part 21 notification dated May 12, 1989.

Question:

III. Each proposed requirement or staff position shall contain the sponsoring office's position as to whether the proposal would increase staff requirements or staff positions, or would implement existing requirements or staff positions.

Response:

A. Transmitters manufactured by Rosemount are utilized extensively in safety-related systems in nuclear power plants. The performance of a transmitter that is leaking fill-oil gradually deteriorates and may eventually lead to failure. Although some failed transmitters have shown symptoms of loss of fill-oil prior to failure, the failure of a transmitter that is leaking fill-oil will very likely not be detectable during operation. Loss of fill-oil may result in a transmitter (or transmitters) not performing its (their) intended safety function. Accordingly, the staff has determined that it is necessary to request that licensees and construction permit holders for nuclear power reactors perform the actions requested in the bulletin.

B. The actions requested in the proposed bulletin would ensure compliance with General Design Criterion (GDC) 21 of 10 CFR Part 50, Appendix A and with 10 CFR 50.55a(h) related to protection system reliability and testability. These requirements state that transmitters must be reliable and that means must be provided to check and test for failures when the reactor is in operation.

Question:

IV. The proposed method of implementation along with the concurrence (and any comments) of OGC on the method proposed.

Response:

The method of implementation will be the proposed bulletin (Enclosure 1). A copy of this bulletin has been reviewed by OGC. OGC's comments have been incorporated. OGC has no legal objection.

Question:

V. Regulatory analysis generally conforming to the directives and guidance of NUREG/BR-0058 and NUREG/CR-3568.

Response:

This is a compliance issue. In accordance with 10 CFR 50.109, a formal value/impact analysis was not performed.

Question:

VI. Identification of the category of reactor plants to which the generic requirement or staff position is to apply (that is, whether it is to apply to new plants only, new OLs [operating licensees] only, OLs after a certain date, all OLs, all plants under construction, all plants, all water reactors, all PWRs [pressurized water reactors] only, some vendor types, some vintage types such as BWR 6 and 4, jet pump and nonjet pumps plants, etc).

Response:

The proposed bulletin would apply to all holders of operating licenses or construction permits for nuclear power reactors.

Question:

VII. For each category of reactor plants, an evaluation which demonstrates how the action should be prioritized and scheduled in light of other ongoing regulatory activities. The evaluation shall document for consideration information available concerning any of the following factors as may be appropriate and any other information relevant and material to the proposed action:

A. Statement of the specific objectives that the proposed action is designed to achieve....

Response:

The primary objective of the proposed bulletin is to ensure that addressees promptly detect and replace transmitters that may be leaking fill-oil.

Question:

B. General description of the activity that would be required by licensees in order to complete the action...

Response:

To complete the action, addressees would need to identify transmitters that were manufactured by Rosemount or that contain Rosemount manufactured sensing modules and are utilized in either safety-related systems or systems installed in accordance with 10 CFR 50.62 (the ATWS rule), review plant records to determine whether any of the identified transmitters may already be exhibiting symptoms indicative of loss of fill-oil, develop and implement an enhanced surveillance program to monitor the identified transmitters for symptoms indicative of loss of fill-oil, and replace transmitters that are either identified as having exhibited symptoms of loss of fill-oil and do not meet the operability acceptance criteria or are from the suspect manufacturing lots and are installed in the reactor protection or engineered safety features actuation systems.

Question:

C. Potential change in risk to the public, from the accidental offsite release of radioactive material....

Response:

The proposed bulletin would ensure that the risk to the public from the accidental offsite release of radioactive material is consistent with that intended by the promulgation of the previously identified GDC's and the design bases of the FSAR.

Question:

D. Potential impact on radiological exposure of facility employees and other onsite workers....

Response:

Certain activities associated with the actions requested in the bulletin do not necessitate access to the transmitters themselves; therefore, no increase in radiological exposure of facility employees and other onsite workers is expected for these activities.

Certain activities associated with the actions requested in the bulletin may necessitate access to the transmitters themselves. However, since the transmitters are normally accessed every refueling outage for calibration related activities, the additional radiological exposure of facility employees and other onsite workers resulting from this bulletin is expected to be minimal.

Question:

E. Installation and continuing costs associated with the action, including the cost of facility downtime or construction delay...

Response:

The NRC staff's estimate of the impact of the actions requested in the proposed bulletin is as follows:

There are approximately 6,000 Rosemount transmitters (Models 1151, 1152, 1153 and 1154) in safety-related NPRDS reportable systems. The staff estimates that general record review for these transmitters will necessitate approximately 3 staff-hours per transmitter. Assuming a rate of $100 per staff-hour results in an impact of approximately $1,800,000 to industry.

Assuming records for 1,000 transmitters may need a more detailed review results in an additional impact of approximately $300,000 to industry.

It is the staff's understanding that the cost of replacing a safety-related transmitter is on the order of $5,000 to $10,000 per transmitter. Assuming 250 safety-related transmitters will need to be replaced results in an impact of approximately $1,250,000 to $2,500,000 to industry.

It is the staff's understanding, based upon feedback from industry, that an effective transmitter monitoring program can be developed and implemented for approximately $10,000 to $100,000 per facility. Assuming the actions requested in the proposed bulletin affect 110 facilities, this results in an impact of approximately $1,100,000 to $11,000,000 to industry.

Thus, the staff concludes that the actions requested in the proposed bulletin will result in an impact of from approximately $4,450,000 to $15,600,000 to industry. The impact to an individual plant will vary significantly and will be strongly influenced by the number of potentially affected transmitters utilized and by the number of transmitters that need to be replaced.

Question:

F. The potential safety impact of changes in plant or operational complexity, including the relationship to proposed and existing regulatory requirements and staff positions...

Response:

A slight increase in operational complexity may, due to corrective actions taken, result from transmitters that are identified as exhibiting symptoms indicative of loss of fill-oil. However, this increase in operational complexity is not as significant as the operational problems that may arise should a transmitter fail to perform its safety function.

Question:

G. The estimated resource burden on the NRC associated with the proposed action and the availability of such resources....

Response:

The bulletin does require that licensees provide a written response that identifies transmitters that are believed to have exhibited symptoms indicative of loss of fill-oil or have been confirmed to have experienced a loss of fill-oil, as well as confirming that actions consistent with those requested in the bulletin have been or will be taken. The responses will assist the NRC staff in determining transmitter failure rates, as well as confirming actions taken by addressees are consistent with those requested in the bulletin. The impact of reviewing this information on the NRC staff is not expected to be significant. In addition, no requirement for regional review is anticipated.

Question:

H. The potential impact of differences in facility type, design or age on the relevancy and practicality of the proposed action....

Response:

Based on the currently available information, no differences are expected.

Question:

I. Whether the proposed action is interim or final, and if interim, the justification for imposing the proposed action on an interim basis.

Response:

The NRC staff believes that the actions requested in the proposed bulletin are needed to ensure that addressees promptly detect and replace transmitters that may be leaking fill-oil. However, continuing NRC staff review of this and related issues may indicate the need for further regulatory action.

Question:

VIII. For each evaluation conducted pursuant to 10 CFR 50.109, the proposing Office Director's determination together with the rationale for the determination based on the consideration of paragraphs I. through VII. above that:

A. There is a substantial increase in the overall protection of public health and safety or the common defense and security to be derived from the proposal.

B. The direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection.

Response:

The actions requested in this bulletin would ensure that licensees comply with those commitments pursuant to their license which require that safety-related equipment perform their intended function when required to do so. Therefore, under 10 CFR 50.109(a)(4)(i) no further cost analysis is required.

Question:

IX. For each evaluation conducted for proposed relaxations or decreases in current requirements or staff positions, the proposing Office Director's determination, together with the rationale for the determination based on the considerations of paragraphs I. through VII. above that:

A. The public health and safety and the common defense and security would be adequately protected if the proposed reduction in requirements or positions were implemented.

B. The cost savings attributed to the action would be substantial enough to justify taking the action.

Response:

No relaxation of requirements will occur from this bulletin.

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

ACTION

HELIGMES

JAN 10 1990

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

FROM: James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: PROPOSED GENERIC LETTER TO CORRECT A DEFICIENCY IN THE BWR CRITICAL POWER RATIO CALCULATION INVOLVING CHANNEL BOX BOW

NRR requests that the Committee to Review Generic Requirements (CRGR) review the enclosed proposed generic letter at the CRGR's earliest convenience. We would like to issue this generic letter soon in order to facilitate the proposed licensee implementation date.

The proposed generic letter would implement existing requirements and staff positions in that it would correct a known deficiency in calculating the critical power ratio (CPR). This deficiency results from greater than expected BWR channel box bow effect which reduces the CPR to lower values than predicted by calculations. If this deficiency is not corrected in a timely manner, it could mislead operators about the technical specifications operating limit margin and, in the worst case, could result in violations of the CPR safety limit with potential fuel failure. The deficiency became known as a result of dryout with fuel failures in a foreign BWR.

This generic letter is addressed to all holders of operating licenses or construction permits for BWRs and is sponsored by Ashok Thadani, Director, Division of Systems Technology.

The proposed generic letter and background information required by the CRGR charter are enclosed.

James H Sniezek

James H. Sniezek, Deputy Director
Office of Nuclear Reactor Regulation

Enclosures:
As stated

CONTACT: Peter C. Wen, NRR
492-1172

9001170496 XA

Enclosure

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

TO: ALL HOLDERS OF OPERATING LICENSES OR CONSTRUCTION PERMITS FOR BOILING-WATER REACTORS (BWRs)

SUBJECT: CORRECTION OF DEFICIENCY IN BWR CRITICAL POWER RATIO CALCULATION DUE TO CHANNEL BOX BOW

As a result of information obtained at a meeting on fuel failures caused by dryout at a foreign BWR facility and at meetings with BWR fuel vendors, the NRC issued Information Notice 89-69 on September 29, 1989, to alert addressees to the potential problems associated with excessive channel box bow that could result in a loss of thermal margin. The generic concern applicable to the U.S. BWRs is that the channel box bow effect is not properly taken into account in the critical power ratio (CPR) calculation. Each BWR fuel vendor plans to submit a methodology to address this concern. A copy of NRC Information Notice 89-69, "Loss of Thermal Margin Caused by Channel Box Bow," is attached.

Because the possibility of fuel failure as a result of channel box bow exceeding that assumed in the CPR calculation is remote, the staff has agreed that no immediate corrective actions are needed. However, in order to verify compliance with the current licensing basis, the staff has concluded that the validity of the CPR calculation used to ensure conformance to the plant technical specifications should be evaluated and corrected, if necessary, in a timely manner. Therefore, the staff is requesting that each operating reactor licensee ensure that the effects of channel box bow on the CPR calculation are properly taken into account in the first fuel reload scheduled after April 30, 1990. This may be accomplished by either of the following two options:

A. By ensuring that the procedures for monitoring thermal limits during reactor operation will prevent violation of the minimum CPR operating limit by the use of an NRC approved methodology that takes into account the new data on fuel channel bowing, or

B. By imposing on the operating CPR limit (or limiting curve) a channel bow penalty of either 0.03 delta-CPR for D-lattice plants or 0.02 delta-CPR for C (or S) lattice plants and by removing any channel boxes that are being reused after their first bundle lifetime.

All construction permit (CP) holders are requested to complete the above actions before the date scheduled for fuel loading.

To determine if any license or construction permit for facilities covered by this request should be modified, suspended, or revoked, the staff requires, pursuant to Section 182 of the Atomic Energy Act and 10 CFR 50.54(f), that each operating reactor licensee advise the NRC by letter (1) whether it will implement one of the options listed for the first reload scheduled after April 30, 1990, including the date of the reload, and (2) whether or not channel boxes are being used for a second bundle lifetime. If the licensee does not intend to adopt either option A or B above, it should include in its response a justification for its position. The NRC staff requires each licensee to provide a response within 60 days of receipt of this letter. Before fuel loading, CP holders for BWRs are requested to advise the NRC in writing that the actions requested in this letter have been implemented.

The written response required above shall be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, a copy shall be submitted to the appropriate Regional Administrator.

The actions requested in this generic letter ensure licensees' compliance with their technical specifications as required under 10 CFR 50.36(c)(1)(i)(A). The CPR is a safety limit under this regulation. A known deficiency in calculating the CPR must be corrected to remain in compliance with the licensing basis. If this deficiency is not corrected in a timely manner, it could mislead operators about the technical specifications operating limit margin and, in the worst case, could result in violations of the CPR safety limit with potential fuel failure. The requested action was evaluated consistent with the provisions of 10 CFR 50.109 and found to be covered by the provisions of paragraph (a)(4)(i).

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires January 31, 1991. The estimated average burden hours is 100 person-hours per licensee response, including assessment of the new recommendations, searching data sources, gathering and analyzing the data, and preparing the required letters. These estimated average burden hours pertain only to the identified response-related matters and do not include the time for actual implementation of the requested actions. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for reducing this burden, to the Information and Records Management Branch, Division of Information Support Services, Office of Information Resources Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Paper Reduction Project (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

If you have any questions about the actions requested in this generic letter, please contact one of the technical contacts listed below or the appropriate NRR project manager.

Sincerely,

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. NRC Information Notice 89-69, "Loss of Thermal Margin Caused by Channel Box Bow "
2. List of Recently Issued NRC Generic Letters

Technical Contacts: Peter C. Wen, NRR
(301) 492-1172

Daniel B. Fieno, NRR
(301) 492-3236

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

September 29, 1989

NRC INFORMATION NOTICE NO. 89-69: LOSS OF THERMAL MARGIN CAUSED BY CHANNEL BOX BOW

Addressees:

All holders of operating licenses or construction permits for boiling-water reactors (BWRs).

Purpose:

This information notice is intended to alert addressees to potential problems involving loss of thermal margin caused by excessive bowing of BWR fuel channel boxes. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

During a refueling outage in August 1988, four failed fuel rods in separate assemblies were identified at a foreign BWR facility. Subsequent evaluation of sipping, visual inspection, gamma scan, and hot-cell data led to the conclusion that these rods failed because the rods were operated under dryout conditions during steady-state operation for an extended period of time (between 2 and 7 days).

The failed fuel rods were located symmetrically in the core. The fuel assemblies containing the rods that had failed were located adjacent to once-burned fuel assemblies with highly exposed fuel channels (see Figure 1). These fuel channels were in their second bundle lifetime and had excessive channel bowing. In each assembly with failed fuel, the corner rod facing the adjacent control rod was heavily oxidized and the cladding was penetrated just below the top spacer grid. In addition, each of the four failed rods had typical secondary internal hydriding damage near the bottom of the fuel rods, resulting in loss of fuel material.

Discussion:

Dryout of the fuel rods in this foreign facility occurred because of modeling errors in the plant process computer, which resulted in nonconservative calculated values of the minimum critical power ratio (MCPR) of the core. These modeling errors were caused by neglecting the effects of channel bowing and the geometric variation between the reloaded and once-burned fuel assemblies. These effects substantially increased the widths of the control rod water gaps for the assemblies that contained these four fuel rods beyond that assumed in the plant process computer calculations. The increased neutron moderation associated with the increased water gap widths led to very high localized power peaking at these four fuel rods. However, these effects were not properly accounted for in the MCPR calculations. For some time, the plant operators were misled by these erroneous MCPR calculations and were operating the plant in steady-state beyond the MCPR safety limit.

The modeling error of generic concern to all BWRs, regardless of the fuel supplier, relates only to the greater-than-expected bowing of fuel channel boxes, which contributed about 15 percent error in the calculated MCPR value for this foreign facility. Channel bowing is a manifestation of differences in the channel growth of opposite sides of the channel box and is proportional to channel growth. The information obtained by the NRC indicates that the channel growth shows an accelerated trend at higher burnup exposure, especially when the fuel channels are being reused in their second bundle lifetime. The effect on core operating MCPR is magnified when fresh fuel is located adjacent to the bowed fuel channels. Core operating limits imposed by technical specifications may be exceeded if the reduction in margin caused by fuel channel bowing is not properly accounted for in the plant process computer for thermal limits monitoring. Based on a preliminary evaluation by BWR fuel vendors of U.S. reactors, the impact of the new data on actual versus calculated MCPR values is expected to range from 0.0 to 0.03 CPR units. However, the impact could be much greater (about 15 percent) for any reactors operating with fuel channels being reused in their second bundle lifetime.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate NRR project manager.

Charles E. Rossi

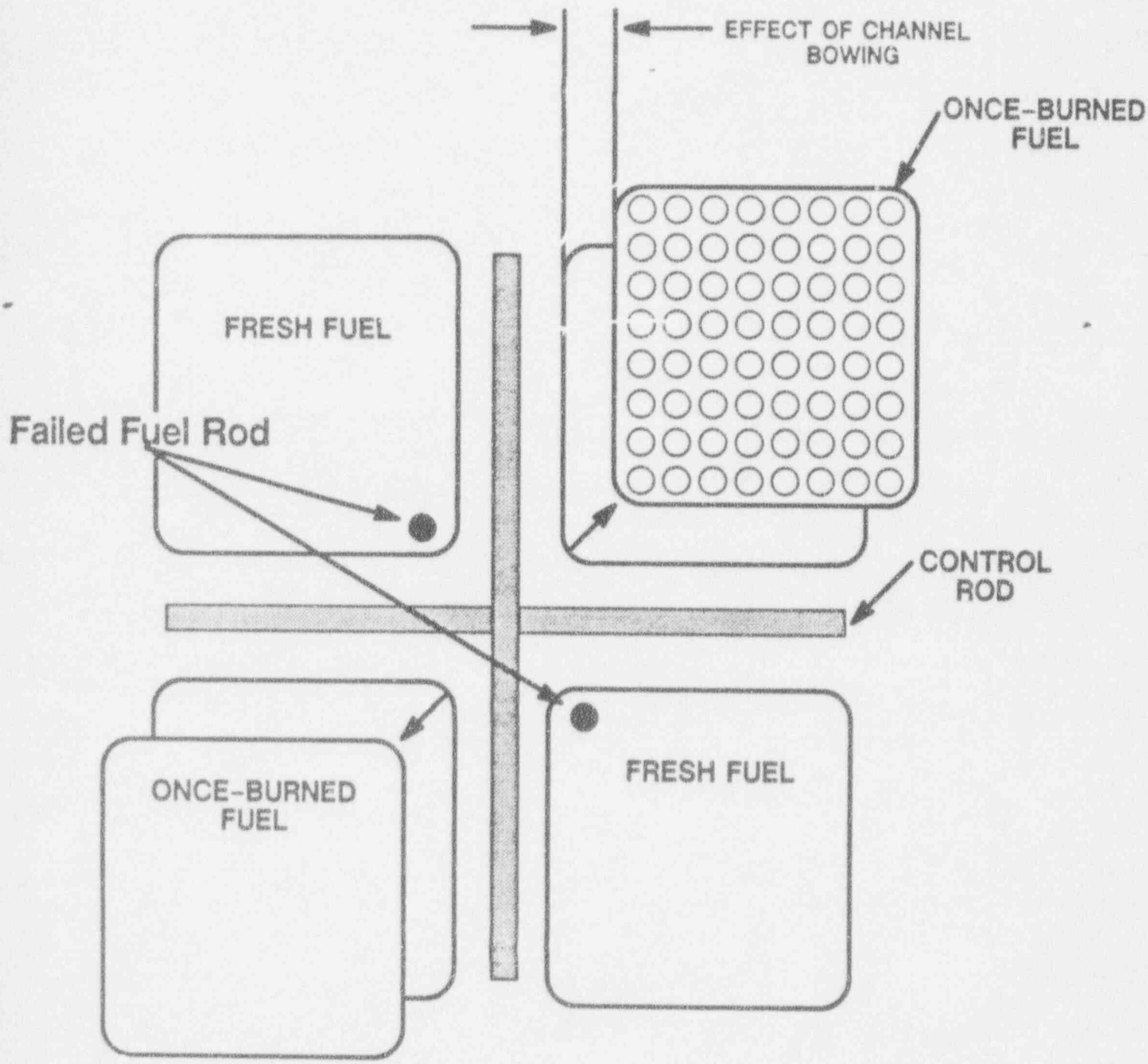
Charles E. Rossi, Director
Division of Operational Events Assessment
Office of Nuclear Reactor Regulation

Technical Contacts: Peter C. Wen, NRR
(301) 492-1172

Daniel B. Fieno, NRR
(301) 492-3236

Attachments:
1. Figure 1, "Channel Bow"
2. List of Recently Issued NRC Information Notices

Figure 1
Channel Bow



LIST OF RECENTLY ISSUED
NRC INFORMATION NOTICES

Information Notice No.	Subject	Date of Issuance	Issued to
89-68	Evaluation of Instrument Setpoints During Modifications	9/25/89	All holders of OLs or CPs for nuclear power reactors.
89-67	Loss of Residual Heat Removal Caused by Accumulator Nitrogen Injection	9/13/89	All holders of OLs or CPs for PWRs.
89-66	Qualification Life of Solenoid Valves	9/11/89	All holders of OLs or CPs for nuclear power reactors.
88-46, Supp. 4	Licensee Report of Defective Refurbished Circuit Breakers	9/11/89	All holders of OLs or CPs for nuclear power reactors.
89-65	Potential for Stress Corrosion Cracking in Steam Generator Tube Plugs Supplied by Babcock and Wilcox	9/8/89	All holders of OLs or CPs for PWRs.
89-64	Electrical Bus Bar Failures	9/7/89	All holders of OLs or CPs for nuclear power reactors.
89-63	Possible Submergence of Electrical Circuits Located Above the Flood Level Because of Water Intrusion and Lack of Drainage	9/5/89	All holders of OLs or CPs for nuclear power reactors.
89-62	Malfunction of Borg-Warner Pressure Seal Bonnet Check Valves Caused By Vertical Misalignment of Disk	8/31/89	All holders of OLs or CPs for nuclear power reactors.
89-61	Failure of Borg-Warner Gate Valves to Close Against Differential Pressure	8/30/89	All holders of OLs or CPs for nuclear power reactors.

OL = Operating License
CP = Construction Permit

CRGR REVIEW PACKAGE

RESPONSE TO REQUIREMENTS FOR CONTENT OF PACKAGE SUBMITTED FOR CRGR REVIEW

(i) The proposed generic requirement or staff position as it is proposed to be sent out to licensees.

* The staff position is provided in the proposed Generic Letter that will be sent to all boiling water reactor (BWR) licensees and applicants. It informs licensees and applicants of the loss of thermal margin caused by channel box bow and requires actions to be taken to prevent potential violations of the safety limit minimum critical power ratio (SLMCPR).

(ii) Draft staff papers or other underlying staff documents supporting the requirements or staff positions.

* 10 CFR 50.36(c)(1)(i)A

This section of the regulations requires safety limits to be set for important process variables which are found to be necessary to reasonably protect the integrity of the physical barriers that guard against the uncontrolled release of radioactivity.

Standard Review Plan Section 4.2

Section 4.2.II.A.2.d discusses the need for a cladding overheating criterion to prevent fuel failures during normal operation and for anticipated operational transients.

Standard Review Plan Section 4.4

Section 4.4.II reiterates the Standard Review Plan Section 4.2 requirements on overheating the cladding.

Information Notice 89-69

The Information Notice informs all BWR licensees and applicants on the loss of thermal margin caused by channel box bow.

(iii) Each proposed requirement or staff position shall contain the sponsoring office's position as to whether the proposal would increase requirements or staff positions, implement existing requirements or staff positions, or would relax or reduce existing requirements or staff positions.

* The proposed Generic Letter would implement existing requirements and staff positions in that it would correct a known deficiency in calculating the critical power ratio (CPR). This deficiency is not taking channel box bow into account in the calculations.

(iv) The proposed method of implementation with the concurrence (and any comments) of OGC on the method proposed.

* The requirements of the proposed Generic Letter would be implemented in the first reload after April 30, 1990 by assuring that the effects of channel box bow are taken into account in determining the critical power ratio. OGC has no legal objection to this proposal and their comments were incorporated.

(v) Regulatory analyses conforming to the directives and guidance of NUREG/BR-0058 and NUREG/CR-3568.

* A formal regulatory analysis is not required because the proposed Generic Letter does not impose any new positions or requirements. The actions proposed by the Generic Letter would maintain current regulatory criteria on cladding overheating. Not taking the actions proposed by the Generic Letter is likely to result in violation of CPR operating limit technical specifications and may result in violation of the CPR Safety Limit with potential fuel failures.

(vi) Identification of the category of reactor plants to which the generic requirement or staff position is to apply.

* This proposed Generic Letter is applicable to all Boiling Water Reactors.

(vii) For each such category of reactor plants, an evaluation which demonstrates how the action should be prioritized and scheduled in light of other ongoing regulatory activities. The evaluation shall document for consideration information available concerning any of the following factors as may be appropriate and any other information relevant and material to the proposed action:

(a) Statement of the specific objectives that the proposed action is designed to achieve;

* The proposed action is being taken to ensure that the SLMCPR is not violated so that fuel failures will not occur. The proposed action also ensures that the requirements of 10 CFR 50.36(c)(1)(i)A are met.

(b) General description of the activity that would be required by the licensee or applicant in order to complete the action;

* The licensees would have to modify the process computer evaluation of the critical power ratio either by incorporating changes to parameters used in the calculation of critical power ratio or by assuming a bounding estimate of the magnitude of the channel bow effect on the critical power ratio.

(c) Potential change in the risk to the public from the accidental release of radioactive material;

* The actions required by the proposed Generic Letter would reduce the risk to the public by reducing the risk of fuel failures caused by cladding overheating.

(d) Potential impact on radiological exposure of facility employees and other onsite workers.

* The actions required by the proposed Generic Letter would reduce the risk to facility employees and onsite workers by reducing the risk of fuel failures caused by cladding overheating.

(e) Installation and continuing costs associated with the action, including the cost of facility downtime or the cost of construction delay;

* Each BWR will probably require a plant-specific analysis to determine if sufficient margin is available to account for fuel channel box bow in the process computer determination of the critical power ratio. Some plants will probably require additional analyses for subsequent reloads. It is expected that corrective actions will not be needed for several plants. The most severe economic impact would be for those plants operating near the CPR operating limit which elect to impose a direct CPR penalty on the operating limit. In those cases, the estimated potential impact is up to $500,000 per 0.01 CPR penalty for fuel cycle and power reduction costs. However, it is expected that a less costly analytical resolution will be selected for those plants.

(f) The potential safety impact of changes in plant or operational complexity, including the relationship of proposed and existing regulatory requirements and staff positions;

* There will be no potential impact of changes in plant operational complexity. The actions required by the proposed Generic Letter will require that parameters used in the critical power correlation be modified so that the process computer evaluation of the critical power ratio will include the effects of channel box bow. The modified parameters would be supplied by the fuel vendors as part of the updating of process computer parameters during refueling outages.

(g) The estimated resource burden on the NRC associated with the proposed action and the availability of resources;

* The resource burden to the NRC would involve the review of two topical reports, one from General Electric and one from Advanced Nuclear Fuels Corporation, on the methodology developed to account for the effect of channel box bow on the critical power ratio determination. Any alternate proposed by individual licensee would also need review by the staff. Once the Generic Letter is issued, then each affected plant's project manager would monitor the implementation of approved methodology and procedures at his plant on the schedule required by the Generic Letter. This effort by the project managers is judged to be minimal. The resources required to review the two topical reports are available in the Reactor Systems Branch/DST/NRR (not expected to exceed two-man-months). In addition, up to six-man-months NRR resources may be needed to review all licensee actions for conformance to this generic letter.

(h) The potential impact of differences in facility type, design or age on the relevancy and practicality of the proposed action;

* The guidance in the proposed Generic Letter is applicable to all BWRs because the only important parameter is the effect of channel bow on the critical power ratio. Other factors such as facility type, design, or age have no effect on the proposed guidance of the Generic Letter.

(i) Whether the proposed action is interim or final, and if interim, the justification for imposing the proposed action on an interim basis.

* This is the final staff position in that no further changes are under consideration.

(viii) For each evaluation conducted pursuant to 10 CFR 50.109, the proposing Office Director's determination, together with the rationale for the determination based on the consideration of paragraph (i) and (vii) above, that:

(a) There is reasonable increase in the overall protection of public health and safety or the common defense and security to be derived from the proposal; and

(b) The direct and indirect costs of implementation, for the facilities affected, are justified in view of this increased protection.

* Because the guidance in the proposed Generic Letter corrects a known deficiency in the calculation of the critical power ratio, there is an increase in the overall protection of public health and safety, backfit considerations are not applicable. The direct and indirect costs of implementation are judged to be minimal.

(ix) For each evaluation conducted for proposed relaxations or decreases in current requirements or staff positions, the proposing Office Director's determination, together with the rationale for the determination based on the considerations of paragraphs (i) through (vii) above, that:

(a) The public health and safety and the common defense and security would be adequately protected if the proposed reduction in requirements or positions were implemented; and

(b) The cost savings attributed to the action would be substantial enough to justify taking the action.

* This proposed Generic Letter does not relax or decrease current requirements or staff positions.