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

APR 15 1991

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

FROM: Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

SUBJECT: CRGR PACKAGE FOR THE RESOLUTION OF GENERIC SAFETY ISSUE - 29, "BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS"

Enclosed for your information and possible review is the CRGR package for the resolution of GSI-29, "Bolting Degradation or Failure in Nuclear Power Plants," which includes: (1) draft memorandum to the EDO describing the proposed resolution of the subject generic issue, (2) proposed generic information letter, (3) regulatory analysis (NUREG-), (4) recommendations regarding a new SRP Section, and (5) NUREG-1339, "Resolution of Generic Safety Issue 29." NRR has concurred in the attached proposed generic information letter and OGC has expressed no legal objection to the generic letter.

GSI-29 was established in 1982 to address staff concerns about degradation and failure of safety-related bolting in nuclear power plants. The RES staff has concluded that sufficient basis now exists for the resolution of GSI-29.

We do not believe the proposed generic information letter for plants currently holding an OL or CP necessitates CRGR review, since it does not require licensee action or response. We do recommend that a new Standard Review Plan (SRP) Section on "Safety-Related Bolting" be developed by NRR for the review of future plants and be included in a future revision to the SRP. As part of the resolution of GSI-29, RES is transmitting to NRR specific recommendations for bolting-related topics to be addressed in the SRP (Enclosure 4).

We would be happy to provide a presentation on the resolution of GSI-29 to the CRGR if they so wish. Please advise us within two weeks as to whether or not the CRGR wishes to review the proposed resolution of GSI-29 with the staff.

Eric S. Beckjord

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

Enclosures: As stated

cc: See Next Page

ENCLOSURE 1

MEMORANDUM FOR:	James M. Taylor Executive Director for Operations
FROM:	Eric S. Beckjord, Director Office of Nuclear Regulatory Research
SUBJECT:	RESOLUTION OF GENERIC SAFETY ISSUE - 29, BOLTING DEGRADATION OR FAILURES IN NUCLEAR POWER PLANTS

The purpose of this memorandum is to formally document the resolution of the referenced generic safety issue.

GSI-29 was established in 1982 to address staff concerns about degradation and failure of safety-related bolting in nuclear power plants. The staff has performed a Regulatory Analysis (NUREG-) and concluded that sufficient basis now exists for the resolution of GSI-29. A generic information letter (Generic Letter 91-) together with NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," have been forwarded to NRR for issuance to plants currently holding an OL or CP, to inform them of the technical findings and resolution of GSI-29.

The resolution of GSI-29 is based largely on work performed by the industry through a program developed by the Joint Atomic Industrial Forum (AIF)/Materials Properties Council (MPC)/Electric Power Research Institute (EPRI) Task Group on Bolting, and resulted in two volumes of EPRI Report NP-5769, two volumes of EPRI Good Bolting Practices reference manual, and three video training tapes. As discussed in NUREG-1339, with some exceptions and qualifications, the staff endorses the industry findings and the industry recommended actions.

The resolution of GSI-29 is also based on the fact that the staff has taken actions in the past on several specific issues related to threaded fasteners in a number of bulletins, generic letters and information notices. Major areas of concern which have been addressed are: PWR Reactor Coolant Pressure Boundary (RCPB) bolting and component degradation due to boric acid corrosion (Bulletin 82-02 and Generic Letter 88-05), stress corrosion cracking (SCC) of internal bolting in certain types of check valves (Bulletin 89-02), non-conforming, misrepresented, counterfeit and fraudulent bolting (Bulletin 87-02, Information Notices 89-22, 89-56, 89-70, Generic Letter 87-02), and traceability and material control of fasteners (Information Notice 86-25). Many of the above mentioned bulletins and generic letters required the licensees not only to take short-term actions to resolve the problem but also to develop and implement continuing programs to minimize the likelihood of recurrence. Details of these can also be found in NUREG-1339.

Although value-impact studies on GSI-29 were performed by our contractors (Appendices A and B of Regulatory Analysis, NUREG-), the staff judged the studies to be inconclusive regarding a mandatory program on safety-related bolting for operating plants, and, therefore, additional requirements could not be justified in accordance with the provisions of 10 CFR 50.109 for operating plants. In addition, based on (1) bolting operating experience in both nuclear and conventional power plants, (2) the actions already taken through bulletins, generic letters, and information notices, and (3) the industry proposed actions, the Regulatory Analysis concluded that a sufficient technical basis exists for the resolution of GSI-29. The staff further concluded that leakage of bolted pressure joints is possible but catastrophic RCPB joint failure which will lead to significant accident sequences is highly unlikely.

Generic Letter 91- and the accompanying NUREG-1339 therefore suggest (but do not require) that the best way to resolve GSI-29 would be for utilities (1) to implement the industry bolting integrity program as presented in EPRI reports and video tapes and (2) to continue their actions in accordance with commitments made in response to a number of generic letters and bulletins.

RES believes that it is desirable to document guidance regarding bolting for future plants. In order to improve the review of future plants and the review of submittals from operating plants for significant plant modifications, it is recommended that a new Standard Review Plan (SRP) Section on "Safety-Related Bolting" be developed by NRR to codify existing guidance and industry-developed recommendations. This new SRP should be included in a future revision to the SRP. As part of the resolution of GSI-29, RES has transmitted to NRR specific recommendations for bolting-related topics to be addressed in the SRP.

With the issuance of the Generic Letter 91- and NUREG-1339, and the proposal to develop a new Standard Review Plan Section, Generic Safety Issue-29 is considered resolved.

Eric S. Beckjord, Director
Office of Nuclear Regulatory Research

cc: T. Murley, NRR
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ENCLOSURE 2

GENERIC LETTER

TO: All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants

SUBJECT: GSI-29, Bolting Degradation or Failure in Nuclear Power Plants (Generic Letter 91-)

This letter informs licensees of the technical findings resulting from the NRC resolution of GSI-29, including those resulting from an industry-sponsored program on bolting degradation and failure in nuclear power plants. Bolting in this context includes all safety-related bolts, studs, embedments, machine/cap screws, other special threaded fasteners, and all their associated nuts, and washers.[1] Both the industry findings and the NRC staff resolution of this issue are documented in NUREG-1339.[2] It is expected that recipients will review the information for applicability to their facilities and consider appropriate actions, if necessary. However, the suggestions contained in this letter do not constitute NRC requirements; therefore no specific action or written response is required.

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(1) It is to be noted that concerns regarding reactor vessel closure studs are being addressed under Generic Safety Issue 109, "Reactor Vessel Closure Failure," and therefore are not considered under GSI-29.

(2) NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants" was published by U.S. NRC in June 1990, and is enclosed with this generic letter.

BACKGROUND

Both the NRC and industry noted that from 1964 to the early 1980s the incidence of reported failures of high-strength bolting in Class I components, component supports, and other safety-related equipment had increased. Critical bolting applications in nuclear power plants constitute an integral part of the reactor coolant pressure boundary (RCPB) and include closure studs or bolts on reactor vessels, pressurizers, reactor coolant pumps, and steam generators. Failure of these bolts or studs could result in the loss of reactor coolant and jeopardize safe operation of the plant. Bolting applications are also an integral part of the pressure boundary of other safety-related systems. These and other bolting applications, such as component support and embedded anchor bolts or studs, are essential for withstanding transient loads created during abnormal or accidental conditions. Generic Safety Issue (GSI)-29, "Bolting Degradation or Failure in Nuclear Power Plants," was therefore established in 1982 to address these staff concerns.

In June 1982, NRC issued IE Bulletin No. 82-02 which addressed the staff concern about degradation of RCPB bolting from borated water. The bulletin required responsive actions by all PWR licensees because, as more and more plants became operational, threaded fasteners showed an increased frequency of degradation due to a variety of underlying mechanisms. In response to NRC actions, the Atomic Industrial Forum (AIF) joined with the

Materials Properties Council (MPC) and Electric Power Research Institute (EPRI) to form the Joint AIF/MPC/EPRI Task Group on Bolting. The Task Group was composed of representatives from AIF member organizations--utilities, vendors, architect-engineers--plus representatives from EPRI and MPC.

There is some evidence from the responses to IE Bulletin No. 82-02, as reported in NUREG-1095[3], that the increase in bolting degradation and failure observed from 1964 to the early 1980s was a function of the increased number of installed bolts. However, there is also evidence that as plant maintenance personnel accumulated experience from plant operation, the incidence of leaking joints and reported failures decreased.

Common characteristics among the reported incidents included fasteners that had high, sustained tensile stresses; out-of-specification torquing; an aqueous environment caused by high humidity; primary water leakage; borated water leakage; or materials that were overly hard. The most frequently observed failure mode for the structural bolting was stress corrosion cracking. Low-alloy steels, quenched and tempered steels, and maraging steels all were degraded by stress corrosion cracking. A small number of overstress failures were traced to improper

[3] NUREG-1095, "Evaluation of Responses to IE Bulletin 82-02," May 1985, U.S. NRC.

heat treatment or low-strength material. Several pressure-retaining bolts failed because of corrosion wastage. The RCPB components that were involved in these failures included steam generator manway closures, reactor coolant pumps, pressurizer manway closures, reactor vessel closures, chemical and volume control system isolation valves, check valves in the ECCS that form part of the RCPB, and other check valves. Some reactor vessel internals, mainly the lower thermal shield bolts and upper core barrel bolts, had been degraded due to fatigue and stress corrosion cracking. In some plants, the degraded bolting required extensive and expensive replacement. Evaluation of reported events led the NRC and industry to conclude from the nature and frequency of the evaluated failures that significant levels of degradation can occur among safety-related fasteners.

The Joint AIF/MPC/EPRI Task Group on Bolting developed the technical bases for resolving GSI-29. In working toward resolution of GSI-29, EPRI assumed the lead for completing 19 general bolting tasks. Results of the work of the Joint AIF/MPC/EPRI Task Group were presented in detail in a two-volume report, "Degradation and Failure of Bolting in Nuclear Power Plants," EPRI NP-5769.[4] Since the early 1980s the Institute of Nuclear Power Operation (INPO) has issued a number of documents

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[4] EPRI NP-5769 was published in April 1988, and is available from the Electric Power Research Institute, Palo Alto, California 94303.

(notably SOER No. 84-5) and recommended certain actions in response to potentially unsafe conditions involving degraded bolting.

Further refinements in codes and standards are underway by the responsible committees in the ASME Boiler and Pressure Vessel Code and ASTM (e.g., Committee F16 on Fasteners). All of these industry actions and their contributions to the resolution of GSI-29 are discussed in NUREG-1339.

The Nuclear Management and Resources Council (NUMARC) issued a letter to its members on July 6, 1989, notifying them of the publication of EPRI Reports NP-5769 and NP-5067[5], and stating that they provide the industry's technical basis for resolution of GSI-29. This letter encouraged them to refer to these reports to perform appropriate root cause analyses and implement proper corrective actions in response to NRC Bulletin 87-02 ("Fastener Testing to Determine Conformance with Applicable Material Specifications").

The NRC has taken several steps that were factored into the resolution of the issue. The NRC staff and its contractors

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[5] EPRI NP-5067, "Good Bolting Practices Manuals· Vol. 1: Large Bolt Manual," was published in 1987. Vol. 2 of Good Bolting Practices Manuals ("Small Bolts and Threaded Fasteners") was published in 1990. Both are available from EPRI.

produced several documents (NUREG and NUREG/CR reports) dealing with bolting issues. The staff also addressed several specific bolting-related issues in bulletins, generic letters and information notices. The bulletins and some of the generic letters required both one-time actions and continuing programs. The requirements and recommendations of these generic communications are discussed in NUREG-1339.

CONCLUSIONS/SUMMARY

Based on the above, the NRC staff has concluded that by considering all of the available information from industry and regulatory sources, and previous and ongoing licensee actions, a sufficient basis exists for the resolution of GSI-29.

The NRC staff has reviewed the technical findings developed by the industry and presented in EPRI NP-5769, and with some exceptions and qualifications as discussed in Section 3, "Conclusions," of NUREG-1339, endorses the findings in the two-volume EPRI report.[6]

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[6] EPRI NP-5769 proposes that bolted connections that satisfy certain criteria would exhibit "leak-before-break" characteristics and be subject to less stringent inservice inspection criteria. A related proposal for an ASME code case has been submitted and is under review by the ASME Section XI Subcommittee. If the code case is approved by the ASME, NRC will then consider endorsement. General endorsement of EPRI NP-5769 does not imply NRC endorsement of the proposed code case.

The NRC staff believes that there are potential benefits from implementing the industry-developed recommendations delineated in the EPRI reports and supports appropriate implementation by all licensees. In order to efficiently implement these industry-developed recommendations, the staff believes the following steps may be helpful to licensees:

First, review of the following industry-developed information:

1. EPRI NP-5769, Vols. 1 and 2.
2. EPRI Good Bolting Practices manuals; Vol. 1: "Large Bolt Manual," and Vol. 2: "Small Bolts and Threaded Fasteners," NP-5067.
3. Videotapes: "Pressure Boundary Bolting Problems," Parts I, II and III.[7]

Second, review of the NRC staff report, NUREG-1339, which discussed the NRC's evaluation of, and exceptions to, EPRI NP-5769.

The staff agrees that an effective means of ensuring bolting reliability, as recommended in EPRI NP-5769, would be through the development and implementation of plant-specific bolting integrity programs that encompass all safety-related

[7] These videotapes are available from EPRI.

bolting. NUREG-1339 includes recommendations and guidelines for the content of a comprehensive bolting integrity program. Additional details on bolting integrity can be found in EPRI NP-5769. The plant-specific bolting integrity program may incorporate licensee commitments for continuing actions made in response to the previously issued NRC bulletins and generic letters listed in NUREG-1339.

Finally, bolting may be one of the components for which age related degradation may be significant and, therefore, should be considered in identifying which systems, structures, and components are important as a plant ages. This could possibly be an issue for license renewal.

The information in this letter does not constitute NRC requirements; therefore, no specific action or written response is required. With this generic letter, the staff considers the broad safety issue involving bolting degradation or failure resolved; however, additional regulatory actions may be warranted if specific problems concerning safety-related bolting should occur in the future. If you have any questions about the information in this letter, please contact one of the technical contacts listed below.

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Enclosure: NUREG-1339

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ENCLOSURE 3

REGULATORY ANALYSIS

Resolution of Generic Safety Issue No. 29, "Bolting Degradation or Failure in Nuclear Power Plants."

I. Statement of the Problem

From 1964 to the early 1980s, the NRC observed that the number of degradation events (bolt cracking, corrosion, failure, etc.) of threaded fasteners reported by licensees of operating reactors had increased. Many of the events were related to the reactor coolant pressure boundary (RCPB) components and major component support structures. This caused an increasing concern regarding the integrity of the RCPB and the reliability of the component support structures following a loss-of-coolant accident (LOCA) or seismic event.

Originally an integral part of USI A-12, "Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports," the bolting safety issue was separately identified as Generic Safety Issue (GSI)-29, "Bolting Degradation or Failure in Nuclear Power Plants." The technical reason for that action was that the types and variety of failure mechanisms active in bolting safety problems were distinctly different from those to be addressed in structural steel supports. Bolting in the context of GSI-29 includes bolts, studs, embedments, cap/machine screws, other special threaded fasteners, and all their associated washers and nuts.*

When the NRC prioritized generic issues in November 1982, GSI-29 received a HIGH rating. The safety aspects of GSI-29 can be summarized as follows. Critical bolting applications in nuclear power plants constitute an integral part of the RCPB and include closure studs or bolts on reactor vessels, pressurizers, reactor coolant pumps, and steam generators. Failure of these bolts or studs could result in the loss of reactor coolant and jeopardize safe operation of the plant. Bolting also is an integral part of the pressure boundary and component supports of other systems, both safety-related and not. These and other bolting applications are essential for withstanding transient loads created during abnormal or accident conditions.

Failures reported by licensees involved a variety of threaded fasteners and several causes. As a result, several different failure mechanisms had to be considered. Most frequent were wastage (corrosion/erosion) from boric acid attack and stress corrosion cracking (SCC). The former

* It is to be noted that concerns regarding reactor vessel closure studs are being addressed under Generic Safety Issue 109, "Reactor Vessel Closure Failure," and therefore are not considered under GSI-29.

occurred more often at RCPB joints; the latter occurred mostly in structural bolting. Details regarding the nature and extent of bolting degradation and failure, including a review of the relevant available literature, can be found in NUREG-1339, "Resolution of Generic Safety Issue-29: Bolting Degradation or Failure in Nuclear Power Plants" (Ref. 1).

2. Summary and Conclusions

The conclusion that GSI-29 can be resolved is based on the content and availability of material developed by industry and the NRC, and the actions taken by both to address specific bolting problems observed in nuclear power plants.

Industry actions include the program developed by the Joint Atomic Industrial Forum/Materials Properties Council/Electric Power Research Institute (AIF/MPC/EPRI) Task Group on Bolting. This effort resulted in two volumes of EPRI report EPRI NP-5769 (Ref. 2), three video training tapes, and the two volumes of EPRI Good Bolting Practices reference manual (Ref. 3). Industry representatives established the Bolting Technology Council (an MPC affiliate) to take the lead in sponsoring bolting research, recommending practices, gathering and providing information, and promoting education on installation, application, behavior, and interactions of fasteners. The Institute of Nuclear Power Operation (INPO) has also taken a number of actions over the years in response to potentially unsafe conditions of degraded bolting and bolting related issues. Further refinements in codes and standards are underway by the responsible committees in the ASME Boiler and pressure Vessel Code and ASTM (e.g. Committee F16 on Fasteners). All of these industry actions and their impacts on GSI-29 are discussed in NUREG-1339.

Since 1982, the NRC addressed a number of specific bolting issues and took several additional steps that were considered in the resolution of the issue. The NRC has issued 3 bulletins, 2 generic letters, and 8 information notices dealing with specific bolting problems, as listed in NUREG-1339. Many of the generic letters and bulletins required licensees not only to take short-term actions to resolve the problem but also to develop and implement continuing programs to minimize the likelihood of recurrence. Details can be found in NUREG-1339.

Although value-impact studies on GSI-29 were performed by our contractors (Appendices A and B of this Regulatory Analysis), the staff judged the studies to be inconclusive regarding a mandatory program on safety-related bolting for operating plants, and, therefore, additional requirements could not be justified in accordance with the provisions of 10 CFR 50.109 for operating plants. In addition, based on (1) bolting operating experience in both nuclear and conventional power plants, (2) the actions already taken through bulletins, generic letters, and information notices, and (3) the industry proposed actions, the staff concluded that a sufficient technical basis exists for the resolution of GSI-29. The staff further concluded that leakage of bolted pressure joints is possible but catastrophic RCPB joint failure which will lead to significant accident sequences is highly unlikely. More detailed discussion on these considerations are provided in Sections 5.a and 5.b below. For future plants, however, it was concluded that a new Standard Review Plan section should be developed to codify existing bolting requirements and industry-developed initiatives, including the development and implementation of a plant-specific bolting integrity program.

3. Objective

The objective of the proposed resolution of GSI-29 is to provide assurance that integrity of safety-related threaded fasteners is maintained.

4. Alternatives

Several possible alternatives for the resolution of GSI-29 have been considered by the staff. They are listed as follows:

a. Take no further action beyond those already covered by existing NRC regulations, NRC bulletin/information notices and generic letters, and the ASME Code.

b. Issue a generic letter (for information only) to owners of plants that currently have an OL or CP that suggests, but does not require, certain actions. The suggested actions include: (1) review relevant industry-developed information and NRC documents, and (2) develop and implement a plant-specific bolting integrity program applicable to all safety-related joints.

c. Develop a new SRP section to provide guidance to the staff for the review of future plants. The elements of the review would include all safety-related joint design, threaded fastener material selection, and programmatic aspects dealing with bolting integrity during construction and operation/maintenance.

d. Require that plant owners currently holding an OL or CP to: (1) perform an engineering evaluation (or reevaluation) of all existing safety-related joint designs, fastener materials and field practices (construction and maintenance), and (2) replace bolting (with redesigned joints, if necessary) that do not meet the industry-developed criteria or NRC requirements.

5. Evaluation of Alternatives and Decision Rationale

a. Costs and Benefits

Alternative 4.a (taking no further action) was rejected because, although many requirements and much guidance have been published for bolting, no overall staff conclusions on the adequacy or effectiveness of these actions have been issued. Therefore, rather than taking no action (Alternative 4.a), Alternatives 4.b and 4.c were selected. Alternative 4.b applies to plants that currently have a CP or OL, and Alternative 4.c applies to future plants. The basis for selecting these alternatives is described in more detail in the following paragraphs. In summary, it was the judgement of the staff that existing ASME and ASTM Codes and Standards, NRC requirements and licensee actions (including bulletins and generic letters), and information available from the industry (e.g., EPRI NP-5769 report, etc.) and NRC (NUREGs and NUREG/CRs), would adequately limit the risk resulting from safety-related bolting failure in current plants. However, it was decided to inform licensees of the staff conclusions regarding the adequacy or effectiveness of the above mentioned bolting-related requirements, guidance, information and activities, and to suggest, but not require, that current licensees develop and implement a plant-specific overall bolting integrity program that includes current NRC requirements and reflects the information and recommendations made by the industry-sponsored program. Such a plant-specific bolting integrity program should incorporate licensee commitments for continuing actions made in response to the previously issued NRC bulletins and generic letters listed in NUREG-1339.

New plants would be reviewed in accordance with a new Standard Review Plan section that would codify existing guidance and industry-developed recommendations. This guidance is justified for future plants because this represents only a codifying of existing guidance and practices.

The alternatives selected were partially based on the two value-impact analyses (Appendices A and B), one on the RCPB bolting and the other on safety-related bolting in systems other than the RCPB. The staff's judgement of uncertainties and the impact of on-going activities which are not reflected in the value-impact analyses were also major factors in the decision-making process. Regarding the PNL value-impact analysis of the RCPB bolting (Appendix A), the best estimate indicated that the proposed action had the potential to reduce risk by 9,819 person-rem for the whole industry. This was based on a best estimate of a reduction in core melt frequency of 2.73E-6/reactor-year for PWRs and 2.9E-7/reactor-year for BWRs. This magnitude of risk reduction is not considered by the staff to satisfy the 10 CFR 50.109 criteria that a required action results in a substantial increase in the overall protection of the public health. Further, in the staff's opinion, these estimated values of risk reduction probably erred on the high side. Considering the bolting operating experience in both nuclear and conventional power plants (see 5.b. below), the actions (through bulletins, generic letters, and information notices) already taken since reference 1 was prepared, and the industry proposed actions, the staff concluded that leakage of bolted pressure joints is possible but catastrophic RCPB joint failure which will lead to significant accident sequences is highly unlikely.

The PNL value-impact analysis, however, did result in a best estimate cost-benefit ratio of $239 per person-rem, which is very favorable. When cost savings in public and onsite property damage are considered (these values, in the staff's opinion, have very high uncertainties), the cost-benefit ratio is even more favorable. As a matter of fact, because of the potential cost savings, the cost-benefit ratio turned out to be negative.

The best estimate analysis was based on the assumption that all carbon or low-alloy steel bolts would be susceptible to boric acid wastage and would be replaced by stainless steel bolts. Such a program would be quite expensive for plants already constructed and the staff feels that the PNL study underestimated the cost. Furthermore, an extensive RCPB bolting inspection and replacement program (beyond that required by the Section XI of the ASME code and the requirements of IE Bulletin 82-02) might require increased duration of refueling outages. Those costs were not included in the PNL study.

If more realistic cost estimates are employed, an increased cost-benefit ratio would result. In the staff's judgement, a more realistic estimate of the cost benefit ratio would exceed \$1000/person-rem for plants currently holding an OL or CP.

The staff, therefore, concluded that a mandatory replacement program for RCPB bolting could not be justified for plants that currently have an OL or CP, and instead Alternative 4.b was selected. However, the staff further concluded that for future plants, the PNL value-impact analysis was more valid. It should be noted however, that in the proposed SRP, the staff does not specifically recommend that only stainless steel should be used for RCPB bolting. Bolting material selection should be made after careful consideration of all of the concerns addressed in the NRC and EPRI publications discussed in the Generic Letter. The SRP proposed by the staff for future plants includes provisions for a comprehensive bolting integrity program that deals with initial design, material selection, and construction and maintenance practices. Therefore, Alternative 4.c was selected for future plants.

The INEL study (Appendix B) examined the risks related to failure of safety-related bolting in systems other than the RCPB. Approximately ten safety systems were examined for risk sensitivity. In addition, the primary coolant system component supports also were examined in the risk analysis. The risk analysis was based on a hybrid probabilistic risk analysis (PRA) model developed after review of six plant-specific PWR PRA models. Based on this hybrid PRA model, it was concluded that the most significant risk associated with degraded bolting was the failure of either the on-site emergency power system (including associated support systems) or the reactor coolant system (RCS) component supports during a severe seismic event. Although the analysis was based on PWR plants, it is the staff's judgement that the results are also generally applicable to BWRs since the risk is dominated by seismic consideration.

The INEL best estimate of core melt frequency was 3.5E-5/reactor year and the corresponding public risk reduction was 7300 person-rem based on 67 operating PWRs. The corresponding cost-benefit ratio reported by INEL was $3700/person-rem. This cost-benefit ratio excluded consideration of on-site property damage and averted occupational radiation exposure. When INEL included this consideration, the result was a net cost saving.

The INEL study has considerable uncertainties that are discussed in the following paragraphs.

(1) Calculation of Reduction In Core Melt Frequency

As stated above, the INEL risk study was based largely on seismic risk. INEL calculated that there was a significant increase in the risk frequency of core melt that would result from severe seismic events if RCS component supports or the emergency power system anchorage was degraded. The INEL study assumed that all of this risk was associated with degraded threaded fasteners.

Since RCS component supports and equipment anchorages consist of more than threaded fasteners (e.g., welded anchorages for electrical cabinets), this is clearly an over-estimate of the benefit that could be achieved by surveying and testing of threaded fasteners, and replacement of those found to be degraded.

Moreover, the risk contribution from vibratory equipment such as pumps and air compressors probably was overestimated. Since they are subjected to vibratory loading and stress under normal operating conditions, degraded bolts will be uncovered during normal operation or during normal maintenance and inspection.

(2) Calculation of Person-Rem

The INEL calculation of the public exposure resulting from a core melt was based on the Seabrook plant, which was one of the six plants used for constructing the hybrid PRA model. The Seabrook plant has a more robust containment than a typical PWR. As a result, the person-rem calculated was relatively low considering the high estimate of core melt frequency. In this regard, the staff's judgement is that the public exposure would be higher than resulted from the INEL analysis for a typical operating plant, if the core melt frequency was as high as that calculated by INEL.

(3) Cost Estimates

The cost estimates in the INEL study probably were too low given that the proposed sequential steps of surveying, testing and replacement be carried out. These steps likely would require an extensive bolting inspection and replacement program, and might require increased duration of refueling outages which was not included in the cost analysis.

Considering the uncertainties cited above, the staff judged the INEL proposed program to be marginal for plants currently holding an OL or CP from the viewpoints of reduction in risk to the public and cost-benefit ratio. The staff, therefore, concluded that requiring a program such as the one proposed by INEL for safety-related bolting other than RCPB applications could not be justified for plants that currently hold an OL or CP. Instead Alternative 4.b was selected. However, the staff further concluded that for future plants, a more effective review as delineated in a proposed new SRP section should be followed.

The elements of the proposed SRP review would include all safety-related joint design, threaded fastener material selection, and programmatic aspects dealing with bolting integrity during construction and operation/maintenance. When the proposed SRP is implemented at new construction, essentially all the potential risk reduction associated with safety related bolting can be readily achieved. The staff believes the risk reduction and costs presented in Appendices A and B should be more reasonably applicable for a new plant. Also, the staff concern that the cost estimates did not consider the impact on outage duration would not apply to new plants.

The fourth alternative (4.d), although clearly sufficient to ensure the integrity of bolted connections, is not warranted on the basis of the observed failures to date and because of the high cost of implementation. It is judged that only a small additional reduction in risk would be achieved by Alternative 4.d relative to Alternative 4.b for plants that currently have an OL or CP. However, the engineering, labor, and material costs for alternative 4.d would be considerably higher than those for Alternative 4.b. More important from a cost viewpoint, some additional outage time would be needed to replace suspect bolting, even if that work was performed during planned refueling outages. The combination of replacement power costs and higher labor and material costs would result in a much less favorable cost-benefit ratio than Alternative 4.b. Therefore, Alternative 4.d was not selected.

b. Operating Experience

The inconclusive nature of the contractors' value-impact analyses on GSI-29 regarding a mandatory program on safety-related bolting for operating plants, as mentioned in 5.a above, prompted the staff to look into the operating experiences on bolting in nuclear and conventional power plants, especially those in the pressure boundary applications. A summary of the findings is given below.

(I) Experience on pressure boundary applications: During a meeting of the ACRS Subcommittee on Materials and Metallurgy on January 9, 1991, the staff's proposed resolution of GSI-29 was reviewed. The following is a direct quote from the transcript of the meeting (Ref. 4), as stated by John Bickford, past chairman of the Bolting Research Council, on his twenty years of operating experience regarding pressure boundary bolting.

> "The thing that we were concerned about [in a nuclear power plant].... was radiation released which might be caused by a large or small LOCA or damage to components which would prevent a smooth shutdown in case of an emergency or just in general.
>
> ".... None of those things had been actually reported. .... we were generating this information from safety-related reports from the operating plants.
>
> "These things as far as our committee work was concerned, LOCAs and so forth, might have been caused by either simultaneous failure of several bolts -- in other words a joint

failure, unzipping as has been talked about, or loose parts in the system and those things might be preceded by the rupture of individual bolts or the loss of individual bolts.

"Now loose parts in the system had been observed and were reported. Rupture of individual bolts had been observed and reported. Loss of individual bolts had been reported. Simultaneous joint failure had not been reported. (Emphasis added).

"I think it might be pertinent to say that .... [I am aware of a] Tampa Electric Company [incidence which involved] the total failure of a [pressure] joint. I believe it was in a heat exchanger in a conventional power plant.

"The problem was that the joint had been sealed with Furmanite [sic], which had trapped corrosive materials and so forth inside this thing and the joint just suddenly exploded and one person I believe was killed. That is the only incident that I am aware of in 20 years of bolting where a pressure vessel joint has failed catastrophically like that. (Emphasis added).

"Many times leaks, many times partial failures but never -- that's the only incident I know of, of that kind."

It is worthwhile to note from the above presentation that to date there was no LOCA, large or small, reported in any of the operating nuclear plants due to failure of bolted RCPB joints. The only "unzipping" type of failure of pressure boundary of any kind in any industry, based on J. Bickfords' extensive experience on bolting, was at a conventional power plant heat exchanger, and that was due to the use of Furmanite sealant. In NRC IE Bulletin 82-02, the licensees were cautioned on the prolonged used of improper sealant, and advised on the proper selection, procurement and application of fastener sealant compounds to minimize fastener susceptibility to SCC environments.

(2) Estimate from contractors' value-impact report versus operating experience on nuclear power plants:

The PNL value-impact analysis (Appendix A) estimated that, for a PWR plant, the probability of a small-small LOCA (0.38" < D ≤ 1.2") due to bolting failure would be 4.39 E-3/reactor year. If the PNL estimate was correct, during almost 6 years of operation of 80 PWR plants since the issuance of the PNL value-impact analysis, one would expect the occurence of 6 x 80 x 4.39 E-3, or about 2 small-small LOCAs on the RCPB due to failure of bolting. Actually no LOCAs of any size have been experienced during this period. This tends to confirm the conclusion discussed earlier in 5.a that the PNL risk estimation was conservatively high.

(3) Survey of Reports on Precursors to Potential Severe Core Damage Accidents:

NUREG/CR-4674, "Precursors to Potential Severe Core Damage Accidents: A Status Report" is a report published annually which documents the findings of the Accident Sequence Precursor (ASP) Program conducted by the Nuclear Operations Analysis Center at Oak Ridge National Laboratory. The ASP Program reviews the licensee event reports

(LERs) of operational events that have occurred at nuclear power plants and identify and categorize precursors to potential severe core damage accident sequences. Accident sequences of interest are those that, if additional failures were to have occurred, would have resulted in inadequate core cooling and that would have potentially resulted in severe core damage. Accident sequence precursors are events that are important elements in such accident sequences. Such precursors could be infrequent initiating events or equipment failures that, when coupled with one or more postulated events, could result in a plant condition leading to severe core damage.

A LER search was performed by ORNL to list all LERs during the period from 1985 to 1989 that mentioned bolting or threaded fasteners. These LERs were compared to the list of precursors identified in NUREG/CR-4674 as having a conditional core damage probability greater than 1E-6 during the same time period. Only 14 LERs were identified and these are summarized in Table 1. Among the 14 cases listed, bolting problems are generally only partial contributors to the precursors. A large number of these bolting problems were related to internals bolting, others are related to improper torquing, loose bolts and nuts, missing bolts, etc. The recommended bolting integrity program delineated in the resolution of GSI-29 is not designed to address all of these types of bolting problems. Problems related to loose or improperly aligned internal set screws, missing bolts, loose bolts or nuts, etc., are better handled by programs such as in-service inspection, regular maintenance, or the implementation of other on-going NRC programs, such as USI A-46 and Individual Plant Examinations for External Events (IPEEE), which address the inadequacies of supports and their bolting due to design and installation.

c. Other Considerations

When the scope of GSI-29 was limited to the RCPB, another GSI was established to cover bolting associated with other components, particularly structural supports where SCC had led to failures. This issue, GSI-62, "Reactor Systems Bolting Application," was re-evaluated by the NRC staff in August, 1988. It was concluded that the safety concerns of GSI-62 would be addressed under the broadened scope of GSI-29. Therefore, GSI-

62 is considered subsumed by the resolution of GSI-29.

6. Implementation

This regulatory analysis provides bases for resolution of GSI-29 and recommends: (1) issuing a generic letter for information to plants having an OL or CP, and (2) developing a new SRP section dealing with safety-related threaded fasteners for future plants.

It is expected that recipients of the generic letter will review the information cited in the generic letter for applicability to their facilities and consider appropriate actions, if necessary, to avoid future problems. However, the suggestions transmitted by the generic letter do not constitute NRC requirements; therefore, no specific action or written response is required by the generic letter.

For future plants, a more effective review in a future SRP section should be used. The elements of the review would include all safety-related joint design, threaded fastener material selection, and programmatic aspects dealing with bolting integrity during construction and operation/maintenance.

References

1. NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," U.S. NRC, June 1990.

2. EPRI NP-5769, "Degradation and Failure f Bolting in Nuclear Power Plants," Volumes 1 and 2, Prepareu by Applied Science & Technology for EPRI, April 1988.

3. EPRI NP-5067, "Good Bolting Practices--A Reference Manual for Nuclear Power Plant Maintenance Personnel, Volume 1: Large Bolt Manual," and Volume 2: "Small Bolts and Threaded Fasteners," EPRI, 1987 and 1990, respectively.

4. Official Transcript of Proceedings of NRC ACRS Subcommittee on Materials and Metallurgy, January 9, 1991, Bethesda, Maryland, pages 99 to 101.

TABLE 1

LERs ('85 - '89) RELATED TO THREADED FASTENERS THAT RESULTED IN CONDITIONAL CD PROBABILITY ≥ 10^{-6}

	LER No.	Conditional CD probability	Description
1.	333/85 - 025	1.1×10^{-6}	A loosened locking screw in a motor operator on the HPCI steam supply isolation valve caused the torque switch to be set improperly, resulting in failure of the valve and unavailability of HPCI. Since RCIC was also unavailable due to maintenance, plant was placed in 24 hour LCO.
2.	373/85 - 045	7.2×10^{-5}	Loss of Circulating and Non-Safety Service Water, due to fatique failure of circulating pump discharge valve gear operator mounting bolts (Installation error and wrong torque value).
3.	249/86 - 013	2.7×10^{-6}	2/3 EDG failed to close manually onto bus 33-1 due to loose terminal block screw in a junction box, but was able to be synchronized manually to bus 23-1.
4.	389/86 - 011	2.6×10^{-6}	Train "2A" EDG failed to start because of mechanical problem with the governer. Train "2B" EDG was also shutdown due to rubbing between its cooling fan and the shroud. This was caused by loosening of fan hub set screw resulted from vibration of cooling unit. Therefore EDG system was not available.
5.	293/87 - 014	3.9×10^{-4}	LOOP and 1 EDG out of service (inspection). The pre-lube pump of 2nd EDG failed due to loose mounting bolts.

LERs ('85 - '89) RELATED TO THREADED FASTENERS
THAT RESULTED IN CONDITIONAL CD PROBABILITY ≥ 10^{-6} (Cont'd)

LER No.	Conditional CD probability	Description
6. 324/87 - 001	2.4×10^{-4}	Reactor trip with HPCI unavailable, RCIC full flow test isolation valve failed to close (50% open) due to out of adjustment of limit switch, caused by improperly aligned set screw.
7. 280/88 - 011	1.5×10^{-5}	PORV failure due to incorrectly torqued hold-down screws and bolts which allowed actuator diaphragm to shift.
8. 321/88 - 018	1.5×10^{-5}	Reactor scram with loss of nonessential loads and RCIC degraded operation (RCIC turbine steam supply valve failed to close fully caused by unsecured set screw for yoke stem bushing).
9. 323/88 - 008	4.1×10^{-6}	LOOP with safety injection. Galled thread resulted in RC pump feeder line electrical ground fault.
10. 328/88 - 005	3.8×10^{-4}	Both train A and B centrifugal charging pump speed increasers failed due to back out of gland seal retaining bolts inside the lube oil pump, resulting in inavailability of high head injection system.
11. 339/88 - 004	2.5×10^{-6}	Both EDGs unavailable, one out for maintenance, other the output breakder failed to close (because closing springs were not charged, due to mounting bolts on the charging motor backing out of breaker housing, allowing the charging motor to disengage).

LERs ('85 - '89) RELATED TO THREADED FASTENERS THAT RESULTED IN CONDITIONAL CD PROBABILITY $\geq 10^{-6}$ (Cont'd)

LER No.	Conditional CD probability	Description
12. 324/89 - 009	3.6×10^{-5}	Reactor scram caused a LOOP. One of two trains of LPCI/RHR was inoperable due to stucked-closed injection valve, caused by unscrewing of valve disc nut (inadequate insertion of locking pin).
13. 400/89 - 006	4.4×10^{-6}	Several mounting bolts connecting the flanged junction box to the "B" main feedwater pump (MFP) were missing. Incorrect action by Fire Protection Technicians subsequent to difficulty in resetting fire protection deluge valves over the MFPs resulted in water sprayed on the "B" MFP, causing internal short in the junction box which led to reactor trip/turbine trip.
14. 483/89 - 008	1.2×10^{-6}	During a turbine trip, the failure to reset a protective relay (due to a loose calibration set screw) for the main generator output breaker led to complete loss of power to the 4KV Safeguards Bus. This resulted in inoperable radiation monitors and also ESF actuation of turbine driven aux feed pump.

Total
(5 yrs. 100 reactors) 1.23×10^{-3}

Average Conditional CD probability = $1.23 \times 10^{-3}/500$ = $\underline{2.46 \times 10^{-6}}$/RY

ENCLOSURE 4

Recommendations Regarding a New SRP Section on "Safety-Related Bolting"

In the course of resolving GSI-29, "Bolting Degradation or Failure in Nuclear Power Plants," the RES staff concluded that a mandatory program on safety-related bolting cannot be justified for plants currently holding an OL or CP. Instead a generic information letter will be issued to owners of these plants which recommends, but does not require, certain actions. RES concluded that it is justifiable and desirable to provide additional guidance for future plants. To facilitate staff review of these future plants and review of submittals from operating plants for significant plant modifications, it is recommended that a new Standard Review Plan (SRP) Section on "Safety-Related Bolting" be included in a future revision to the SRP.

Provided below is a summary of what RES believes should be the major elements of this new SRP Section.

I. This new SRP Section should be developed to guide the NRC staff to review the programmatic aspects related to the integrity of safety-related bolting during construction, operation, testing and maintenance. Bolting in this context includes all safety-related bolts, studs, embedments, cap/machine screws, other special threaded fasteners, and all their associated washers and nuts, but with the exception of the reactor pressure vessel RCPB (reactor coolant pressure boundary) joint bolting and internal bolting. Reactor pressure vessel RCPB joint bolting (with the exception of reactor vessel closure studs) and internal bolting are covered elsewhere in SRP Sections 5.3.1, "Reactor Vessel Materials," and 5.2.3, "Reactor Coolant Pressure Boundary Materials." However, possible bolting degradation caused by borated water corrosion (wastage) is not discussed in these documents and probably should be addressed in this new SRP Section. Concerns regarding reactor vessel closure studs are being addressed under Generic Safety Issue 109, "Reactor Vessel Closure Failure."

2. Subsections of this new SRP Section should include: areas of review, acceptance criteria, review procedures, evaluation findings, and implementation. In each subsection, the following programatic aspects of bolting integrity should be addressed: bolting material specifications (including guidelines for certification), bolting material selection, traceability/control and design, bolting mechanical design, compatibility of bolt materials with the environment (including lubricants and sealants) and the thermal insulation, fabrication and processing

practices of bolting materials, safety-related joint design, testing (destructive fracture toughness tests and nondestructive examination) and inspection (including receipt/preinstallation and inservice inspection) procedures of bolting, installation of bolting (including guidelines for tightening such as preloading and/or torquing of bolts), and bolting storage requirements.

3. The SRP Section should include a review of (1) the applicant's proposed implementation of industry-developed recommendations on safety-related bolting, and (2) the applicant's plans to address NRC staff bulletins, generic letters and information notices dealing with threaded fasteners.

The resolution of GSI-29 is based largely on work performed by the industry through a program developed by the joint Atomic Industrial Forum (AIF)/Materials Properties Council (MPC)/Electric Power Research (EPRI) Task Group on Bolting. This resulted in two volumes of EPRI report EPRI NP-5769 (Ref. 1), two volumes of the Good Bolting Practices reference manual (Ref. 2) and three video training tapes (Ref. 3). In addition, industry representatives established the Bolting Technology Council (an MPC affiliate) to take the lead in sponsoring continued bolting research, recommending practices, gathering and providing information, and promoting education on installation, application, behavior, and interactions of fasteners. The Institute of Nuclear Power Operation (INPO) has taken action in response to potentially unsafe conditions of degraded bolting. Further refinements in codes and standards are underway by the responsible committees in the ASME Boiler and pressure Vessel Code and ASTM (e.g., Committee F16 on Fasteners). All of these industry actions, the technical findings of EPRI NP-5769 and the staff positions on them are discussed in a staff NUREG-1339 (Ref. 4).

During the period in which GSI-29 was being resolved, the staff addressed several specific issues on threaded fasteners in bulletins, generic letters and information notices (e.g., PWR reactor coolant pressure boundary bolting and component degradation due to boric acid corrosion, stress corrosion cracking of internal bolting in certain types of check valves, traceability and material control of fasteners, and non-conforming, misrepresented, counterfeit and/or fraudulent bolting). These details also can be found in NUREG-1339. The staff concluded that by considering all of the available information from industry and regulatory sources, previous and ongoing licensee actions, and bolting operating experience in both nuclear and conventional power plants (Section 5.b of GSI-29 Regulatory Analysis, NUREG-   , Ref. 5), a sufficient basis existed for resolution of GSI 29 for operating plants.

The RES staff concluded that lessons learned from the technical findings for operating plants set forth in EPRI NP-5769 and NUREG-1339 should also be applicable as the basis for plant-specific bolting integrity programs for future plants, and recommends that this information be considered in preparing the new SRP Section in order to prevent bolting degradation and/or failure from occurring and to ensure bolting integrity in future plants. In addition, it is also recommended that information and requirements contained in previous NRC Bulletins, Generic Letters and NRC Information Notices issued to operating plant owners regarding threaded fasteners be reviewed by the NRR staff, and pertinent information or requirements be factored in the new SRP Section when applicable.

Refer nces

1. EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," Volumes 1 and 2, prepared by Applied Science & Technology for EPRI, April 1988.

2. EPRI NP-5067, "Good Bolting Practices-A Reference Manual for Nuclear Power Plant Maintenance Personnel, Volume 1: Large Bolt Manual," and Volume 2: "Small Bolts and Threaded Fasteners," EPRI, 1987 and 1990, respectively.

3. EPRI NP-6316, "Guidelines for Threaded-Fastener Application in Nuclear Power Plants," prepared by Looram Engineering, Inc., for EPRI, EPRI, July 1989.

4. NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," U. S. NRC, to be published.

5. NUREG- , Regulatory Analysis for the Resolution of Generic Safety Issue No. 29, "Bolting Degradation or Failure in Nuclear Power Plant." (to be published).

ENCLOSURE 5

NUREG-1339

Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

Richard E. Johnson

NUREG-1339

Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants

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ABSTRACT

This report describes the U.S. Nuclear Regulatory Commission's (NRC's) Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants," including the bases for establishing the issue and its historical highlights. The report also describes the activities of the Atomic Industrial Forum (AIF) relevant to this issue, including its cooperation with the Materials Properties Council (MPC) to organize a task group to help resolve the issue. The Electric Power Research Institute, supported by the AIF/MPC task group, prepared and issued a two-volume document that provides, in part, the technical basis for resolving Generic Safety Issue 29. This report presents the NRC's review and evaluation of the two-volume document and NRC's conclusion that this document, in conjunction with other information from both industry and NRC, provides the bases for resolving this issue.

CONTENTS

ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AIF	Atomic Industrial Forum
AISC	American Institute of Steel Construction
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BTC	Bolting Technology Council
B&W	Babcock and Wilcox
BWR	boiling-water reactor
CGWT	cylindrically guided wave technique
EPRI	Electric Power Research Institute
GSI	generic safety issue
HSLA	high-strength, low-alloy (steel)
IE	(Office of) Inspection and Enforcement (NRC)
IGSCC	intergranular stress-corrosion cracking
ISI	inservice inspection
LAQT	low-alloy quenched and tempered (steel)
LOCA	loss-of-coolant accident
MPC	Materials Properties Council
NDE	nondestructive examination
NRC	Nuclear Regulatory Commission
NRR	(Office of) Nuclear Reactor Regulation (NRC)
NSSS	nuclear steam supply system
PWR	pressurized-water reactor
RCPB	reactor coolant pressure boundary
SCC	stress-corrosion cracking
SRP	Standard Review Plan
SOER	Significant Operating Event Report
USI	unresolved safety issue
UT	ultrasonic test
UTS	ultimate tensile strength
WOG	Westinghouse Owners' Group

1 INTRODUCTION

1.1 The Bolting Safety Issue

The bolting safety issue originally was an integral part of the Nuclear Regulatory Commission's (NRC's) Unresolved Safety Issue (USI) A-12, "Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports." Recognizing that the types and variety of failure mechanisms active in bolting safety problems were distinctly different from those to be addressed in structural steel supports, the NRC staff separated the bolting safety issue from USI A-12 (Ref. 1) and identified it as Generic Safety Issue (GSI) 29, "Bolting Degradation or Failure in Nuclear Power Plants." The identification of bolting integrity as a separate issue received impetus from the Advisory Committee on Reactor Safeguards (ACRS) in October 1981. The ACRS recommended that the NRC staff expand its concerns about stress-corrosion cracking (SCC) of high-strength, low-alloy (HSLA) steel bolts to include a more comprehensive approach to the degradation and failure of bolting and threaded fasteners. Separating the bolting issue from USI A-12 created no conflict with the other USI A-12 goals. No structural materials were liable to the same kinds of degradation mechanisms as bolting materials, and adequate fracture toughness criteria for bolting materials were available within the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).

The safety aspects of GSI 29 can be summarized as follows. Critical bolting applications in nuclear power plants constitute an integral part of the reactor coolant pressure boundary (RCPB) and include closure studs or bolts on reactor vessels, reactor coolant pumps, and steam generators. Failure of these bolts or studs could result in the loss of reactor coolant and jeopardize safe operation of the plant. Bolting also is an integral part of the pressure boundary and other safety-related systems, such as component supports and embedded anchor bolts or studs. In June 1982, the NRC staff issued Office of Inspection and Enforcement (IE) Bulletin No. 82-02 (Ref. 2). The bulletin required responsive actions by all pressurized-water-reactor (PWR) licensees because threaded fastener failures had shown an increasing frequency of occurrence and a variety of underlying mechanisms. Motivated by the issuance of NRC requirements regarding fastener integrity, the Atomic Industrial Forum (AIF) joined with the Materials Properties Council (MPC) to form the Joint AIF/MPC Task Group on Bolting, also in June 1982. This task group was composed of representatives from AIF member organizations—utilities, vendors, and architect-engineers—plus representatives from the Electric Power Research Institute (EPRI). The coordinated industry responses to Bulletin 82-02 and, earlier, to the "For Comment" version of NUREG-0577 (issued in October 1979) gave added emphasis to the importance of GSI 29.

When the NRC prioritized generic issues in November 1982, GSI 29 was assigned a high priority (Ref. 3).

1.2 Problem

The NRC noted (Ref. 4) that from 1964 to the early 1980s the incidence of reported failures in high-strength bolting in Class 1 component supports and other safety-related equipment increased. Common characteristics among the reported incidents included materials that were subjected to high sustained tensile stresses, out-of-specification pretorquing, an aqueous environment caused by high humidity, primary and borated water leakage, and materials that were overly hard and out of specification. The most frequently observed failure mode for the structural bolting was SCC. Both low-alloy quenched and tempered (LAQT) steels and maraging steels were degraded by SCC. A small number of overstress failures was traced to improper heat treatment or low-strength material. Pressure-retaining bolts failed from corrosion wastage*. Included in the RCPB components were steam generator manway closures, reactor coolant pumps, pressurizer manway closures, reactor vessel closures, chemical and volume control system isolation valves, check valves in the emergency core cooling system that form part of the RCPB, and other check valves. Some reactor vessel internal components, mainly the lower thermal shield bolts and upper core barrel bolts, were degraded, requiring extensive and expensive replacement of bolts in some plants.

Millions of threaded fasteners, including nuts, bolts, studs, and capscrews, are used in a nuclear power plant. The most important application of these fasteners is their use as an integral part of the RCPB, such as in pressure-retaining closures on reactor vessels, pressurizers, reactor coolant pumps, and steam generators. The NRC received reports of a number of degraded threaded-fastener incidents that involved the RCPB and major component supports. Although none of the reported incidents resulted in an accident,they do reflect an undesirable level of degradation in operating nuclear power plants by impairing the integrity of the RCPB or component supports.

Most of the reported bolting degradations were discovered either during refueling outages or scheduled inservice inspections (ISIs) or maintenance and repair outages.Thus far, bolting degradation has not caused an accident, and has not produced any immediate adverse effect on public health and safety. However, NRC is somewhat concerned that inadequate ISI of fasteners, either because of ineffective nondestructive examination (NDE) methods or failure to include fasteners in the ISI program, could contribute to the potential for reduction in the integrity of the RCPB and structural supports. Experience has shown that either wastage from borated-water corrosion or SCC can go undetected. Furthermore, such

*See Ref. 2 for a description of the wastage problem.

degradation in bolting important to the RCPB integrity could lead to a loss-of-coolant accident (LOCA) because of bolting failures.

1.3 Plan for Resolution

The NRC considered possible solutions to GSI 29 as part of the process for prioritizing generic issues. The NRC noted in Ref. 3 that bolting has a wide range of application in nuclear power plants and that no single solution to the problem is known. Therefore, to minimize potential bolting problems in new power plants, improvements in one or all of the five following areas could be recommended: design, materials, fabrication, installation, and inservice inspection. The NRC suggested that the efficiency and adequacy of the ISI program be emphasized.

The NRC action plan for GSI 29, as it finally evolved, included four tasks in its scope of work:

(1) Develop the technical bases for bolting application requirements by the NRC staff through a technical assistance contractor at the Brookhaven National Laboratory.

(2) Review licensees' responses to IE Bulletin No. 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants."

(3) Draft staff recommendations for proposed criteria and guidelines to be incorporated into the NRC Standard Review Plan (SRP).

(4) Develop a proposed plan for implementing bolting application requirements.

Meanwhile, the Joint AIF/MPC Task Group on Bolting became more active. The original charter of the task group was oriented toward a coordinated industry response to IE Bulletin No. 82-02 and to the bolting aspects of the related document, NUREG-0577, issued in 1979 as the "For Comment" version. The industry response was to emphasize (1) a bolting survey, (2) stress-corrosion-cracking susceptibility criteria, and (3) corrective actions to deal with the problem. However, through meetings of the task group, by itself and with NRC staff, a more comprehensive industry program evolved. The 19-task Generic Bolting Program was presented for review and comment to the parent AIF Subcommittee on Material Requirements in February 1983; the program was officially transmitted by the AIF to the NRC in July 1983. The basic objective, which was attractive to the NRC staff, was that the nuclear industry itself would provide the technical basis for resolution of GSI 29; therefore, NRC's activities regarding this generic issue were delayed until the NRC received industry's findings.

2 INDUSTRY RESOLUTION

2.1 Planned Program

The technical program eventually formulated by the AIF/MPC Task Group on Bolting was a comprehensive, 19-task action plan aimed at resolving GSI 29. The Electric Power Research Institute (EPRI) organized a matrix-managed Generic Bolted Joint Integrity Program to carry out the research. The results, to the extent that they provide relevant technical findings, are summarized in Section 2.2 of this report.

Early in its existence, the task group reviewed information on fastener service failures and categorized them into four basic groups:

Group I - Degradation or Failure of Pressure Boundary Bolting Caused by General Borated Water Corrosion (Wastage or Erosion/Corrosion)

Group II - Degradation or Failure of Pressure Boundary Bolting Caused by Stress-Corrosion Cracking

Group III - Degradation or Failure of Internals Bolting Caused by Fatigue and Stress-Corrosion Cracking

Group IV - Degradation or Failure of Supports and Embedment Bolting Caused by Stress-Corrosion Cracking, with Two Sub-Classes Separated at the Minimum Specified Yield Strength Level of 150,000 psi

Three of these groups (Group III was excluded) formed the basis for what was called the Generic Bolting Program. EPRI assumed the lead for technical integration and research support in this generic program. Work related to Group III (internals bolting) failures was assigned to individual vendor owners' groups. Resolution of the fastener integrity issue involved many disciplines. Input was needed from the areas of metallurgy, fracture mechanics, nondestructive examination, design, specifications and standards, quality assurance, manufacturing or quality control, corrosion engineering, joint design, and tensioning control. The AIF/MPC task group considered all of these disciplines. Assessment of priorities related to fastener applications led to the focus of action on primary pressure boundary components. The action plan designed by the task group encompassed the following 19 tasks (several of which were divided into sub-tasks). These 19 tasks were grouped under 3 headings: General Bolting Tasks, Pressure Boundary Bolting Tasks, and Tasks Associated with Internals Bolting.

2.1.1 General Bolting Tasks (Tasks 1 Through 9)

Task 1 — Assessment of Priorities and Safety Significance

Task 1.1. – To monitor bolting priority ranking and to assess the failure and success history for each of the four degradation or failure groups listed previously. (See Section 2.1 of this report.)

Task 1.2. – To conduct a pilot scoping study, under EPRI direction, on the use of decision analysis for bolting aimed at developing a methodology for determining the technical parameters that influence the likelihood of bolt failure.

Task 2 — Literature Survey of Fastener Corrosion

To perform a literature survey of carbon and alloy steel fastener corrosion in PWRs.

Task 3 — Stress-Corrosion Cracking

To study and evaluate the effects of water environments on quenched and tempered low-alloy steel bolting materials.

Task 3.1 – Fracture Mechanics Analysis. To develop stress intensity factors for realistic flaw shapes and loading conditions in bolts.

Task 3.2 – Data Review. To obtain detailed descriptions of failures involving stress-corrosion of HSLA material from previously unpublished accounts.

Task 4 — Include Hardness Test Data into the Bolting Database

To include data obtained by utilities from hardness surveys of installed and spare bolting in a bolting database and to assess impact of these data on the issue.

Task 5 — Bolting Database

To maintain a database containing hardness data and other properties of bolting materials and to update the database as necessary to support industry efforts.

Task 6 — Development of Bolting Specifications and Standards for Nuclear Power Plant Applications

Task 6.1 – To develop a general specification for bolting requirements that eventually could be adopted by nuclear utilities.

Task 6.2 – To initiate action in American Society for Testing and Materials (ASTM) Committee F-16 (responsible for structural bolting) to revise sampling requirements in new and existing specifications to be more consistent with end-product expectations.

Task 6.3 - To prepare a draft ASTM standard entitled, "Standard Test Method for Leeb Hardness Testing of Metallic Materials," based on Equotip hardness test experience.

Task 7 - ASME Code Bolting Requirements

Task 7.1 - To prepare a critique regarding ASME Code bolting requirements, particularly as related to pretensioning of both pressure boundary and structural bolting joints.

Task 7.2 - To review ASME Code Section III bolting requirements to determine the need for revising or improving.

Task 8 - Develop Field NDE Techniques to Detect Wastage and Stress-Corrosion Cracking

To focus pilot studies that were under way on the development of field techniques, utilizing advanced ultrasonic techniques to detect wastage or stress-corrosion cracking in pressure boundary and support fasteners.

Task 9 - Information Exchange

Task 9.1 - To hold bolting workshops to exchange information on industry efforts regarding bolting integrity.

Task 9.2 - To produce and distribute to utilities videotapes on the behavior and maintenance of flanged pressure boundary connections as aids to improving ' olting design, installation, and maintenance.

Task 9.3 - To produce a videotape on design, behavior, and tensioning practices as applicable to structural joints if warranted from the Task 16 results.

2.1.2 Pressure Boundary Bolting Tasks (Tasks 10 Through 17)

Task 10 - Screening Strategy and Corrective Action for Pressure Boundary Bolting

To develop a strategy for identifying bolts in pressure boundary applications that may be susceptible to boric acid corrosion or stress-corrosion cracking and recommend corrective actions.

Task 11 - Recommend ASME Code Section XI Changes

To review ASME Code Section XI requirements and send comments and recommendations to the code committees for action, including (1) appropriateness of Section XI size limits for inspection requirements;

2) provisions to ensure adequate visual inspection; and 3) assurance that NDE inspections are effective in de-ecting corrosion wastage and stress-corrosion cracking.

ask 12 — Recommend Research Programs to EPRI on Degradation of Fasteners

o recommend three projects to EPRI that would in-rease the understanding of (1) accelerated boric acid ttack of carbon and alloy steel fasteners, (2) the effect of MoS_2, and (3) sealants for PWR primary system compo-ents (the impact of the recommendations on the con-racted work under Task 3 also was evaluated).

ask 13 — Alternative Materials and Coatings

o recommend alternative materials and coatings and rovide guidance regarding selection criteria for the pur-ose of eliminating borated water corrosion concerns his task was included in the contracted work under ask 3).

ask 14 — Component Support Bolting Screening Criteria

o develop a strategy for identifying component support olts that may be susceptible to stress-corrosion cracking nd recommend plant-specific methods for resolving ndings regarding materials that require review.

ask 15 — Assess Fastener Integrity Based on Fracture Mechanics

o develop a technique to evaluate the integrity of bolting aterial in component support fasteners.

ask 16 — Preload Technology Assessment

o evaluate the need for high preloads, to identify poten-al relief in preload requirements, to investigate preload pplication techniques and resulting preload variability, nd to recommend optimum techniques. Also, to discuss ethodologies for estimating existing preloads based on nowledge of the tensioning techniques, sampling, or ome combination of information and to discuss risks of etensioning existing joints.

ask 17 — Develop UT Procedures for Inspection of Ultra-High-Strength Low-Alloy Maraging Steels

o develop a field procedure for ultrasonic test (UT) spection of ultra-high-strength bolts in the lower sup-ort feet of the Westinghouse-designed steam generator, sing Westinghouse Owners' Group (WOG) funding.

2.1.3 Tasks Associated with Internals Bolting (Tasks 18 and 19)

Task 18 – High-Strength Bolting

To conduct research to improve the stress-corrosion resistance of high-strength, age-hardenable, Ni-Cr-Fe alloys and A 453 Gr. 660 (A286) bolting materials and to investigate the influence of irradiation and stress and strain on the behavior of structural materials.

Task 19 – Owners' Groups Liaison

To maintain liaison with owners' groups to ensure that duplication of effort is minimized and that pertinent information on the efforts of the task group is exchanged.

The 19 tasks were modified during the progress of the program. Redistribution of effort reflecting reassessment of relative priorities among the tasks occurred.

2.2 Technical Findings

The results of the Joint AIF/MPC Task Group on Bolting Program, described in this section, constitute a recommended technical basis for resolution of GSI 29 by the nuclear industry. The program, outlined in Section 2.1 of this report, was presented in detail along with the results from the executed tasks in a two-volume report, EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants" (Ref. 5). The report was published in two volumes to make the results of the research easy to review and to aid utility engineers in addressing plant-specific bolting and fastener problems with a single source document. Volume 1 included the background information, a description of the action plan for the AIF/MPC Task Group on Bolting, the approach to resolution of the bolting issue and the basis for this resolution, summaries of the findings from the 19 action plan tasks (Section 2.1.1), and the conclusions and recommendations.

Volume 2 included more complete supporting references and data. Publication of the two-volume report completed a comprehensive, generic review and analysis program. A major finding indicated that the design of critical closure joint bolting involves enough redundancy to ensure that there is virtually no pressing cause for concern regarding bolting integrity. A brief review of the EPRI report by volume and section follows.

2.2.1 Basis for the Resolution

EPRI NP-5769, Volume 1.

- Section 1, "Introduction," provided an historical review and the planned tasks (i.e., those given in Section 2.1 of this report).
- Section 2, "Industry Resolution of the Bolting Issue," presented additional detail on the Joint Task

Group approach to the problem, the results obtained from research according to the task action plan, and brief discussions of two principal contributions of the work—development (1) of a generalized closure integrity model and (2) of joint leak-tightness criteria. The resulting information led to the conclusion that the technical basis for resolution of the generic bolting issue was reached.

- Section 3, "Pressure Boundary Bolting," concluded that closure integrity can be ensured through application of a leak-before-break criterion and provides three analyses (primary manway cover, reactor coolant pump main flange, and check valve flange) to illustrate the methods. The work led to a proposed ASME Code Section XI code case (see the discussion of Section 6, Volume 1, EPRI NP-5769, in this report) on inspection of bolted closures with observed or detected leakage.

- Section 4, "Structural and Component Support Bolting," presented results from Task 14 of the action plan. Component support bolting that may be susceptible to SCC was identified, and both generic and plant-specific review procedures were recommended. It was concluded that application by licensees of the proposed screening and disposition (of materials which failed to pass the screen) steps would be an adequate bolting integrity program and would serve to resolve GSI 29 with regard to component support bolting.

- Section 5, "Owners' Groups Summary," presented results from the Babcock and Wilcox (B&W) and Westinghouse Owners' Group programs on primary pressure boundary bolting. The B&W program lent further support to findings by others (see the discussion of Section 3, Volume 1, EPRI NP-5769, in this report) that a leak-before-break approach and conservative failure criteria can be used to ensure the integrity of bolted closures. Failure mechanisms included SCC and chemical wastage. A steam generator manway closure with SCC and a reactor main coolant pump closure with wastage were analyzed as examples to demonstrate the adequacy of this approach and these criteria. The WOG program originally included the following 10 tasks (Ref.6) (see Table 5-1, Vol. 1, EPRI NP-5769):

  (1) Determine the bolting material, number of bolts, bolt dimensions, and gasket material used for primary boundary closures (i.e., pumps, valves, steam generators, and pressurizers).

  (2) Provide installation procedures for bolting for primary boundary closures.

  (3) Catalogue service experiences for primary boundary closures, and identify service-sensitive closures based on utility input.

  (4) Follow the AIF/MPC bolting programs, provide liaison for the WOG plants, and prevent any duplication of effort.

  (5) Develop nondestructive methods for bolting for primary boundary closures.

  (6) Prepare specifications for primary boundary bolting, including quality assurance requirements for procurement, receipt, and preinstallation inspections.

  (7) Evaluate and qualify sealants for primary boundary closures.

  (8) Evaluate and qualify lubricants for primary boundary closure.

  (9) Establish the number of "failed" bolts in closures resulting in one-gallon-per-minute leakage (or limits set by technical specifications), and determine margins of safety for bolting in primary boundary closures.

  (10) Establish feasibility of having an inventory of bolting (considering Task 3) for primary boundary closures.

Of the original 10 tasks, the WOG actually sponsored the first 4 tasks; the other tasks were judged to be adequately covered by separate efforts sponsored by EPRI, ASTM bolting standards committees, ASME Code committees, or other MPC activities. The WOG did assume responsibility for Task 17 (see Section 2.1.2 of this report), the development of an ultrasonic field procedure applicable to ultra-high-strength bolts.

A discussion of the first 4 of the 10 tasks for the WOG's program on bolting follows (Ref. 6):

- Task 1 supported a leak-before-break approach to closure integrity by addressing the complete bolted closure rather than individual fasteners and resulted in publication of a user's manual.

- Task 2 investigations resulted in recommended installation procedures. To arrive at these procedures, the WOG compared several preload measurement techniques, including those for torque wrenches, stud heaters, stud tensioners, and the Bolt Gage (a Raymond Engineering, Inc., development). The Bolt Gage was the preferred technique.

- Under Task 3, from available data, the WOG listed and analyzed primary boundary closure leakage. It was concluded that (1) the available information useful in determining service-sensitive closures is limited, (2) the leak-before-break approach recommended by the AIF/MPC Task Group on Bolting is a sound engineering attack, and (3) the ASME Code should be changed to address bolt/stud flaw limits based on closure integrity and fastener redundancy.
- Task 4 provided for liaison with others in the AIF/MPC task group.

An NDE field procedure was developed for bolts under AIF/MPC Task 17 that provided *in situ* techniques for inspection of Westinghouse PWR steam generator support bolts. The procedure can obviate costs previously borne by utilities for removal and surface examination.

Section 6, "ASME & ASTM Codes and Standards," included the results of Tasks 6 and 7 of the action plan (Section 2.1 of this report). Subtask 6.1 was completed with the preparation of "Utility Recommendations and Guidelines for the Purchase Specification and Receipt/Installation Inspection Requirements for ASME Section III, American Institute of Steel Construction (AISC), American National Standards Institute ANSI/ASME B31.1, and ANSI B31.5 Bolts and Threaded Fasteners." The report is published in its entirety as Section 1 of Vol. 2, EPRI NP-5769. Subtask 6.2 addressed the advisability of changing the sampling requirements for structural bolting specifications under the jurisdiction of ASTM Subcommittee F16.02, and, at the time of publication of EPRI NP-5769, approval of the proposed changes was still pending. Subtask 6.3 resulted in preparation of the draft standard, "Standard Test Method for Equotip Hardness Testing of Metallic Materials," its submittal to ASTM Subcommittee E28.06, and its publication as Section 2, Vol. 2, of EPRI NP-5769. Task 7 resulted in two products: First, the many places in the ASME Code where rules are given for the design and construction of bolted joints, scattered among voluminous rules for welding and other fabrication methods, were listed and explained in Section 9, Vol. 2, EPRI NP-5769. Until such time as changes in the ASME Code make the Section 9, EPRI NP-5769, listing obsolete, it will serve as a useful single reference source. Second, the results of a review of ASME Code bolting requirements were published as Section 10, "Critique of Bolt Preload Aspects of ASME and AISC Codes," Volume 2, EPRI NP-5769. The material in this section was limited to ASME Code treatment of bolt preload with respect to design, assembly methods, and quality control, even though additional aspects of fastener preload could have been considered.

- Section 7, "NDE of Bolting," presented the results of studies designed to attain the goal of developing field techniques as stated in Task 8 of the AIF/MPC bolting action plan (see Section 2.1 of this report). Conventional UT methods were evaluated and their limitations determined. EPRI funded three separate contracts and asked the contractors to develop and evaluate new techniques that lend themselves to field application and that are capable of detecting both SCC and wastage. One contractor developed acoustic resonance and reverberation techniques for detection of wastage, another contractor further developed the acoustic resonance technique to detect SCC, and the third contractor developed the cylindrically guided wave technique (CGWT) for detection of SCC as well as wastage.

  The reverberation technique used a spectrum analyzer to quantify the frequency content of a pulse-echo envelope and detect characteristic time spacing changes and then compared the reverberation spectrum of a target bolt to that of an unflawed bolt to detect degradation.

  The CGWT provided an inspection method applicable to most studs or bolts over a range of 16 in. to 112 in. (406 mm to 2,844 mm) in length and 1 in. to 4.5 in. (25.4 mm to 114 mm) in diameter. The technique could be used to detect cracklike defects as small as 0.05 in. (1.27 mm) deep in the threaded region of the bolt. In addition, the CGWT could be used to detect corrosion wastage greater than 25% of the bolt diameter.

- Section 8, "Lubricants and Sealants," included three sub-tasks as part of Task 12 in the AIF/MPC task plan (Section 2.1.2 of this report). Several projects and studies, described in EPRI NP-5769, provided useful data. The text of Section 8 was adapted from a more detailed report (Ref. 7). The influence of several lubricants on boric acid wastage was studied with results too varied to review in detail here. However, one result reported was the detrimental influence of MoS_2 and the difficulty of removing it from fasteners that have been exposed to service conditions. The studies of leak sealants and concerns derived from them led to several recommendations; principal among them was that the responsible design organization (e.g., ASME) should establish standards for leak sealants.

- Section 9, "Alternative Materials," the task identified as number 13 in Section 2.1.2 of this report, was drawn from the more detailed reports of "Stress-Corrosion Cracking of Alternative Bolting Alloys," EPRI RP-2058-12 (Ref. 8) and Section 3, Vol. 2, EPRI NP-5769. This work also is related to Task 3 of the action plan. The resulting four conclusions can be restated briefly as follows: (1) whereas low alloy steels are vulnerable to boric acid corrosion, other alloy steels generally are resistant; (2) galvanic corrosion, depending on specifics of the material composition and environmental chemistry, can occur, but data are needed for each combination if sensible decisions are to be made; (3) MoS_2 lubricant was shown to be a factor in laboratory corrosion testing when conditions favored the liberation of hydrogen sulfide, but its role in service-related failures remains to be clarified; (4) more K_{ISCC} data are needed if a damage-tolerant methodology is to be adopted.

- Section 10, "Training Package," consisted of a brief description of two EPRI-sponsored actions aimed at information exchange. First, a workshop was held November 2 through 4, 1983, at Knoxville, Tennessee. Participants included representatives from the U. S. Nuclear Regulatory Commission, the Atomic Industrial Forum, the Electric Power Research Institute, its contractors and consultants, and the nuclear power generating industry. The stated objectives included alerting industry to the NRC's generic bolting safety issue and the regulator's perspective of the issue. Also, speakers reviewed the AIF program and the EPRI efforts toward resolution, including bolting design criteria, codes and standards, specifications, fabrication, quality control, tools, procedures, and general bolting problems. Second, a set of three videotapes was produced and made available to any interested party. They are identified as "Electric Power Research Institute Pressure Boundary Bolting Problems; Part I: The Basics; Part II: Engineering Problems; Part III: The Mechanic and Bolting." Although they were aimed at the manager, the engineer, and the mechanic, respectively, viewed together, they constitute a rather complete tutorial on bolting.

- Section 11, "Conclusions and Recommendations," summarized the many conclusions derived from completion of the Joint AIF/MPC Task Group on Bolting 19-task program. The diverse disciplines and the many activities were joined and integrated to provide what the industry believed to be a basis for resolution of the NRC GSI 29.

2.2.2 Supporting Data for the Resolution

EPRI NP-5769, Volume 2.

- Section 1, "Utility Recommendations and Guidelines for the Purchase Specification and Receipt/Preinstallation Inspection Requirements for ASME Section III, AISC, ANSI/ASME B31.1, and ANSI B31.5 Bolts and Threaded Fasteners," presented recommended guidelines for utilities constructing or operating nuclear power plants, including certification, identification, NDE, and storage requirements for bolting material (bolts, studs, and nuts) to be used in permanent features. It also included recommended guidelines for receipt or preinstallation inspection designed to help ensure fastener integrity. Recommended guidelines were given for tightening fasteners when neither preloading, torquing, nor both are specified by other documents. The guidelines were written specifically for ASME Code Section III Code-of-Record plants. They were given as adequate for pre-ASME Code Section III Code-of-Record plants (i.e., ANSI B31.1 and B31.7), but, for plants of such vintage, the user was cautioned to consider the safety class of the system in which the bolting is used and to provide a commensurate level of quality. For instance, a plant having ANSI B31.1 as the code of record may choose to use ASME Section III Class 1 requirements for systems classified American Nuclear Society Safety Class 1.

- Section 2, "Standard Test Method for Equotip Hardness Testing of Metallic Materials," covered the use of the Equotip Hardness Tester to determine the Leeb hardness of metal components. The discussion included definitions, test procedures, instrument verification, test-block calibration, and a table of hardness conversion. As previously noted under Section 6, Vol. 1, EPRI NP-5769, the text of Section 2, Vol. 2, was a draft ASTM standard, submitted to ASTM Subcommittee E 28.06. As will be explained in Section 3 of this report, the NRC understands the need for *in situ* hardness measurement as part of a program by licensees to ensure conformity to codes and standards, and the NRC agrees that properly conducted Leeb hardness tests can be part of that program. However, Section 2 of EPRI NP-5769 contained what appear to be technical errors because tabulated hardness conversion values disagree with published ASTM Standards. The apparent disagreements must be clarified, presumably by ASTM Subcommittee E 28.06.

- Section 3, "Evaluation of Bolting Experiences in Primary Pressure Boundary Closures," presented the results of compiling and analyzing 125 incidents of bolting failure reported by nuclear utilities. The principal failure mechanisms, in order of decreasing importance, were boric acid corrosion (wastage), maintenance damage, corrosion pitting, and stress-corrosion cracking. Not included in the analysis were a number of flange bolt problems in the control rod drive mechanism that were judged to be related mainly to one nuclear steam supply system (NSSS).

Although the fastener rejection rate* varied with the component and generally was small (not more than about 10%), the rate was sufficient to justify ranking NRC GSI 29 as a high priority task.

Section 4, "Sampling Inspection and Acceptance Criteria for Bolted Connections," provided a statistical evaluation of fastener loads. The evaluation addressed the main concerns in highly stressed high-strength fasteners, that is, failures that were due either to the external load exceeding the preload across the joint (overload) or to stress-corrosion cracking. If the preload of a given fastener were known, then one could decide about preload adequacy simply by checking whether the preload is inside or outside the acceptable design range for that fastener. Because uncertainties exist about the actual value of preload, the deterministic checking procedure must be replaced with a probabilistic criterion.

Ideally, standards for bolted connections susceptible to overload and stress-corrosion cracking should be set by establishing maximum acceptable probabilities for the occurrence of each failure mode, based on the severity of the consequences. To apply such standards, uncertainties on the external loads, the state of preload, and the maximum flaw size should be quantified. Unfortunately, the state of knowledge (e.g., about SCC) and current deterministic practice make it impossible to fully implement probabilistic standards. A semi-probabilistic format, explicitly recognizing uncertainties on fastener preload, but avoiding failure probability calculations, was proposed. The uncertainty is changed by sampling inspection, for which a simple method of sample size determination and uncertainty updating was proposed, consistent with the format of the acceptance criteria.

Section 5, "Nuclear Structural Bolting Preload Evaluation," reported on the results of completing Task 16 (see Section 2.1.2 of this report). The task consisted of evaluating the need for high preloads, identifying potential relief in preload requirements, and investigating preload application techniques and variability. Section 4 of EPRI NP-5769 provided the statistical nature of the preloading process, and Section 5 evaluated existing preload design requirements, the relationship of the specified joint preload to the minimum preload required to carry design loads, and the effect of potential loading relief on minimum preload requirements for one heavy component support structural joint. The report discussed conclusions that were reached about design criteria, load relief, preload range acceptability, preload estimation, and preload accuracy.

- Section 6, "The Bolting Database: An Example of a Numeric Database Application in the Nuclear Power Industry," briefly discussed the nature of the database, the stored information and classification scheme, access, software, and applications.

- Section 7, "Assessment of Field Hardness Measurements on Low-Alloy Quenched and Tempered (LAQT) Bolting Materials at Midland," presented the collection (from four nuclear sites) and analysis of LAQT fastener steels. Significant deviation from specification requirements was observed. It was estimated that the portion of bolts at one site (Midland) with a hardness indicating a susceptibility for SCC was less that 1% of the total population; not a serious concern.

- Section 8, "Good Bolting Practices," briefly reviewed the two reference manuals for nuclear power plant maintenance personnel that were developed under EPRI sponsorship and were intended for rapid-access field or office use by utility staff who must disassemble and assemble bolted joints in nuclear power plants. This section described bolting practices that should help staff members identify, understand, and solve (or minimize) bolted joint problems such as leaks, vibration loosening, fatigue, and stress-corrosion cracking.

  The first of the manuals was entitled, "Good Bolting Practices: Large Bolt Manual;" the second was entitled, "Good Bolting Practices: Small Bolt Manual." The manuals described the problem-reducing steps in order of increasing complexity and cost, recognizing that the options available to maintenance personnel are generally limited.

  The manuals were not intended for use by designers; therefore, the theories behind the recommended procedures were not discussed at any length. The encyclopedia format for the manuals was intended to make the topics easy to locate. Topics were listed alphabetically and identified by legends printed in bold face. Each topic was described briefly, with typical data, if pertinent, and with cross-references to related topics, also in bold-face type.

- Section 9, "Bolting Rules of the ASME Boiler and Pressure Vessel Code," presented a detailed, point-by-point, review of the ASME Code with explanations, interpretations, and suggestions for improvement. The many scattered bolting requirements were collected in this one section of the EPRI report to provide a source document for reference. This was noted in the discussion of Section 6, Vol. 1, EPRI NP-5769.

*Rejection rate was the relative number of failures, degradations, inspection call-outs, etc., as a percent of the total fasteners inservice.

- Section 10, "Critique of Bolt Preload Aspects of ASME and AISC Codes," is a companion piece to Section 9, Vol. 2 (and was cited in Section 6, Vol. 1) of the EPRI report. The stated assignment was "to critique existing preload sections of the ASME Code." The Joint AIF/MPC Task Group on Bolting posed two questions to guide this effort:

  (1) Do provisions of the ASME Code contribute to the types of bolting failure experienced in the last decade or so by the nuclear industry?

  (2) Do omissions in the ASME Code contribute to the types of bolting failure experienced by the nuclear industry?

  The investigators were directed to limit the responses to preload aspects of bolting problems. The review identified several provisions of the AISC and ASME Codes that could be troublesome. The points raised and the rectifications suggested were too numerous to be reported here, but are described in EPRI NP-5769. In addition to general observations concerning the ASME Code, preload philosophy, preload and installation method codification, education, quality control, and the role of the mechanic, five specific problems were cited and solutions offered. Four appendices completed the section; they went into greater detail on specific problems and gave support to the conclusions and recommendations stated in the text of Section 10. The titles of the appendices follow:

  (1.) Appendix 10A, "AISC Specification for Structural Joints Using ASTM A325 or A490 Bolts"

  (2.) Appendix 10B, "ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF"

  (3.) Appendix 10C, "ASME Boiler and Pressure Vessel Code, Section VIII, Division 1"

  (4.) Appendix 10D, "Comparison of ASME Codes on Pressure Boundary Bolting"

- Section 11, "Evaluation Procedure for Assuring Integrity of Bolting Material in Component Support Applications," is the companion piece on structural bolting to the RCPB discussion, Section 3, Vol. 2, EPRI NP-5769. The Section 11 presentation describes an evaluation procedure for general application to bolting products used in component supports and fabricated from steels commonly used for support bolting. The evaluation procedure could be used to justify serviceability of questionable materials. It was anticipated that this section would be useful to a utility as part of a plant-specific plan to deal with materials that require some evaluation under the generic issue.

  The primary objective was to present the procedural steps and required information to determine allowable bolt loads to avoid SCC under steady-state or long-term normal operating conditions. Allowable bolt stress as a function of material hardness, bolt size, and thread pitch could be determined with the procedures. The allowable bolt stress then could be compared with actual bolt stresses calculated for the design. A requirement of the procedure was that hardness testing be performed on the population of bolts so that hardness limits could be statistically determined. Also, as part of the evaluation objective, allowable bolt stresses to prevent fracture under short-term (accident) loads must be established when low toughness was implied by the hardness data.

- Section 12, "Alternate Alloys," consisted of brief reports on five separate research projects, all sponsored by EPRI and dealing with steel corrosion in nuclear reactor environments. Each project had been reported in more detail elsewhere; the five published documents were cited in the EPRI NP-5769, Volume 2, reference list.

  The Combustion Engineering Project grew out of interest in fastener corrosion and is entitled, "Literature Survey of Carbon and Alloy Steel Fastener Corrosion in the PWR Plants." The objective of this project was to determine the extent of low-alloy steel fastener corrosion problems in the domestic PWR industry and to review available data in the literature on boric acid corrosion and stress-corrosion cracking of fasteners. Service failures from both mechanisms were collected and analyzed. A common factor in six SCC events involving steam generator primary manway closure studs, which pose a potential for a LOCA, was the use of MoS_2 lubricant. Decomposition at high temperatures can yield hydrogen, which can induce SCC in HSLA steels even at low concentrations.

  The Battelle, Columbus, Laboratories Project was closely allied to the Combustion Engineering Project. It involved a review of joint failures in nuclear components from either boric acid wastage or SCC. The primary objective was to determine if austenitic, age-hardenable materials could be used for bolting applications. A secondary objective was to review the boric acid corrosion and stress-corrosion cracking behavior of currently used low-alloy steels and issues relating to lubricants and sealants. Based on the review, recommendations were made for further work to improve the industry's capability for dealing with the bolting problem. It was concluded that austenitic materials seem to be resistant to boric acid wastage but vulnerable to SCC. The report presented so many qualifications regarding the use of high-strength high-alloy steels that Battelle could hardly be accused of recommending them. Much of the report discussed conditions that can lead to failure of low-alloy steels without recommending alternative resistant materials.

The Materials Engineering Associates Project was part of a failure analysis of Type 410 stainless steel valve studs purchased to ASTM specification A 193, Grade B6, with a supplemental requirement of 125 ksi tensile strength specified by the utility. The utility's investigation focused on improper heat treatment of the studs, resulting in temper embrittlement. The embrittlement permitted SCC, with stud failure occurring once the critical flaw size was achieved. Mechanical property tests confirmed that the material exhibited low fracture toughness. In combination with the experimentally determined rather high tensile strength and somewhat reduced ductility (less than 50% reduction in area in three of six specimens), the studs would be vulnerable to SCC.

The Westinghouse Electric Project sought a solution to the cracking of age-hardenable Ni-Cr-Fe alloys in PWRs and boiling-water reactors (BWRs). Several instances of stress corrosion or, in some cases, corrosion fatigue in bolts, beams, and pins were observed in reactors using Alloys X-750, I-718, and A 286. The object was to examine the three alloys in different heat-treated conditions. Alloy X-750 with increased amounts of zirconium, which previously had been shown to be beneficial, also was included. Stress-corrosion cracking studies, involving both crack initiation and crack propagation specimens in PWR and BWR conditions, were conducted on alloy X-750 in 11 conditions, alloy I-718 in 2 conditions, and alloy A 286 in 2 conditions. The operating conditions of BWRs were shown to be more detrimental to the alloys than operating conditions of the PWRs. Alloy X-750 exhibited the most resistance to cracking (or propagation) in one of the several heat treatments that were applied, but, in a different condition, it was the least resistant. Long-term (more than 10,000 hours) tests are continuing.

The Babcock and Wilcox Project was a companion to the Westinghouse project, using the same high-strength, age-hardenable Ni-Cr-Fe alloys, X-750, I-718, and A 286. Service failures were attributed to fatigue, corrosion fatigue, and intergranular stress-corrosion cracking (IGSCC). Susceptibility to failure by these mechanisms depended strongly on the metallurgical condition produced by thermo-mechanical processing. The project included detailed microstructural characterization and corrosion testing of the alloys subjected to 15 different combinations of melting practice and thermomechanical processing. As in the Westinghouse study, preliminary findings indicated that Alloy X-750 had the best resistance to SCC when in a particular metallurgical condition. As in the companion study, the conclusions were not solid and unambiguous except for recommending further studies.

- Section 13, "Standard Specification for Supplemental Requirements for Structural Fasteners for Nuclear Applications," consisted of some background and introductory material and the proposed ASTM standard: *F XXX—"Standard Specification for Quality Assurance and Inspection Requirements for Structural Fasteners for Nuclear and Other Special Applications." Experience with fasteners has created several concerns. The draft specification includes requirements for nuclear fasteners as follows:

  - Establish sampling and quality levels for all series of structural fasteners on a uniform basis.
  - Establish mandatory lot control and traceability of fasteners. By maintaining such control, prevent mixing and possible contamination of parts intended for nuclear systems.
  - Require positive identification and source of fasteners intended for nuclear system as evidence of adherence to required quality level.
  - Require preferential full-scale testing of finished fasteners in lieu of reliance on possible machined coupons from fasteners. Actual full-scale testing is designed to confirm integrity of finished fastener not possible by coupon evaluation.
  - Permit utilization of state-of-the-art technology and beneficial effects of heading and thread rolling by specific callout. Such other major industries as automotive and aerospace have similarly mandated such requirements.
  - Recognizing the potential long-term degradation resulting from the presence of discontinuities such as cracks and seams, establish specific requirements to define acceptable and rejectable criteria for nuclear system use.

This very important standard, now in the hands of the cognizant ASTM committee, was supported by the Joint AIF/MPC Task Group on Bolting.

*This standard is being developed; the number will be assigned after it is completed and approved.

- Section 14, "The Bolting Technology Council" (BTC), provided a brief description of the Council, its activities, and its makeup. The BTC is affiliated with the Materials Properties Council, Inc., formerly the Metals Properties Council, which provides administrative services as required. The BTC was formed to provide opportunities for threaded fastener and tool users to engage in a variety of cooperative activities. As stated in its bylaws, the purpose of the Council is "to sponsor research; to recommend practices; to act as a clearing house for information; and to provide education concerning the art and science of the installation and behavior of mechanical fasteners and their interaction with the joints they are used in." As anyone who has attempted to understand bolted joint behavior will realize, the task selected by the BTC is not a simple one, nor will the effort be inexpensive. Because of the magnitude of the job, members felt that it would be desirable to pool a portion of their technical and financial resources and attack the problems jointly. Results achieved by cooperative efforts, furthermore, often have greater credibility, are more widely accepted, and are most economically achieved. The Council expects to provide benefits to industry through interaction with recognized experts in bolting technology, opportunities to participate in seminars and symposia, opportunities to share in cooperatively funded research to be planned, monitored, and directed by BTC groups, and opportunities to review publications and research results well before general release. It is anticipated that the BTC will identify unresolved bolting problems recognized now and as they arise from experience in the future. Through its resources in personal expertise and in financial assistance, the BTC will be instrumental in developing solutions to generic bolting issues.

3 CONCLUSIONS

The NRC staff has reviewed the technical findings developed by the industry and presented in EPRI NP-5769 (Ref. 5) as well as other relevant industry-generated information. The staff has concluded that the technical basis for resolution of GSI 29 is available.

The conclusion that GSI 29 can be resolved is based on the availability of several relevant documents. Actions taken by the Joint AIF/MPC Task Group on Bolting resulted in EPRI NP-5769, three video training films (see Section 10, Vol. I, Ref. 5) and the Good Bolting Practices reference manuals, Vols. 1 and 2 (see Section 8, Vol. 2, Ref. 5). Industry representatives established the Bolting Technology Council (an MPC affiliate) to take the lead in sponsoring bolting research, recommending practices, gathering and providing information, and promoting education on installation, application, behavior, and interactions of fasteners. The Institute of Nuclear Power Operations can be expected to act in response to potentially unsafe conditions as in the past when Significant Operating Event Report (SOER) No. 84-5 on bolting (Ref. 9) was issued. Further refinement in codes and standards will be provided by the responsible committees in ASME and ASTM (e.g., Committee F16 on Fasteners).

During the period in which GSI 29 was being resolved, the NRC took several additional steps that must be factored into the resolution of the issue. Incidents of threaded fastener degradation and failure from October 1969 to March 1982 were identified and analyzed (see Ref. 4). Five documents were prepared based on results of technical assistance contracts in support of the bolting generic issue (Refs. 10 through 14). In addition, action items included the following NRC notices, bulletins, and generic letters:

- IE Bulletin 74-03, "Failure of Structural or Seismic Support Bolts on Class 1 Components," April 29, 1974.
- IE Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," March 8, 1979.
- IE Bulletin 79-07, "Seismic Analysis of Safety-Related Piping," April 14, 1979.
- IE Bulletin 79-14, "Seismic Analysis for As-Built Safety-Related Piping Systems," July 2, 1979, (also: Revision 1 of page 2 of 3, July 18, 1979; Supplement 1, August 15, 1979; Supplement 2, September 7, 1979).
- IE Bulletin 82-02 (Ref. 2), which resulted in W. Anderson and P. Sterner, "Evaluation of Responses to IE Bulletin 82-02," NUREG-1095, May 1985.
- NRC Compliance Bulletin 87-02, "Fastener Testing to Determine Conformance with Applicable Material Specifications," November 6, 1987 (later: Supplements 1 and 2).
- NRC Bulletin 89-02, "Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design," July 19, 1989.
- NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 17, 1988.
- NRC Generic Letter 89-02, "Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products," March 21, 1989.

- IE Circular 78-14, "HPCI Turbine Reversing Chamber Hold Down Bolting," July 12, 1978.
- IE Information Notice 80-27, "Degradation of Reactor Coolant Pump Studs," June 11, 1980.
- IE Information Notice 80-29, "Broken Studs on Terry Turbine Steam Inlet Flange," August 7, 1980.
- IE Information Notice 80-36, "Failure of Steam Generator Support Bolting," October 10, 1980.
- IE Information Notice 82-06, "Failure of Steam Generator Privacy Side Manway Closure Studs," March 12, 1982.
- IE Information Notice 86-25, "Traceability and Material Control of Material and Equipment, Particularly Fasteners," April 11, 1986.
- IE Information Notice 86-108, "Degradation of Reactor Coolant System Pressure Boundary Resulting from Boric Acid Corrosion," November 1986 (later: Supplements 1 and 2).
- NRC Information Notice 87-56, "Improper Hydraulic Control Unit Installation at BWR Plants," November 4, 1987.
- NRC Information Notice 88-11, "Potential Loss of Motor Control Center and/or Switchboard Function Due to Faulty Tie Bolts," April 7, 1988.
- NRC Information Notice 89-22, "Questionable Certification of Fasteners," March 3, 1989.
- NRC Information Notice 89-59 and Supplement 1, "Suppliers of Potentially Misrepresented Fasteners," August 16 and December 6, 1989.
- NRC Information Notice 89-70, "Possible Indications of Misrepresented Vendor Products, October 11, 1989.

In various ways, these NRC notices, bulletins, letters, and circular will influence actions that the NRC or licensees will need to take in the wake of the resolution of GSI 29. Although they do not, individually or collectively, form a basis for the resolution, neither will these documents nor the responses made to them change as a result of the resolution of GSI 29.

The NRC staff concludes that all of the available information that has been discussed in this report (from industry and regulatory sources combined) provide a sufficient technical basis to resolve GSI 29.

It must be understood that although the NRC staff recognized the value of the several products of the industry effort (the work of the Joint AIF/MPC Task Group on Bolting) in helping to resolve GSI 29, that recognition does not constitute unqualified endorsement of their technical content. The NRC staff found technical disagreement with several specific discussions in EPRI NP-5769, the three videotapes on training, and the Good Bolting Practices reference manuals. The technical disagreements, except for the following, however, generally were not important enough to mention.

First, the staff notes that a general plan for evaluation of bolting integrity can be derived from Section 11, Vol. 2, EPRI NP-5769. Section 11, "Evaluation Procedure for Assuring Integrity of Bolting Material in Component Support Applications," was written to fulfill a specific assignment for the Joint AIF/MPC Task Group on Bolting. With appropriate modifications, the procedure could serve for other than component-support bolting. The screening process should use fastener material properties and fracture mechanics analyses to ensure that safety-related fasteners are unlikely to be susceptible to stress-corrosion cracking. Material properties should be experimentally verified rather than assumed to be as specified.

Second, and closely related to the first comment, inconsistencies found in EPRI NP-5769 regarding the criteria for categorizing bolting steels according to strength must be reconciled. Categorization should be based only on the actual measured yield strength, S_y, of the material (or S_y determined by conversion of measured hardness values) and not on the specified minimum yield strength. The justification for this position is that high-strength steels are vulnerable to SCC. A bolt made of high-strength steels may be obtained through an order which specifies a relatively low-yield strength, but by improper heat treatment, for example, the bolt may develop an actual strength far in excess of the minimum specified. Specifically, that high-strength bolts should be those with $S_y \geq 150$ ksi; medium-strength bolts should be those with 120 ksi $< S_y <$ 150 ksi. The following portions of EPRI NP-5769 need to be modified in order to make them consistent with the above definitions:

- In Vol. 1, Section 4, page 4-3, bolting steels are categorized as "high strength" if: " $S_y >$ 150 ksi, where S_y is the yield strength" (compare the greater-than symbol to the greater-than-or-equal-to symbol recommended). At the same location, medium-strength materials are identified as those with 120 ksi $< S_y <$ 150 ksi, which would be consistent except for the explanatory text that follows on page 4-4: "Therefore, it seems appropriate for the industry to examine the use of materials with specified minimum yield strengths greater than 150 ksi" (emphasis added). The same words are used on page 4-5 at two places.

- On page 4-4, a proposed category is defined by "the range of 120 ksi to 150 ksi specified minimum yield strength" showing that the use of S_y on page 4-3 was not to be taken literally.

- In Vol. I, Section 11, "Conclusions and Recommendations," page 11-5, the words "minimum specified" are used again.

- On the next page (p. 11-6) one finds "specified yield strength." An inconsistency arises in EPRI NP-5769, Vol. 2, Section 1, page 1-17 because the reader is advised to consider materials vulnerable to SCC if the minimum specified ultimate tensile strength (UTS) is greater than 150 ksi or if the actual UTS is greater than 170 ksi.

- Then, in NP-5769 Vol. 2, Section 7, page 7-2, we find: "...the proposed screening limit of $S_y \leq 150$ ksi (1034 MPa)" although in the preceding paragraph the words "specified minimum yield strength" were used to describe the strength range of 120-150 ksi (827-1034 MPa).

A more careful reading might reveal more discrepancies or inconsistencies. For the reasons previously given, the criterion of actual yield strength, $S_y \geq 150$ ksi should be used as the level for consideration of SCC vulnerability.

Third, the data listed in Table 2-1, Vol. 2, EPRI NP-5769 are questionable. Indexing off the values of Rockwell C-scale hardness as given, the corresponding values of Vickers hardness numbers do not agree with those given in the ASTM Standard E 140. From the same R_C start, the corresponding values of tensile strength do not agree with values given in the ASTM Standard A 370. Such errors (there are typographical mistakes, as well) also make the hardness conversions listed in Table 11A-1 of EPRI NP-5769 suspect; they should be audited. Accepting the ASTM standards as the authority, the hardness conversions and hardness-tensile conversions in EPRI NP-5769 should be treated skeptically. Since Table 2-1 was to be part of a draft ASTM standard, the responsible ASTM committee can be expected to make such corrections as may be necessary. Until Leeb hardness values and conversion tables have been incorporated in a standard test method by the ASTM, they should be used "for information only" and not be accepted as evidence in licensing actions or in safety evaluations.

Fourth, the indictment against MoS_2 as a lubricant (found on page 3-5 of EPRI NP-5769, Vol. 2) deserves more emphasis. Facts gleaned from some service failures and from laboratory examinations (Ref. 12) clearly show that MoS_2 is a potential contributor to SCC, especially when applied to high-strength bolting steels. One of the problems posed by MoS_2—difficulty in removing it from parts that have been in service (see page 8-3, Vol. 1., Ref. 5)—may be close to being resolved. Whereas Czajkowski (Ref. 12) found that CS_2 will remove MoS_2, handling CS_2 poses some problems. More recently, tests by Czajkowski of samples of "citrus-based cleaners" were subjected to a cleaning task similar to that reported in Ref. 12, and it was evident that the sulfur component (the active SCC ingredient) had been effectively removed (Ref. 15). Providing that the citrus-based cleaners, themselves, are not SCC promoters, an answer to the MoS_2 cleaning problem may be at hand.

Fifth, although the fracture mechanics analyses by Cipolla cited in Section 9, Vol. I, EPRI NP-5769, are useful and could well be employed in engineering problems where values for the stress intensity factor, K_I, are needed, other more recent results are available. In a report published in 1988, "Review and Synthesis of Stress Intensity Factor Solutions Applicable to Cracks in Bolts" (Ref. 16), values for K_I for cracks in round bars, both threaded and unthreaded, subject to either tension or bending, were reviewed. Available solutions were synthesized into forms appropriate to analyses of bolts and studs. The K_I solutions published in Reference 16 should be used in fracture mechanics analyses of threaded fasteners.

The importance of maintaining adequate traceability* and control of material of fasteners at nuclear power plants was set forth in IE Information Notice No. 86-25 (Ref. 17). Because plant-specific bolting integrity programs should include steps to ensure bolting traceability and material control and to prevent introduction of incorrect or defective materials or components, the central ideas from this notice follow:

Awareness of 10 CFR Part 50, Appendix B, Criterion VIII, "Identification and Control of Materials, Parts, and Components," and applicable codes and specifications is important. Measures have been established and implemented by the NRC for identification and control of materials, parts, and components and for traceability both to the approved design basis and to the source. It is important that required identification of items be maintained by heat number, part number, serial number, or other appropriate means, either on the item itself or on records traceable to the item as required, and that required markings be on the item.

*In Attachment 2 to NRC Bulletin No. 88-10, November 22, 1988, verifiable traceability was defined as (with minor editing for this report): Documented evidence such as a certificate of compliance that establishes traceability of purchased equipment to the manufacturer. If the certificate of compliance is provided by any party other than the manufacturer, the validity of the certificate must be verified by the licensee or permit holder through an audit or other appropriate means.

It is the licensee's responsibility to use qualified individuals to examine markings on material and equipment and to verify that the markings represent material and equipment as specified by the design drawings and specifications. In the case of fasteners, compliance with the applicable material specification (e.g., ASTM or ASME material and grade) is verified by required markings on bolts and nuts and certified material test reports or certificates of conformance as required by procurement drawings and orders and by applicable codes and specifications. When vendor-supplied equipment assemblies contain fasteners, it is important to verify compliance with approved vendor drawings and specifications and such other information as materials used for equipment qualification tests analyses. The required markings on material and equipment, including fasteners, not only must exist, but the markings must indicate the correct material and grade as specified.

The NRC staff resolved GSI 29 based on the findings presented herein, including the following three conditions.

First, all earlier NRC notices, bulletins, and generic letters that bear on the issues involved in bolting, degradation or failure, some of which were noted earlier in this section, should remain in effect.

Second, it was concluded that an effective means of ensuring bolting reliability, as recommended in Ref. 5, would be through development and implementation of plant-specific bolting-integrity programs. These programs should be comprehensive and include all relevant NRC requirements and guidance and the recommended positions of the industry-sponsored programs.

Third, it is recommended that a new section of the Standard Review Plan (SRP) be prepared to provide guidance to the staff for the review of future plants. The elements of the review would include all safety-related joint design, threaded fastener material selection, and programatic aspects dealing with bolting integrity during construction, operation, and maintenance, except for closure studs which are addressed in SRP Sections 5.3.1 and 5.2.3.

In light of the facts that many safety-related systems and components rely in large measure on fastener integrity and that there have been numerous reported instances of degradation or failure of threaded fasteners, completion of the studies under GSI 29 has led to the conclusion that fastener integrity needs to be procedurally controlled. The information reviewed in this report showed that the safety issue related to fastener integrity involves a very large number of parts in each plant, a number of potential failure mechanisms (therefore, a corresponding number of protective or corrective actions), and several technical and engineering disciplines. Although the resolution of GSI 29 was found to be rather complex, sufficient guidance is available to resolve this issue, mainly from EPRI NP-5769 (Ref. 5). Specifically, the NRC staff concurred with the recommendations and guidelines provided in Section 1*, Vol. 2, of EPRI NP-5769. The recommendations and guidelines apply to threaded fasteners with regard to certification, identification, nondestructive examination, storage and tightening procedures, except when storage and tightening procedures are specified in other design documents or drawings. Implementing Section 1 and other technical guidelines in the EPRI report would help ensure fastener integrity.

A comprehensive bolting integrity program for a nuclear power plant would include all safety-related bolting, especially bolting used to close the primary pressure boundary and used for component support.

Of particular importance to safety are component support fasteners in the onsite power distribution system, including those power sources, distribution systems, and vital supporting systems provided to supply power to safety-related equipment and capable of operating independently of the offsite power system. The onsite power system includes an ac distribution system, a dc power system, an uninterruptible ac power system, and the emergency (diesel generator) power system. Fasteners in the auxiliary feedwater system and its support systems are also important to safe operation of a plant.

The work done to resolve GSI 29 has shown that (1) existing requirements, (2) the implementation of leak-before-break criteria for RCPB joints (proposed in EPRI NP-5769, Volume 1, Section 3), and (3) the ongoing programs (e.g., implementation of USI A-46 and the development of individual plant examinations for external events) should adequately limit the risk resulting from, and minimize the severity of, the failure of safety-related bolting in current plants. However, licensees with operating plants could avoid many of the problems recorded in the past by developing and implementing plant-specific bolting-integrity programs that include current requirements and reflect the information and recommendations made by the industry-sponsored program managed by EPRI (with NRC staff exceptions as discussed in Section 3 of this report). New plant licensees, however, could meet stringent bolting requirements with only a very small cost increase if established before they begin operating their plants.

Guidance regarding bolting for staff reviewers in the NRC Office of Nuclear Reactor Regulation (NRR) performing safety reviews of all new nuclear power plants could be provided by a new section in the NRC Standard Review Plan. Such a section, entitled, for example, "Safety-Related Bolting," would expand the limited coverage on fasteners now included in the SRP and provide a

*The Section 1 title is: "Utility Recommendations and Guidelines for the Purchase Specification and Receipt/Preinstallation Inspection Requirements for ASME Section III, AISC, ANSI/ASME B31.1, and ANSI B31.5 Bolts and Threaded Fasteners."

systematic method for implementation of the staff position regarding the basis for resolution of GSI 29. As part of the resolution of GSI 29, the staff noted the absence of an SRP section on general reviews of bolting and recommended that one be prepared and issued.

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14. U.S. Nuclear Regulatory Commission, "Preloading of Bolted Connections in Nuclear Reactor Component Supports," NUREG/CR-3853, October 1984.

15. C. J. Czajkowski, Brookhaven National Laboratory, letter to Richard E. Johnson, U.S. Nuclear Regulatory Commission, August 4, 1988.

16. L. A. James and W. J. Mills, "Review and Synthesis of Stress Intensity Factor Solutions Applicable to Cracks in Bolts," *Engineering Fracture Mechanics*, Vol. 30 (No. 5), pp. 641-654, 1988.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

This report describes the U.S. Nuclear Regulatory Commission's (NRC's) Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants," including the bases for establishing the issue and its historical highlights. The report also describes the activities of the Atomic Industrial Forum (AIF) relevant to this issue, including its cooperation with the Materials Properties Council (MPC) to organize a task group to help resolve the issue. The Electric Power Research Institute, supported by the AIF/MPC task group, prepared and issued a two-volume document that provides, in part, the technical basis for resolving Generic Safety Issue 29. This report presents the NRC's review and evaluation of the two-volume document and NRC's conclusion that this document, in conjunction with other information from both industry and NRC, provides the bases for resolving this issue.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Bolting; Degradation; Failure; Generic safety issue; Resolution; Fastener integrity; Fracture; Fracture mechanics; Corrosion; Stress-corrosion cracking; Nuclear power plants.

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GSI-29 CRGR Package

Information Required by Section IV.B of the CRGR Charter

(i) The proposed generic requirement or staff position as it is proposed to be sent out to licensees. Where the objective or intended result of a proposed generic requirement or staff position can be achieved by setting a readily quantifiable standard that has an unambiguous relationship to a readily measurable quantity and is enforceable, the proposed requirement should merely specify the objective or result to be attained, rather than prescribing to the licensee how the objective or result is to be attained.

Answer:
As outlined in Enclosure 1 of the CRGR package, the staff is proposing to issue a generic information letter (Enclosure 2) together with NUREG-1339, "Resolution of Generic Safety Issue (GSI) 29: Bolting Degradation or Failure in Nuclear Power Plants," (Enclosure 5) to plants currently holding an OL or CP, to inform them of the technical findings and resolution of GSI-29. The letter does not require licensee action or response.

RES is however, recommending that a new Standard Review Plan (SRP) Section on "Safety-Related Bolting" should be developed by NRR for the review of future plants and be included in a future revision to the SRP. Specific recommendations for bolting-related topics to be addressed in the SRP have been transmitted to NRR (Enclosure 4). NRR would send any actual proposed changes to the SRP to the CRGR for their review.

With the issuance of the generic information letter and NUREG-1339, and the recommendation to develop a new Standard Review Plan Section, GSI-29 would be considered resolved.

(ii) Draft staff papers or other underlying staff documents supporting the requirements or staff positions. (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 reference material for his or her use.)

Answer:
Staff documents supporting the staff positions mentioned in (i) are the following:

- Regulatory Analysis (Enclosure 3)
- NUREG-1339, "Resolution of Generic Safety Issue 29" (Enclosure 5)

(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.

Answer:

The proposed staff position will neither increase nor reduce existing requirements. The proposed generic information letter (Enclosure ?) and the accompanying NUREG-1339 (Enclosure 5) do [illegible]st (but do not require) that the best way to resolve GSI-29 would be for utilities to:

(1) Implement the industry developed bolting integrity program as presented in two volumes of EPRI report NP-576[illegible] he volumes EPRI Good Bolting Practices Refer[illegible]ce Manual, and three video training tapes, (as discussed in NUREG-1339, with some exceptions and qualifications, the staff endorses the industry recommended actions).

(2) Continue their actions in accordance with commitments made in response to a number of generic letters and bulletins related to bolting issues as described in NUREG-1339.

(iv) The proposed method of implementation along with the concurrence (and any comments) of OGC on the method proposed. The concurrence of affected program offices or a explanation of any non-concurrences.

Answer:
As mentioned in (i) above, the proposed method of implementation will be through the issuance of a generic information letter together with NUREG-1339. NRR has concurred in the proposed generic information letter and OGC has expressed no legal objection to the generic letter. In addition, ACRS has reviewed the GSI-29 resolution and agreed that NUREG-1339 provided a satisfactory basis for the resolution of GSI-29. We anticipate a positive response from the ACRS to the proposed generic letter after their May 1991 meeting.

(v) Regulatory analyses generally conforming to the directives and guidance of NUREG/BR-0058 and NUREG/CR-3568. (This does not apply for backfits that ensure compliance or ensure, define, or redefine adequate protection. In these cases a documented evaluation is required as discussed in IV.B.(ix).)

Answer:

A regulatory analysis (Enclosure 3) was developed for the resolution of GSI-29, and it documents the rationale why no additional requirement should be imposed on the operating nuclear power plants.

Although value-impact studies on GSI-29 were performed by our contractors (Appendices A and B of Regulatory Analysis, Enclosure 3), the staff judged the studies to be inconclusive regarding a mandatory program on safety-related bolting for operating plants, and, therefore, additional requirements could not be justified in accordance with the provisions of 10 CFR 50.109 for operating plants (this will be discussed further in (vii)). In addition, based on (1) bolting operating experience in both nuclear and conventional power plants, (2) the actions already taken through bulletins, generic letters, and information notices, and (3) the industry proposed actions, the Regulatory Analysis concluded that a sufficient technical basis exists for the resolution of GSI-29.

Based on the above considerations, the proposed generic information letter and the accompanying NUREG-1339 suggest (but do not require) that the best way to resolve GSI-29 would be for utilities to: (1) implement the industry developed bolting integrity program and (2) continue their actions in accordance with commitments made in response to a number of generic letters and bulletins. For future plants, it was concluded that a new Standard Review Plan section should be developed to codify existing bolting requirements and industry-developed initiatives, including the development and implementation of a plant-specific bolting integrity program.

(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 only, OLs after a certain date, OLs before a certain date, all OLs, all plants under construction, all plants, all water reactors, all PWRs only, some vendor types, some vintage types such as BWR 6 and 4, jet pump and nonjet pump plants, etc.).

<u>Answer</u>

The proposed generic information letter and the associate NUREG-1339 will be issued to all nuclear plants currently holding an OL or CP. The new SRP section on "Safety-Related Bolting" to be developed by NRR would be applicable to all future plants.

(vii) For backfits other than compliance or adequate protection backfits, a backfit analysis as defined in 10 CFR 50.109. The backfit analysis shall include, for each category of reactor plants, an evaluation which demonstrates how action should be prioritized and scheduled in light of other ongoing regulatory activities. The backfit analysis 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;

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

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

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

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

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

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

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

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

(j) How the action should be prioritized and scheduled in light of other ongoing regulatory activities. The following information may be appropriate in this regard:

1. The proposed priority or schedule,
2. A summary of the current backlog of existing requirements awaiting implementation,
3. An assessment of whether implementation of existing requirements should be deferred as a result, and
4. Any other information that may be considered appropriate with regard to priority, schedule or cumulative impact. For example, could implementation be delayed pending public comment?

Answer

Two value-impact analyses were performed by contractors, one on the RCPB bolting and the other on safety-related bolting in systems other than the RCPB (more detailed discussion of these analyses can be found in Enclosure 3). Regarding the value-impact analysis of the RCPB bolting, the best estimate indicated that the proposed action had the potential to reduce risk by 9,819 person-rem for the whole industry. This was based on a best estimate of a reduction in core melt frequency of 2.73E-6/reactor-year for PWRs and 2.9E-7/reactor-year for BWRs. This magnitude of risk reduction is not considered by the staff to satisfy the 10 CFR 50.109 criteria that a required action results in a substantial increase in the overall protection of the public health. In the staff's opinion, these estimated values of risk reduction probably erred on the high side. This value-impact analysis did result in a best estimate cost-benefit ratio of $239 per person-rem, which is very favorable.

The best estimate analysis was based on the assumption that all carbon or low-alloy steel bolts would be susceptible to boric acid wastage and would be replaced by stainless steel bolts. Such a program would be quite expensive for plants already constructed and the staff feels and the study underestimated the cost. Furthermore, an extensive RCPB bolting inspection and replacement program (beyond that required by the Section XI of the ASME code and the requirements of IE Bulletin 82-02) might require increased duration of refueling outages. Those costs were not included in the study.

If more realistic cost estimates are employed, an increased cost-benefit ratio would result. In the staff's judgement, a more realistic estimate of the cost benefit ratio would exceed $1000/person-rem for plants currently holding an OL or CP.

The staff, therefore, concluded that a mandatory replacement program for RCPB bolting could not be justified for plants that currently have an OL or CP.

A study by another contractor examined the risk related to failure of safety-related bolting in systems other than the RCPB. Approximately ten safety systems were examined for risk sensitivity. In addition, the primary coolant system component supports also were examined in the risk analysis. Although this analysis was based on PWR plants, it is the staff's judgement that the results are also generally applicable to BWRs since the risk is dominated by seismic consideration.

The best estimate of core melt frequency of this study was 3.5E-5/reactor year and the corresponding public risk reduction was 7300 person-rem based on 67 operating PWRs. The corresponding cost-benefit ratio reported was $3700/person-rem.

This study has considerable uncertainties in the calculations of reduction in core melt frequency and person-rem, and the cost estimates. Considering the uncertainties, the staff judged the contractor proposed program of surveying, testing and replacement to be marginal for plants currently holding an OL or CP from the viewpoints of reduction in risk to the public and cost-benefit ratio. The staff, therefore, concluded that requiring a program such as the one proposed by the contractor for safety-related bolting other than RCPB applications <u>could not be justified</u> in terms of the 10 CFR 50.109 criteria for plants that currently hold an OL or CP.

The inconclusive nature of the contractors' value-impact analyses on GSI-29 regarding a mandatory program on safety-related bolting for operating plants prompted the staff to look into the operating experiences on bolting in nuclear and conventional power plants, especially those in the pressure boundary applications (see Section 5.b of Enclosure 3). The operating experiences indicated that: (1) leakage of bolted pressure joints is possible but catastrophic RCPB failure which will lead to significant accident sequences is highly unlikely, (2) a search of LERs and "Precursors' Reports to Potential Severe Core damage Accidents" (from 1985 to 1989) for events involving bolting which have a conditional core damage probability greater than 1E-6 yielded only 14 LER cases. Among these 14 cases bolting problems are generally only partial contributors to the precursors. A large number of these bolting problems were related to internals bolting, others are related to improper torquing, loose bolts and nuts, missing bolts, etc. The recommended bolting integrity program delineated in the resolution of GSI-29 is not designed to address all of these types of bolting problems. Problems related to loose or improperly aligned internal set screws, missing bolts, loose bolts or nuts, etc., are judged better handled by programs such as in-service inspection, regular maintenance, or the implementation of other on-going NRC programs, such as USI A-46 and Individual Plant Examinations for External Events (IPEEE), which address the inadequacies of supports and their bolting due to design and installation.

Based on the above discussion, the staff concluded that a mandatory program of surveying, testing and replacement of safety-related bolting could not be justified.

If there were any proposed action, it would be to require that utilities to implement the industry developed bolting integrity program. It was brought to the staff's attention that NUMARC issued a letter to its members on July 6, 1989, notifying them of the publication of EPRI Reports NP-5769 and the Good Bolting Practices Reference Manuals, and stating that they provide the industry's technical basis for resolution of GSI-29 and encouraged members to refer to those reports. In addition, INPO issued SOER 84-05, "Bolt Degradation or Failure in Nuclear Plants" on September 20, 1984, informing utilities of the general findings of the industry research on this issue and recommended five actions based on the findings. The staff was informed by INPO through NUMARC verbally that subsequent INPO audit indicated near 100% implementation by utilities on the recommendations of SOER 84-05. The staff therefore concluded that it is not justifiable to require therefore concluded that it is not justifiable to require the utilities to implement the industry program. Instead, the alternative of issuing a generic information letter to plants that currently have an OL or CP was chosen. This generic information letter suggests, but does not require, certain actions.

(viii) For each backfit analyzed pursuant to 10 CFR 50.109(a)(2) (i.e, not adequate protection backfits and not compliance backfits) 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) 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; and

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

<u>Answer</u>

Not applicable.

(ix) For adequate protection or compliance backfits evaluated pursuant to 10 CFR 50.109(a)(4)

(a) a document evaluation consisting of:

(1) the objectives of the modification
(2) the reasons of the modification
(3) The basis for invoking the compliance or adequate protection exemption.

(b) In addition, for actions that were immediately effective (and therefore issued without prior CRGR review as discussed in III.C) the evaluation shall document the safety significance and appropriateness of the action taken and (if applicable) consideration of how costs contributed to selecting the solution among various acceptable alternatives.

<u>Answer</u>

Not applicable.

(x) 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 or 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.

Answer

Not applicable.

(xi) For each request for information under 10 CFR 50.54(f) (which is not subject to exception as discussed in III.A) an evaluation that includes at least the following elements:

(a) A problem statement that describes the need for the information in terms of potential safety benefit.
(b) The licensee actions required and the cost to develop a response to the information request.
(c) An anticipated schedule for NRC use of the information.
(d) A statement affirming that the request does <u>not</u> impose new requirements on the licensee, other than for the requested information.

<u>Answer</u>

Not applicable.

(xii) An Assessment of how the proposed action relates to the Commission's Safety Goal Policy Statement.

Answer

Not applicable since no action was proposed.

Item No. 44 206

Rcv'd in AEOD 4/25/91
Forwarded to Members 5/1/91

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

APR 24 1991

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

FROM: C. J. Heltemes, Jr., Deputy Director for Generic Issues and Rulemaking, Office of Nuclear Regulatory Research

SUBJECT: FINAL RULEMAKING ON THE EMERGENCY RESPONSE DATA SYSTEM (ERDS)

Enclosed for review by CRGR is a final rulemaking package to amend Part 50 to establish regulations for the implementation of ERDS. This final rule is needed to ensure that the NRC receives timely and accurate data on a limited set of parameters whose values indicate the condition of the plant during a declaration of an alert or higher emergency classification.

The enclosed rulemaking package was reviewed by your committee in its proposed form in June 1990. The rule has not changed in its substance since your earlier review.

Please note that, in order to submit the rulemaking to the EDO in May 1991, we are forwarding this rulemaking package for your review while requesting review from the other offices. However, we expect to have received concurrence from the cognizant offices when we meet with you to discuss the package. The RES contact on this rulemaking is Mark Au (X23749).

CJ Heltemes
C. J. Heltemes, Jr., Deputy Director
for Generic Issues and Rulemaking
Office of Nuclear Regulatory Research

Enclosure:
As stated

~~9104230115~~

PREDECISIONAL

Date		SECY-91-

FOR: The Commissioners

From: James M. Taylor, Executive Director for Operations

Subject: EMERGENCY RESPONSE DATA SYSTEM

Purpose: To obtain Commission approval of a notice of final rulemaking.

Issue: Establishment of Nuclear Regulatory Commission (NRC) regulations for implementing the Emergency Response Data System (ERDS).

Background: The Three Mile Island (TMI) accident in 1979 prompted the NRC to substantially improve its capability to acquire data on plant conditions during emergencies. The staff developed alternative conceptual approaches, and identified the Emergency Response Data System (ERDS) as the most feasible option that would meet the NRC's need for data acquisition during nuclear emergencies (SECY-84-481). The ERDS would utilize electronic data transmission systems already being developed by licensees for their own emergency response facilities.

The Commission approved the ERDS in March 1985, and directed the staff to develop the concept further before making ERDS a new regulatory requirement. The subsequent development of ERDS in the 1985 through 1988 timeframe was discussed in SECY-89-193. Detailed surveys of existing hardware and software, conducted at 59 sites (92 units) during 1985 and 1986, indicated that the necessary parameters existed to a sufficient extent on licensees' computer systems for the ERDS concept to be effective. In 1987, the staff conducted successful data transmission under a prototype testing program with Duke Power and Commonwealth Edison reactor units. And in 1988, the staff retained a contractor to design, procure, and install a computer system at the NRC Operations Center (NRCOC) that would be compatible with the systems at various sites.

Contact:
M. L. Au, PE, RES
301-492-3749

In July 1989, the Commission approved the ERDS voluntary participation program as well as initiation of rulemaking for ERDS. The staff issued a Generic Letter 89-15 on August 15,1989 that encouraged industry participation among those utilities that had not already volunteered in the ERDS program, and forwarded the proposed ERDS rule to the Commission for approval (SECY-90-256). In a Staff Requirements Memorandum (SRM) dated August 29, 1990 (Enclosure 1), the Commission approved issuing the proposed rule for public comment. On October 9, 1990, a Notice of Proposed Rulemaking (NPRM) (55 FR 41095) was published for public comment. The public comment period expired December 24, 1990.

Currently, 27 licensees representing 67 power reactor units have agreed to participate in the ERDS program on a voluntary basis. Of these units, eleven are capable of transmitting ERDS data to the NRCOC. Without the rule, NRC would continue its efforts to achieve voluntary implementation of ERDS at the remaining power reactor units. However, given that no additional interest in the voluntary program has been identified since October 1990, the staff does not expect meaningful improvements in the participation rate to occur.

Discussion: The objective of the final rule is timely and effective implementation of ERDS so as to provide increased assurance that a reliable and effective communication system, that will allow the NRC to monitor available critical parameters during an emergency, is in place at all operating power reactors, except Big Rock Point and those that are permanently or indefinitely shut down. Implementation of ERDS is to be accomplished no later than 18 months after the effective date of the final rule.

In response to the NPRM, the NRC received comments from 31 respondents: 2 from interested individuals, 1 from a citizens group, 1 from a former Senior Reactor Operator and Emergency Director at a utility, 1 from the Nuclear Management and Resources Council (NUMARC), 1 from the Nuclear Utility Backfitting and Reform Group (NUBARG), 20 from reactor licensees, 1 from a non-power reactor licensee, and 4 from State authorities. Many of the letters contained comments that were similar in nature. A number of comments were grouped together when appropriate, and so addressed. (Enclosure 2).

The most significant comments were that ERDS would not substantially increase the overall protection of the public health and safety, and that implementing the ERDS would increase the operator's labor burden because industry personnel will have the added burden of having to interface with NRC as well as State or local government agencies receiving the ERDS data, some of which will be staffed by personnel that lack sufficient system specific knowledge to understand the data.

In the regulatory analysis, made available upon publication of the proposed rule, the staff argued that a substantial increase in public health and safety will be achieved. Although the degree to which this rule will provide substantial additional protection is subject to differing judgement, the staff believes that given the nature and importance of NRC's responsibilities in the management of emergency and protective actions, and the improvement in the staff's ability to implement these responsibilities, that substanial additional protection will result, and is fully consistent with the estimated costs. This was based on our view that implementation of the ERDS would improve the reliability and timeliness of data transmission and help ensure that any reactor unit in distress would be suitably monitored. Further, the availability of ERDS should enable the licensee to better use its time and resources to effectively and efficiently deal with the emergency at hand. It remains the conclusion of the staff that the combination of better and more timely assessments of licensee actions by the NRC and the focusing of licensee resources to better deal with the emergency will reduce the overall risk to the public health and safety.

Regarding the concern that implementing ERDS would increase the operator's labor burden, the NRC believes the availability of near real time data depicting the plant conditions would enable it to be more fully aware of the situation while requiring less voice contact with the plant operating staff, thus reducing -- not increasing -- the labor burden of the operators.

Revisions have been made to the proposed rule as a result of these as well as other public comments, but they are mainly editorial and clarifying in nature. Having considered all of the public comments the staff recommends that a final rule be promulgated to implement ERDS.

Coordination: The Office of General Counsel has no legal objection.

Recommendations: That the Commission:

1. Approve the amendments to 10 CFR Part 50 for publication in the Federal Register (Enclosure 3).

2. Certify that this rule, if promulgated, will not have a significant economic impact on a substantial number of small entities, in order to satisfy the

requirements of the Regulatory Flexibility Act, U.S.C. 605(b).

Note that:

a. NUREG-1394, "Emergency Response Data System (ERDS) Implementation", will be issued as a final document in conjunction with these amendments (Enclosure 4).

b. The regulatory analysis will be placed in the NRC Public Docket Room (Enclosure 5).

c. The Commission finds that no significant environmental impact is expected as a result of this action, thus, no environmental impact statement need be prepared. (Statement contained in Enclosure 3, pg. 21)

d. The Chief Counsel for Advocacy of the Small Business Administration will be informed of the Certification and the reason for it as required by the Regulatory Flexibility Act.

e. Congress will be informed of the Commission's action by letter. (Enclosure 6)

f. The final rule contains information requirements that are subject to review by the Office of Management and Budget (OMB). The information requirements have been approved under OMB clearance number 3150-0011.

g. The Office of Public Affairs has prepared a public announcement for release when the final rule is published in the Federal Register. (Enclosure 7)

h. Copies of the Federal Register notice will be distributed to affected licensees, commenters, and other interested parties.

James M. Taylor
Executive Director
for Operations

Enclosures:
1. SRM - August 29, 1990
2. Summary of Public Comments
3. Federal Register notice
4. NUREG-1394
5. Regulatory Analysis
6. Draft Congressional letter
7. Draft Public Announcement

Enclosure 2

Summary of Public Comments

LIST OF COMMENTERS

1. Diane M. Smith
2. Dean Baker
3. R. H. Lagdon, Jr.
4. South Carolina Electric & Gas Company
5. Maryland Department of the Environment
6. Wisconsin Electric Power Company
7. University of Missouri--Rolla
8. Maine Department of Human Services
9. NUMARC
10. Wisconsin Public Service
11. Philadelphia Electric Company
12. Florida Power and Light Company
13. Nuclear Utility Backfitting and Reform Group
14. Alabama Power Company
15. Georgia Power Company
16. Ohio Citizens for Responsible Energy
17. Florida Power Corporation
18. Indiana Michigan Power Company
19. Carolina Power & Light Company
20. Yankee Atomic Electric Company
21. Illinois Department of Nucear Safety
22. Washington Public Power Supply System
23. New Jersey Department of Environmental Protection
24. Boston Edison
25. Baltimore Gas and Electric
26. Centerior Energy (Toledo Edison)
27. Tennessee Valley Authority
28. Duquense Light Company
29. Northeast Utilities
30. Arizona Public Service Company
31. Virginia Electric and Power Company

CATEGORIES OF COMMENTS

1. Computer Security
2. Inadequate Justification/Backfit
3. Alternate Systems & Mods
4. Volunteers Implementation
5. Operator Burden/Interference
6. State Require Licensees to Pay for ERDS
7. NRC Lack of Trust
8. Time on ERDS Header
9. Exclude Non-Power Reactors
10. Relieve Data Sheet Requirement
11. NRC Should Provide ERDS Software
12. Configuration Control
13. Availability of ERDS Data
14. Data Update Frequency
15. Location of ERDS/Timing of Actuation
16. ERDS Implementation Period (18 mo.)
17. Licensee Hardware & Software Requirements
18. ERDS Testing/Test Frequency
19. Isolation Requirements
20. Quality Tags
21. ENS Manning

Note: The Category numerical designators correspond to those used in the ERDS rule, Analysis of Public Comments section.

CONCERNS / COMMENTER	1. Computer Security	2. Inadequate Justification/Backfit	3. Alternate Systems & Mods	4. Volunteers Implementation Plan	5. Operator Burden/Interference	6. State Require Licensees to Pay for ERDS	7. NRC Lack of Trust	8. Time on ERDS Header	9. Exclude Non-Power Reactors	10. Relieve Data Sheet Requirement	11. NRC Should Provide ERDS Software	12. Configuration Control	13. Availability of ERDS Data	14. Data Update Frequency	15. Location of ERDS/Timing of Actuation	16. ERDS Implementation Period (18 mo.)	17. Licensee Hardware & Software Requirements	18. ERDS Testing/Test Frequency	19. Isolation Requirements	20. Quality Tags	21. ENS Manning	Endorsed NUMARC, No Other Comments	Supported NUMARC Position	For ERDS	Against ERDS	Against Rule; For Voluntary Program
1. Diane M. Smith	●																							●		
2. Dean Baker		●	●																						●	
3. R. H. Lagdon, Jr.		●			●	●	●														●					●
4. South Carolina Electric & Gas Company					●	●						●									●					
5. Maryland Department of the Environment													●											●		
6. Wisconsin Electric Power Company								●				●	●	●	●			●								
7. University of Missouri--Rolla									●																	
8. Maine Department of Human Services													●											●		
9. NUMARC		●	●	●	●					●		●		●	●	●	●		●	●	●					●
10. Wisconsin Public Service			●	●	●										●											●
11. Philadelphia Electric Company			●								●	●	●		●			●					●			●
12. Florida Power and Light Company		●			●																●		●		●	
13. Nuclear Utility Backfitting and Reform Group		●			●																		●			●
14. Alabama Power Company					●																		●			●
15. Georgia Power Company					●																		●			●
16. Ohio Citizens for Responsible Energy			●																					●		
17. Florida Power Corporation		●			●											●									●	
18. Indiana Michigan Power Company					●										●	●									●	
19. Carolina Power & Light Company																						●	●			●
20. Yankee Atomic Electric Company		●			●																				●	
21. Illinois Department of Nucear Safety																								●		
22. Washington Public Power Supply System												●											●			●
23. New Jersey Department of Environmental Protection																								●		
24. Boston Edison															●											
25. Baltimore Gas and Electric		●			●							●			●	●		●	●						●	
26. Centerior Energy (Toledo Edison)																						●	●			●
27. Tennessee Valley Authority					●																				●	
28. Duquense Light Company					●										●				●							
29. Northeast Utilities		●	●		●							●		●	●		●	●	●	●	●		●			●
30. Arizona Public Services Company																						●	●			●
31. Virginia Electric and Power Company			●		●													●						●		
TOTAL	1	9	7	2	16	2	1	1	1	1	1	7	4	3	9	4	2	5	4	2	5	3	10	7	7	12

Enclosure 3

Federal Register Notice of Final Rule

PREDECISIONAL

[7590-01]

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150 - AD32

Emergency Response Data System

AGENCY: Nuclear Regulatory Commission

ACTION: Final rule.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is amending its regulations to require licensees of operating nuclear power facilities to participate in the Emergency Response Data System (ERDS) program. This action is needed to ensure that the NRC receives timely and accurate data on a limited set of parameters whose values indicate the condition of the plant during a declaration of an alert or higher emergency classification. This action will also ensure that all licensees establish a definite schedule for implementation of the ERDS program. This rule applies to all licensed nuclear power reactor facilities, except Big Rock Point and those that are permanently or indefinitely shut down. However, units shut down for maintenance, or authorized for fuel loading only, or low power operations, are required to report under ERDS. Big Rock Point is exempt because configuration of the facility does not make available as transmittable data a sufficient number of parameters for effective participation in the ERDS program.

EFFECTIVE DATE: [Insert a date 30 days following publication in the Federal Register.]

FOR FURTHER INFORMATION CONTACT: M. L. Au, P.E., Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, Washington, DC 20555, telephone: (301) 492-3749.

SUPPLEMENTARY INFORMATION:

Background

The Commission published the proposed rule on this subject in the Federal Register on October 9, 1990 [55 FR 41095]. The rule proposed to amend 10 CFR Part 50 to provide for an Emergency Response Data System (ERDS) direct electronic data link between computer data systems used by licensees of operating reactors and the NRC Operations Center (NRCOC) during the declaration of an alert or higher emergency classification. The ERDS supplements the voice transmission of information over the currently installed Emergency Notification System (ENS). The ERDS is activated by a licensee when an alert or higher emergency occurs at a licensed nuclear power facility.

The objective of the final rule is timely and effective implementation of ERDS so as to provide increased assurance that a reliable and effective communication system that will allow the NRC to monitor critical parameters during an emergency is in place at operating power reactors.

Many of the elements of the rule are currently implemented under the

ERDS voluntary program in which over half of the licensed units have volunteered to participate. The ERDS program is not expected to require any advancements in the state of the art, and the configuration of most power reactors is such that the relevant parameter values are available as transmittable data. Therefore, there should be no cause for delay in timely implementation of this rule.

Public Comments

Interested parties were invited to submit comments in connection with the proposed amendment after publication in the Federal Register. There were 113 comments made by 31 commenters to the Notice of Proposed Rulemaking (NPRM): two from interested individuals, one from a citizens' group, one from a former Senior Reactor Operator and Emergency Director at a utility, one from the Nuclear Management and Resources Council (NUMARC), one from the Nuclear Utility Backfitting and Reform Group (NUBARG), 20 from power reactor licensees, one from a non-power reactor licensee, and four from State authorities. Many of the letters contained comments that were similar in nature, hence comments were grouped together when appropriate, and so addressed. The NRC identified 21 separate issues that cover the significant points raised. Public comments received on the proposed rule were docketed and may be examined at the Commission's Public Document Room located at 2120 L Street NW (Lower Level), Washington, DC. Upon consideration of the comments received, the Nuclear Regulatory Commission has adopted the proposed regulations, with certain modifications as set forth below.

Analysis of Public Comments

1. Comment. The ERDS data would be subject to distortion by terrorists or computer hackers which could cause the NRC to respond improperly in their recommendations to the licensee, Federal agencies, State and local governments. If the ERDS were hardened, or essential data elements were verified by voice communication, this potential problem would be eliminated.

Response. It is highly unlikely that a computer hacker would be able to locate ERDS transmissions in the NRC's communications network because of the limited access to this system. Also, the communication protocol incorporated for ERDS transmission would make the data unintelligible without knowledge of the specific site link configuration. Error detection/correction has been incorporated into the transmission protocol which would, in all probability, detect any alteration in the data. And finally, as stated in NUREG-1394, "Emergency Response Data System (ERDS) Implementation", and in the Statement of Consideration of this rulemaking, the NRC will continue the requirement for the licensee to maintain voice communication with the NRC during emergencies -- any data indicating rapid unrealistic changes or unexpected conditions would be immediately suspect and subject to verbal corroboration. Therefore, the NRC does not believe the probability for intentional data distortion is sufficiently large to justify resources for further countermeasures.

2. Comment. There is inadequate justification that implementing the ERDS would substantially increase the overall protection of the public health and

safety. This contention was made by nine commenters in addition to the seven commenters who endorsed the consolidated comments from NUMARC and NUBARG without further elaboration. The commenters stated that if there was a substantial increase this should be quantitatively demonstrable. They also stated that the utility is solely responsible for the protection of the public health. They argued that because this rule does not improve the manner in which the emergency director makes decisions, the claim of "unquantifiable but significant increase" in the protection of the public is invalid. One commenter stated the ERDS is an improvement to a system that has been deemed "adequate," and therefore is not necessary.

Response: It is true that the utility has the primary responsibility for emergency management activities at the site locations. However, the foundation for all NRC emergency response activities is to provide support and coordination to those States and local governments who are responsible for the safety of their citizens as well as provide timely advice to the licensees as needed. To fulfill this mission the NRC must have reliable, necessary and sufficient, and timely information to understand and assess the emergency situations. The ERDS provides such information to NRC.

In the regulatory analysis, made available upon publication of the proposed rule, the staff argued that a substantial increase in public health and safety will be achieved. Although the degree to which this rule will provide substantial additional protection is subject to differing judgement, the staff believes that given the nature and importance of NRC's responsibilities in the management of emergency and protective actions, and the improvement in the staff's ability to implement these responsiblities, that substantial additional protection will result. and is fully consistent with the estimated costs. This was based on our view that implementation of ERDS would improve the reliability and timeliness of data transmission and help ensure that any reactor unit in distress would be suitably monitored. Further, availability of ERDS should enable the licensee to better use its time and resources to effectively and efficiently deal with the emergency at hand. It remains the conclusion of the staff that the combination of better and more timely assessments of licensee actions by the NRC and the focusing of licensee resources to better deal with the emergency will reduce the overall risk to the public health and safety.

3. Comment. One commenter believed that the limited group of reactor parameters monitored through ERDS would be inadequate to provide a sound basis for NRC recommendations and therefore requested modifications to ERDS. One commenter urged the NRC to consider a continuous monitoring system, e.g., the Nuclear Data Link considered by the Commission following the Three Mile Island accident. Other commenters stated that the ERDS design uses cumbersome hardware and software, that NRC's communication hardware should be able to accept data from a multiple unit plant through one modem, and that state-of-the art hardware should be allowed.

Response. Although the ERDS data does not portray every detail of a nuclear power reactor in an emergency situation, in the Commission's judgement it does provide the data required by the NRC to perform its role during an emergency. The ERDS parameter list was selected based on the information the

NRC Technical Teams need to perform their emergency response functions. Moreover, the set of ERDS data will not be the only input to the NRC. The Emergency Notification System (ENS), a voice communication system, will still be available to transmit data and any other relevant information that is not available through ERDS. In combination, the NRC will receive information needed to develop timely and appropriate evaluations of the event and to develop the necessary support actions to ensure protection to public health and safety.

The ERDS is designed to transfer needed reactor data from a nuclear power plant only during emergencies. It is not a system to constantly monitor any licensee. The concept of constant monitoring, such as the Nuclear Data Link, was considered after the Three Mile Island accident in 1978. But after much evaluation and deliberation it failed to receive Congressional approval for funding.

The current protocol is already in use at several reactors under the volunteer program and is in the process of being implemented at other facilities. The NRC does not want to impose additional redesign and retest costs on licensees who have already volunteered for the ERDS program.

The ERDS was designed to use commercially available (off-the-shelf) computers which could effectively handle the data requirements, establishing a single link with each unit. To group several units into a single link would result in a data base size incompatible with the ERDS configuration. The ERDS design has been frozen in order to maintain configuration control and standardization in implementing the ERDS volunteer program.

4. Comment. Submittal of an ERDS implementation plan should not be required of licensees that have implemented ERDS under the voluntary program. Similarly, licensees that have submitted the information required by the voluntary program along with a proposed implementation schedule should also be exempt from the schedule and system requirements contained in paragraph VI.1, VI.2 and VI.4 of the proposed rule.

Response. The staff agrees that it is unnecessary for licensees that have implemented the ERDS in an acceptable manner to submit an implementation plan. The rule has been modified so that licensees who have submitted all information consistent with the timetable set in paragraph 4.b of Appendix E, Section VI, are not required to submit an implementation plan.

5. Comment. (a). Nineteen of the commenters, including three that endorsed the NUMARC comments, were concerned that implementing the ERDS would increase the operators' labor burden because the NRC, as well as State or local government agencies receiving the ERDS data, would not be staffed by personnel with sufficient system specific knowledge to understand the data. This would result in extensive inquiries to the licensees to explain the data, thereby distracting the operating staff from their primary functions of accident response and emergency management.

(b). Some of these commenters urged the NRC to limit the data provided to States and local government and direct them regarding the use of the ERDS information to preclude the improper use or release of the data.

(c). Other commenters stated that with the availability of ERDS parametric reactor data, the NRC would modify its oversight role into one of more active participation in event management, a function, the commenters claimed is solely the responsibility of the licensee.

Response: (a). The NRC Operations Center staff are experienced professionals with extensive knowledge of reactors, sufficient to allow them to use the data provided by the ERDS to follow the course of the emergency, chart and analyze trends, and support appropriate recommendations relating to the health and safety of the public. The NRC believes the availability of near real time data depicting the plant conditions would enable it to be more fully aware of the situation while requiring less voice contact with the plant operating staff, thus reducing -- not increasing -- the labor burden of the operators. Further, the NRC is aware that while not all States have the technical knowledge required to interpret raw ERDS data, some have developed significant expertise in responding to emergencies at nuclear power plants. The NRC believes that since the States are responsible for protective actions to ensure the health and safety of their citizens, they should have available sufficient data upon which to base decisions.

(b). The ERDS link will be established with a State government through a Memorandum of Understanding (MOU) with the NRC. The proper use, control, and dissemination of the ERDS data is one of the subjects addressed by the MOU. Under the MOU, the NRC will provide a liaison to the State at the NRCOC for

ERDS data interpretation if such help is requested.

(c). The implementation of ERDS will not alter the respective responsibilities of the utilities and the NRC with respect to emergency management. The utility will retain primary responsibility for emergency management activities at the site locations. The NRC's role remains one of support and coordination to those States and local governments who are responsible for the safety of their citizens, as well as to provide timely advice to the licensees as needed.

6. Comment. States may require the licensee to pay for equipment required to receive and process the ERDS data. Furthermore, providing ERDS data to the States and local governments would increase NRC costs beyond that estimated in the Backfit Analysis.

Response. The NRC has no control or authority over the State governments regarding their funding of ERDS receiving equipment. Each individual State government should determine its equipment and data requirements. However, through a Memorandum of Understanding (MOU) between the State and the NRC regarding the ERDS link, the ERDS data can be made available to a State. One of the functions of the NRC is to provide appropriate support to the States during a nuclear power plant emergency. This responsibility exists independent of the ERDS, and in the staff's view, the ERDS interface between the NRC and the States should not result in additional costs to the NRC.

7. Comment. Implementing the ERDS seems to imply some general concern that the NRC neither trusts its abilities nor those of the licensees' to respond correctly to emergencies using current practices.

Response. ERDS is an enhancement of existing procedures that provides a superior method of assembling and transmitting to the NRC near real time data from a licensee during an alert or higher emergency classification. Accurate and timely data assists the NRC in conducting informed analyses of the plant condition, and facilitates NRC consultation with State or local governments regarding action to ensure protection of public health and safety.

8. Comment. Will the time in the header of the ERDS data packet be some standard time such as GMT, EST, etc.?

Response. The time from the licensee's plant computer will be used with ERDS data. Included in each licensee's ERDS implementation plan will be the time standards used in their computers. This practice will ersure that the particular licensee and all monitors of ERDS data relating to a particular emergency or test are using the same time. There is no requirement for all licensees to adhere to a common standard time.

9. Comment. Non-power reactors should be explicitly exempt from the ERDS requirements.

Response. Since section 10 CFR 50.72 of the regulations applies only to nuclear power reactors, it is not necessary to explicitly exempt non-power reactors in the rule.

10. Comment. Licensees are requested by Generic Letter 89-89 to transmit a significant number of data sheets to the NRC during emergencies. With the implementation of ERDS this should be relieved to allow better use of licensee resources to support ERDS.

Response. The information cited is an Information Notice (IN), and as such, it requires no action on the part of the licensee. The form contained in IN 89-89 is a copy of the work sheet used by NRC Headquarters Operations Center officers in recording routine Event Reports over the ENS. It was provided as information to licensees to aid in structuring their normal event report.

11. Comment. The NRC should provide the software required for ERDS communications to the utilities.

Response. Currently the NRC is evaluating the possibility of providing ERDS software to the utilities.

12. Comment. There were several concerns regarding the configuration control of ERDS hardware and software. Five commenters stated the requirement to notify the NRC within 30 days following changes in individual parameters is overly prescriptive, and they proposed extending the maximum allowable notification period to 90 days, annually, or during Final Safety Analysis Report (FSAR) updates. Two commenters believed the time estimated to perform

the configuration control functions was low by a factor of two or three, and therefore the ERDS would be more costly to the utilities than estimated. One commenter stated there should be specific guidance provided for the configuration control requirements of the utility/ERDS interface; and two were concerned that if the NRC changes its format the licensees are automatically required to change their transmission of data. They recommended that the data should be limited to an initial format with no later changes.

Response. In establishing the current reporting requirement for changes in the ERDS Data Point Library, the staff balanced the time needed by the licensees for its design change control and review processes against the staff's need to know based on safety considerations. The staff views the 30 days as reasonable for the licensees to prepare such a report, and given that such changes can influence the NRC's interpretation of ERDS data does not view any further delay as warranted.

For some licensees, plant to plant variation could result in a greater labor burden associated with configuration control tasks than the 5 person days per reactor year used in the regulatory analysis. However, that value represents an average that, considering the entire nuclear power industry, appears substantially correct. There is an economy of scale for those utilities that can combine submissions from multiple reactor units that reduce the industry average.

The basic guidance information for configuration control of the ERDS is contained in NUREG-1394. Based on the experience of the utilities that have implemented ERDS voluntarily, the configuration control requirements appear to be appropriate.

The proposed rule would require the licensee to change its data transmission if the NRC changes its format, and the staff agrees that this is an unreasonable requirement on the licensees. Therefore the final rule has been revised to require all data transmission to conform to the initial format. As the ERDS matures, or as technical advances increase capabilities, there may be some modifications. However, any such changes will be coordinated with the licensees.

13. Comment: The ERDS rulemaking should clearly state that the ERDS is available to the States; and that all future State and local government requests for on-line data should be made through the NRC. Furthermore, the licensees should have access to the same screens as those available to the NRC.

Response: It is not within the authority of the Commission to specify to the States what data they may or may not receive. However, the NRC does recommend that States desiring an emergency data link to nuclear power plants within their jurisdiction use an ERDS connection from the NRC Operations Center for that purpose. A Memorandum of Understanding with the NRC will provide the State with ERDS data. A provision allowing States to receive ERDS data should not be part of the rule since there is no NRC requirement imposed upon licensees to establish a data link with a State. The concept of providing each licensee with the same work stations as the NRC was considered.

However, it was deemed not cost beneficial to expend in excess of $900,000 for the sole purpose of sending back to the licensee that data which they originally sent to the NRC. Any licensee desiring to do so may establish their own work station based on NRC design.

14. Comments. The requirement for the reactor parametric data to be transmitted to the NRC Operations Center at time intervals of not less than 15 seconds or more than 60 seconds is too prescriptive and may eliminate the use of some existing computer systems currently supporting the licensee's Technical Support Center (TSC)/Emergency Operating Facility (EOF), etc. One commenter suggested that data update frequency should be plant specific. Others argued that the wording in the proposed rule puts the licensee in jeopardy of non-compliance in the event of system or telecommunications line failure, and that considering the conditions, the proper descriptor for the data is "near real time" instead of "real time."

Response. Originally the desired update frequency for ERDS data was 15 seconds, but to minimize the impact on central processing unit (CPU) use, the minimum frequency was reduced by a factor of four, i.e., to at least every 60 seconds. Based upon the experience of those manning the NRCOC, the staff believes that less frequent data collection would diminish the NRC monitors' ability to adequately follow the course of the emergency. Furthermore, allowing update frequencies to range between 15 seconds and 60 seconds should provide sufficient latitude to allow the use of most licensees' existing computer systems. Exceptions to this requirement will be considered on a case by case basis.

It is highly improbable a licensee would be cited or fined for violations resulting from ERDS equipment failure. Nonetheless, in the wording of the final rule, the term "near real time" has been used to describe the ERDS data.

15. Comment. The requirement to activate the ERDS at the time the NRC is notified of the declaration of an alert or higher emergency classification should be relaxed because it places a heavy labor burden on the plant operators at this critical time. Several commenters suggested a delay of one hour in order to allow actuation from the Technical Support Center, thus removing the burden from control room personnel. Four commenters stated the ERDS should not be operated from an on-site computer, and two suggested the rule should allow the ERDS to be activated by computer operations personnel or a software switch. One commenter stated the licensee should be the only entity to activate or deactivate the ERDS for a given plant.

Response. There is no requirement for the ERDS to be activated from the control room or by control room personnel. The use of computer operations personnel or a software switch is acceptable to activate the ERDS. The only requirement is to initiate ERDS data transmission as soon as possible but not later than one hour after declaring an emergency class of alert, site area emergency, or general emergency. This change is reflected in the final rule. The specific methods selected to achieve this should be fully described in each licensee's ERDS implementation plan. The notification requirement is valid in order for NRC to fulfill its mandated role to monitor the licensee during an emergency. A delay of one hour or more could deprive the NRC of vital information necessary to perform its advisory and monitoring role. The licensee is currently required in Section 10 CFR 50.72 to have a shift communicator maintain continuous contact with the NRC Operations Center. This request in not being changed, and this person could be responsible for initiating the ERDS link.

Similarly, the requirement to use an on-site computer does not mean this equipment must be located in the control room. Any on-site location, such as the Technical Support Center or a computer facility, which is capable of meeting the requirement for notification is an acceptable location. However, off-site computers, e.g., at some central location used to service more than one plant site could be prone to additional commercial link vulnerability. This could potentially decrease the ERDS availability and reliability beyond acceptable limits.

The ERDS link will be activated or deactivated by the licensee to transmit the ERDS data to the NRC Operations Center via the NRC provided telephone lines. In the event that NRC perceives the need to disconnect a plant from the NRC Operations Center to allow another plant onto the system, for example, terminating the transmission of exercise data to allow a unit with a real emergency to access the system, this capability must be available to the NRC.

16. Comment. The 18 month ERDS implementation schedule does not provide adequate flexibility for all utilities to install the system. Adhering to that schedule will cause serious operational and cost impacts to some utilities due to the extensive hardware modifications required.

Response. The voluntary program demonstrated that an implementation period of 18 months is generally adequate. However, the NRC realizes there are plant to plant variations which, in certain cases, may require more extensive and time consuming modifications. Utilities experiencing exceptional difficulties in meeting the 18 month implementation schedule should request extension from the NRC. Such requests will be reviewed on a case-by-case basis.

17. Comment. The requirement in the proposed rule, Appendix E, Section VI.2, should be clarified to indicate that the licensee will provide data from each unit via an output port on the appropriate data system and necessary software to assemble the data to be transmitted.

Response. The staff agrees with this clarification. This section of the final rule will be modified appropriately.

18. Comment. Quarterly testing of the ERDS is too frequent. Testing on a semi-annual or periodic, but unspecified schedule should be sufficient. One commenter noted that the rule does not address reporting requirements for system failures during testing. Also for consistency between the discussion section and the rule, the following statement regarding ERDS use during emergency training exercises should be added to 10 CFR 50.72(a)(4) of the rule. Although there is no requirement, the ERDS may also be activated by the licensee during emergency drills or exercises if the licensee's computer system has the capability to transmit the data.

Response. Quarterly testing during the initial year or eighteen months of the ERDS program is necessary for both the licensees and the NRC monitors to gain experience and confidence with the system, as well as prove the availability and reliability of the system. An established schedule allows both the NRC and licensees to plan and allocate time and resources for testing rather than trying to accommodate testing on an unregimented basis. After a period of approximately one year of demonstrated system performance, i.e., proper functioning during quarterly testing, the test frequency may be relaxed to semi-annually.

There are no explicit reporting requirements for failures during testing because the quarterly testing will be conducted with NRC. If there are failures during these tests, the NRC, because of its participation in the tests, will be aware of them. It is unlikely there will be any system testing of which the NRC is unaware, e.g., with State or local governments, since the State links will most probably be through the NRC Operations Center. The recommended additional statement regarding use of ERDS during emergency training exercises has been included in the final rule.

19. Comment. Three commenters stated that this rule should impose no new isolation requirements, and suggested that references to a potential requirement for additional isolation requirements should be deleted.

Response. The reference to the potential need for isolation devices is not a new requirement. It is intended merely to serve to reinforce requirements as a design control mechanism in 10 CFR 50.55a and adds emphasis for adequate protection against spurious electrical signals. More recently constructed nuclear power reactors have adequate isolation of their computer interfaces, but in some older reactors it is conceivable the computer assembling the ERDS data may not be fully buffered, and as such, could require appropriate isolation devices. The statement alerts the licensees to the potential need for additional isolation devices.

20. Comments. There should be more flexibility in acceptable quality indicators (tags) for the ERDS data, thus allowing greater use of existing plant methodologies. Requiring the utilities to use the quality tags prescribed by the NRC would force major software changes and added costs for some licensees.

Response. Using the data quality indicators prescribed by the NRC should necessitate, at the most, only very minor licensee software changes. A simple translation matrix which converts the quality tags used by the licensee to the form to be used by the NRC Operations Center is sufficient. This can be applied to the ERDS data prior to transmission.

There is no requirement for the utilities to change the quality tags used at their facility. However, if each utility transmits ERDS data to the NRC Operations Center using their own quality tags, variation from licensee to licensee could cause confusion to the NRC monitors, thereby necessitating additional telephonic consultation with the licensee.

21. Comment: Four commenters stated that when ERDS is implemented the

requirement for full time manning of the Emergency Notification System (ENS) should be relaxed. Without this relaxation the affected utility will not be able to redirect its efforts as claimed.

<u>Response</u>: It is not the intent to replace the ENS with ERDS; rather, the ERDS is a supplemental system specialized in automatic collection and transmission in near real time of a selected set of parametric reactor data required by the NRC in its emergency monitoring role. Although implementing the ERDS will diminish the current ENS burden, not all functions of the ENS will be subsumed into the ERDS. Therefore, telephone contact will still be required via the ENS. Nevertheless, the effort required by the licensee's personnel to gather the data for periodic relay to the NRC will be greatly reduced, thus permitting their use of personnel in other emergency functions.

Environmental Impact: Categorical Exclusion

The NRC has determined that this proposed regulation is the type of action described in categorical exclusion 10 CFR 51.22(c)(3)(iii). Therefore, neither an environmental impact statement nor an environmental assessment has been prepared for this proposed regulation.

Paperwork Reduction Act Statement

This proposed rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). This rule has been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

The regulatory analysis estimates an annual per reactor level of effort of 5 days for licensee staff and 3 days for NRC staff for the maintenance of the on-site ERDS configuration control program. An integral part of this activity is the preparation of configuration control reports by the licensee and their review by the NRC. This paperwork effort is estimated at less than one-third the overall configuration control level of effort. Thus, the reporting burden per reactor is estimated at less than 2 days per year, and the NRC's review effort is estimated at less than 1 day per reactor year. Send comments regarding this burden estimate or any aspect of this collection of information, including suggestions for reducing the burden, to the Information and Records Management Branch (MNBB-7714), U.S. Nuclear Regulatory Commission, Washington, DC 20555 and to the Paperwork Reduction Project (3150-0011), Office of Information and Regulatory Affairs (NEOB-3019), Office of Management and Budget, Washington, DC 20503.

Regulatory Analysis

The NRC has prepared a preliminary regulatory analysis for the proposed rulemaking on this subject. The analysis examined the costs and benefits of the alternatives considered by the NRC. The NRC requested public comments on the preliminary regulatory analysis. Comments received were considered, but no changes to the regulatory analysis are considered necessary, so a separate regulatory analysis has not been prepared for the final rule.

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR 121.

Backfit Analysis

As required by 10 CFR 50.109, the Commission has completed a backfit analysis for this rule. The Commission concluded that the rule will provide a substantial increase in the overall protection of the public health and safety by ensuring far more accurate and timely flow of data for the NRC to fulfill its role during an alert or higher emergency. The direct and indirect costs estimated for the implementation of this rule are justified in view of this increased protection. Further, the implementation and maintenance requirements of the rule will have no effect on occupational radiological exposure. The backfit analysis on which this determination is based is as follows:

Item 1: Statement of the specific objective that the backfit is designed to achieve.

Response: The objective of the ERDS rulemaking effort is to achieve a high degree of assurance that accurate near real-time data is made available to the NRC to evaluate critical parameters at any operating reactor facility during an alert or higher emergency. This in turn would improve the NRC's understanding of an event and allow the NRC to perform its role more effectively and efficiently which includes: (i) monitoring the licensee to ensure that appropriate recommendations are being made with respect to offsite protective actions; (ii) providing the licensee with technical analysis and logistic support; (iii) supporting offsite authorities; (iv) keeping other Federal agencies and entities informed of the status of the incident; and (v) keeping the media informed of the NRC's knowledge of the status of the incident.

In addition, the implementation of the ERDS would enable the licensee to better use its time and resources to effectively and efficiently deal with the emergency. The combination of better and more timely assessments of licensee actions by the NRC and the focusing of the licensee's resources to better deal with the emergency at hand together will reduce the overall risk to the public health and safety from an emergency.

Item 2: General description of the activity that would be required of the licensee or applicant in order to complete the backfit.

Response: All licensees or applicants would be required to install an NRC-supplied communication link, provide the software necessary to format available selected critical plant condition data for NRC use, provide the necessary hardware from the in-plant computer to interface with the NRC-supplied communication link, provide support for periodic testing of the ERDS, and report any configuration changes to the licensee's ERDS-related hardware and software. Initially, the ERDS will be tested quarterly, unless otherwise determined by NRC based on demonstrated system performance.

Item 3: Potential change in the risk to the public from the accidental offsite release of radioactive material.

Response: The implementation of the ERDS in all operating nuclear power reactors would provide the NRC with more accurate and timely data to fulfill its major role during an alert or higher emergency. The major role, as defined in the 1987 revision to NUREG-0728, is to monitor the licensee to ensure that appropriate recommendations are being made by the licensee with respect to offsite protective actions. Currently, the NRC relies on data verbally transmitted through the Emergency Notification System (ENS) during an emergency. Although deemed adequate, this method of transmission has, on occasion, proven to be unreliable. In addition, data collection is time consuming since various instruments are read and their indications logged on a periodic basis for verbal communication via ENS. The implementation of the ERDS would improve the reliability and timeliness of data transmission and help ensure that any reactor unit in distress can be suitably monitored. Therefore, the NRC would be able to make better and more timely assessments of the licensee's actions regarding management of both emergency and protective actions. Although licensees will be required to maintain voice communication via the Emergency Notification System (ENS), the licensee resources that now are required to collect and relay data and information to the NRC will be available to deal with the emergency. The combination of better and more timely assessments of licensee actions by the NRC, and the focusing of licensee resources to better deal with the emergency at hand together will reduce the overall risk to the public health and safety from an emergency.

Item 4: Potential impact on radiological exposure of facility employees.

Response: The implementation of the proposed ERDS rule would have no effect on routine occupational radiological exposure and would not result in increased radiological exposure of facility employees.

Item 5: Installation and continuing costs associated with the backfit, including the cost of facility downtime or the cost of construction delay.

Response: The cost impact of the rule was estimated to be approximately \$153,000 for one nuclear power reactor (one unit). This figure, expressed in 1990 dollars, represents the incremental worth of installing and operating ERDS for 30 years using a 5 percent discount rate. The overall industry cost of implementing the rule for 118 nuclear power reactor units was estimated at approximately \$18 million. No downtime costs were considered in the cost impact estimates because the installation and operation of the ERDS should have no impact on the operation of a nuclear power plant.

Item 6: The potential safety impact of changes in plant or operational complexity, including the relationship to proposed and existing regulatory requirements.

Response: The ERDS rule should have little or no impact on the operational complexity of the nuclear power reactor units since the required modifications to the hardware and software are minor. The redirection in the labor burden provided by the automatic collection and transmission of selected reactor data would increase the efficiency and effectiveness of nuclear power plant operating personnel during an emergency. This rule is closely associated with Generic Letter 89-15 and complements the ENS that exists at every nuclear power reactor.

Item 7: The estimated resource burden on the NRC associated with the backfit and availability of such resources.

Response: The impact on the NRC resulting from the implementation of the ERDS rule is anticipated to be a one-time cost of about $200,000 for the current population of operational/licensed nuclear reactor units. This figure provides for initial reviews of licensees' implementation plan submittals. After implementation, the NRC cost is estimated to be approximately $4.3 million for 118 nuclear power reactor units. This figure represents the costs for periodic testing and configuration control expressed as the present worth in 1990 dollars and uses a 5 percent discount rate over 30 years.

Item 8: The potential impact of the differences in facility type, design, or age on the relevancy and practicality of the backfit.

Response: The rule is independent of the facility's type, design, or age. There are considerable variations in the instrumentation systems of the nuclear power plants, and the estimated cost impacts were based on an average value for current nuclear power plants to implement the ERDS. There will be no differences, however, in potential impacts between the various facilities on a yearly basis. The rule does not require that licensees monitor more parameters than are presently monitored at each facility.

Item 9: Whether the proposed backfit is interim or final and, if interim, the justification for imposing the proposed backfit on an interim basis.

Response: Implementation of the ERDS in accordance with the final rule will require that all licensees develop and submit an ERDS implementation plan to the NRC within 75 days of the publication of the final rule in the Federal Register. The implementation plan should provide a schedule which identifies the earliest possible time frame for ERDS implementation by the licensee as well as proposed alternate implementation dates. The NRC will establish an industry wide ERDS implementation schedule which will take into account suc' factors as planned computer modifications and scheduled outages. The ERC. must be implemented within 18 months of the publication of the final rule in the Federal Register. Licensees that have submitted the required information under the voluntary implementation program will not be required to resubmit this information. However, they will be required to meet the implementation schedule of eighteen months after the effective date of final rule or before initial escalation to full power :hever comes later. Licensees with currently operational ERDS interfaces approved under the voluntary ERDS implementation program will not be required to submit another implementation plan and will be considered to have met the requirements under this rule.

List of Subjects in 10 CFR Part 50

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

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 553, the NRC is proposing to adopt the following amendment to 10 CFR Part 50.

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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

AUTHORITY: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246, (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235), sec. 102, Pub. L. 91-190, 83

Stat. 853 (42 U.S.C. 4332). Sections 50.13, and 50.54(dd), also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a, and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844) Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 112, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80 through 50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Section 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Appendix F also issued under sec. 187, 68 Stat. 955 (42 U.S.C. 2237).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§ 50.46(a) and (b), and 50.54(c) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 50.7(a), 50.10(a)-(c), 50.34(a) and (e), 50.44(a)-(c), 50.46(a) and (b), 50.47(b), 50.48(a), (c), (d), and (e), 50.49(a), 50.54(a), (i), (i)(1), (l)-(n), (p), (q), (t), (v), and (y), 50.55(f), 50.55a(a), (c)-(e), (g), and (h), 50.59(c), 50.60(a), 50.62(c), 50.64(b), and 50.80(a) and (b) are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.49(d), (h), and (j), 50.54(w), (z), (bb), (cc), and (dd), 50.55(e), 50.59(b), 50.61(b), 50.62(b), 50.70(a), 50.71(a)-(c) and (e), 50.72(a), 50.73(a) and (b), 50.74, 50.78, and 50.90 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

2. In § 50.72, paragraph (a)(4) is redesignated as paragraph (a)(5) and a new paragraph (a)(4) is added to read as follows:

§ 50.72 Immediate notification requirements for operating nuclear power reactors.

(a) * * *

(4) The licensee shall activate the Emergency Response Data System (ERDS)[5] as soon as possible but not later than one hour after declaring an emergency class of alert, site area emergency, or general emergency. The ERDS may be also be activated by the licensee during emergency drills or exercises if the licensee's computer system has the capability to transmit the exercise data.

* * * * *

[5] Requirements for ERDS are addressed in Appendix E, Section VI.

3. Appendix E is amended by adding a new Section VI, Emergency Response Data System, to read as follows:

Appendix E - Emergency Planning and Preparedness for Production and Utilization Facilities

* * * * *

VI. Emergency Response Data System

1. The Emergency Response Data System (ERDS) is a direct near real-time electronic data link between the licensee's onsite computer system and the NRC Operations Center that provides for the automated transmission of a limited data set of selected parameters. The ERDS supplements the existing voice transmission over the Emergency Notification System (ENS) by providing the NRC Operations Center with timely and accurate updates of a limited set of parameters from the licensee's installed onsite computer system in the event of an emergency. When selected plant data are not available on the licensee's onsite computer system, retrofitting of data points is not required. The licensee shall test the ERDS periodically to verify system availability and operability. The frequency of ERDS testing will be quarterly unless otherwise set by NRC based on demonstrated system performance.

2. Except for Big Rock Point and all nuclear power facilities that are shut down permanently or indefinitely, onsite hardware and software shall be provided at each unit by the licensee to interface with the NRC receiving

system. The licensee shall provide necessary software to assemble the data to be transmitted and transmit data from each unit via an output port on the appropriate data system. The hardware and software must have the following characteristics:

a. Data points, if resident in the in-plant computer systems, must be transmitted for four selected types of plant conditions: reactor core and coolant system conditions; reactor containment conditions; radioactivity release rates; and plant meteorological tower data. A separate data feed is required for each reactor unit. While it is recognized that ERDS is not a safety system, it is conceivable that a licensee's ERDS interface could communicate with a safety system. In this case, appropriate isolation devices would be required at these interfaces.[6] The data points, identified in the following parameters will be transmitted:

(i) For pressurized water reactors (PWRs), the selected plant parameters are: (1) Primary coolant system: pressure, temperatures (hot leg, cold leg, and core exit thermocouples), subcooling margin, pressurizer level, reactor coolant charging/makeup flow, reactor vessel level, reactor coolant flow, and reactor power; (2) Secondary coolant system: steam generator levels and pressures, main feedwater flows, and auxiliary and emergency feedwater flows; (3) Safety injection: high- and low-pressure safety injection flows, safety injection flows (Westinghouse), and borated water storage tank level; (4) Containment: pressure, temperatures, hydrogen concentration, and sump levels; (5) Radiation monitoring system: reactor coolant radioactivity, containment

[6] See 10 CFR 50.55a(h) Protection Systems.

radiation level, condenser air removal radiation level, effluent radiation monitors, and process radiation monitor levels; and (6) Meteorological data: wind speed, wind direction, and atmospheric stability.

(ii) For boiling water reactors (BWRs), the selected parameters are: (1) Reactor coolant system: reactor pressure, reactor vessel level, feedwater flow, and reactor power; (2) Safety injection: reactor core isolation cooling flow, high-pressure coolant injection/high-pressure core spray flow, core spray flow, low-pressure coolant injection flow, and condensate storage tank level; (3) Containment: drywell pressure, drywell temperatures, drywell sump levels, hydrogen and oxygen concentrations, suppression pool temperature, and suppression pool level; (4) Radiation monitoring system: reactor coolant radioactivity level, primary containment radiation level, condenser off-gas radiation level, effluent radiation monitor, and process radiation levels; and (5) Meteorological data: wind speed, wind direction, and atmospheric stability.

b. The system must be capable to transmit all available ERDS parameters at time intervals of not less than 15 seconds or more than 60 seconds.

c. All link control and data transmission must be established in a format compatible with the NRC receiving system[7] as configured at the time of licensee implementation.

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[7] Guidance is provided in NUREG-1394.

3. Maintaining Emergency Response Data System

a. Any hardware and software changes that affect the transmitted data points identified in the ERDS Data Point Library[8] (site specific data base residing on the ERDS computer) must be submitted to the NRC within 30 days after the changes are completed.

b. Hardware and software changes, with the exception of data point modifications, that could affect the transmission format and computer communication protocol to the ERDS must be provided to the NRC as soon as practicable and at least 30 days prior to the modification.

c. In the event of a failure of the NRC supplied onsite modem, a replacement unit will be furnished by the NRC for licensee installation.

4. Implementing the Emergency Response Data System Program

a. Each licensee shall develop and submit an ERDS implementation program plan to the NRC by [insert a date 75 days after publication of the final rule]. To ensure compatibility with the guidance provided for the ERDS, the ERDS implementation program plan,[9] must include, but not be limited to, information on the licensee's computer system configuration (i.e., hardware and software), interface, and procedures.

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[8] See NUREG-1394, Appendix C, Data Point Library.

[9] See NUREG-1394, Section 3.

b. Licensees must comply with Appendix E, Section V of this part.

c. Licensees that have submitted the required information under the voluntary ERDS implementation program will not be required to resubmit this information. The licensee shall meet the implementation schedule of Appendix E, Section VI.4d.

d. Each licensee shall complete implementation of the ERDS by [insert a date eighteen months after the effective date of the final rule] or before initial escalation to full power, whichever comes later. Licensees with currently operational ERDS interfaces approved under the voluntary ERDS implementation program[10] will not be required to submit another implementation plan and will be considered to have met the requirements for ERDS under Appendix E, Section VI.1 and 2 of this part.

Dated at Rockville, Maryland, this _____day of _________, 1990.

For the Nuclear Regulatory Commission.

____

Samuel J. Chilk,

Secretary.

[10] See NUREG-1394.

Enclosure 4

NUREG-1394

NUREG-1394

Emergency Response Data System (ERDS) Implementation

U.S. Nuclear Regulatory Commission

Office for Analysis and Evaluation of Operational Data

J. Jolicoeur

9006080154

NUREG-1394

Emergency Response Data System (ERDS) Implementation

Manuscript Completed: April 1990
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Washington, DC 20555

ABSTRACT

The U.S. Nuclear Regulatory Commission has begun implementation of the Emergency Response Data System (ERDS) to upgrade its ability to acquire data from nuclear power plants in the event of an emergency at the plant. ERDS provides a direct real-time transfer of data from licensee plant computers to the NRC Operations Center. The system has been designed to be activated by the licensee during an emergency which has been classified at an ALERT or higher level. The NRC portion of ERDS will receive the data stream, sort and file the data. The users will include the NRC Operations Center, the NRC Regional Office of the affected plant, and if requested the States which are within the ten mile EPZ of the site. The currently installed Emergency Notification System will be used to supplement ERDS data.

This report provides the minimum guidance for implementation of ERDS at licensee sites. It is intended to be used for planning implementation under the current voluntary program as well as for providing the minimum standards for implementing the proposed ERDS rule.

CONTENTS

ACKNOWLEDGMENTS

The author wishes to acknowledge the efforts of Mr. T. LaRosa and Ms. L. Saul of EI International, Inc., the NRC ERDS implementation contractor. A number of the appendices enclosed in this report are the direct result of their efforts in managing the development of the ERDS system.

EMERGENCY RESPONSE DATA SYSTEM (ERDS)

1. Introduction

As a result of the Three Mile Island Unit 2 accident on March 28, 1979, the NRC and others recognized a need to substantially improve the NRC's ability to acquire data on plant conditions during emergencies. Before designing a system to accomplish that task, the NRC first needed to resolve a number of background issues. These issues were: (1) What is the appropriate role for the Commission during an accident? (2) What information is needed by the Commission to support this role? and (3) Are any changes necessary in Commission authority to enhance Commission response to nuclear emergencies?

The Commission has defined the NRC's role in the event of an emergency primarily as one of monitoring the licensee to assure that appropriate recommendations are made with respect to offsite protective actions. Other aspects of the NRC role include supporting the licensee with technical analysis and logistic support, supporting offsite authorities (including confirming the licensee's recommendations to offsite authorities), keeping other Federal agencies and entities informed of the status of the incident, and keeping the media informed on the NRC's knowledge of the status of the incident including coordination with other public affairs groups. This role was studied by the Office of the Executive Legal Director (now Office of the General Counsel) who determined that the NRC's legal authority provides a sufficient basis for the Commission's emergency response role.

To fulfill the NRC's role, the NRC requires accurate timely data on four types of parameters: (1) core and coolant system conditions must be known well enough to assess the extent or likelihood of core damage; (2) conditions inside the containment building must be known well enough to assess the likelihood and consequence of its failure; (3) radioactivity release rates must be available promptly to assess the immediacy and degree of public danger; and (4) the data from the plant's meteorological tower is necessary to assess the likely patterns of potential or actual impact on the public.

Experience with the voice only emergency communications link, currently utilized for data transmission, has demonstrated that excessive amounts of time are needed for the routine transmission of data and for verification or correction of data that appear questionable. Error rates have been excessive; and there have been problems in getting new data and frequent updates. In addition, the current system creates an excessive drain on the time of valuable experts. When errors occur, they can create false issues which, at best, divert experts from the real problems for long periods of time. At worst, incorrect data may cause the NRC to respond to offsite officials with inaccurate or outdated advice that results in inappropriate actions.

2. ERDS Information

2.1 ERDS Design Concept

The system selected to fulfill the data collection needs of the NRC is the Emergency Response Data System (ERDS). The Emergency Response Data System concept is a direct electronic transmission of selected parameters (Figures 1 and 2) from the electronic data systems that are currently installed at licensee facilities.

The ERDS design (Figure 3) utilizes DEC MicroVAX 3600 mini computers as system mainframes. These will be used to receive, sort, and file the incoming data stream. User stations will be PC based stations where the data may be accessed, processed, and displayed. System users will include the NRC Operations Center in Bethesda, MD, the NRC Regional Office, the NRC Technical Training Center, and if requested the States which are within the ten mile EPZ of the site.

The ERDS would be for use only during emergencies and would be activated by the licensees during declared emergencies classified at the ALERT or higher level to begin transmission to the NRC Operations Center. The ERDS would be supplemented with voice transmission of essential data not available on licensee's systems rather than require a modification to the existing system.

2.2 Concept Tests

The concept of electronic data transmission was first tested on July 19, 1984 from the Duke Power Company system at the McGuire facility. The data transfer was accomplished using an electronic mail type arrangement which, although not a real-time system, allowed for electronic data transfer. The data set was limited to a list of 69 specific data points to test the appropriateness of the NRC's parameter list.

A test of data transmission of 60 specific data points was successfully conducted on August 13, 1985 from the Commonwealth Edison system at the LaSalle facility.

A data transmission system was also established for the Zion Federal Field Exercise. The data transmission and receipt methodologies were essentially the same as the test conducted with LaSalle, but several data display techniques for the NRC Operations Center were used. The data set consisted of 65 data points.

The tests of the ERDS concept have demonstrated that there is great value in using electronic data transmission for obtaining a limited set of reliable, time tagged data. The NRC response teams functioned more efficiently and their assessments were more timely. Major improvements in ability to focus on the significant factors and to predict the course of events were noted. The questions that were asked of the licensee were focused on overall status and course of action rather than simple data requests, therefore reducing the volume of communication and increasing the quality of the communication.

2.3 Survey Of Licensee Capabilities

An ERDS Requirements Analysis was conducted in 1986 that included survey visits at 59 plant sites representing 92 reactor units. The focus of the site surveys was to review the design of the data systems on site and availability of the data to be provided to the NRC. The following summarizes the availability of the ERDS parameters for the surveyed facilities:

- The average availability of points for applicable parameters at BWRs is 78.7 percent. No BWRs had 100 percent of the applicable parameters available as transmittable computer points.

- The average availability of points for applicable parameters at PWRs is 92.6 percent. Eleven PWRs had 100 percent availability.

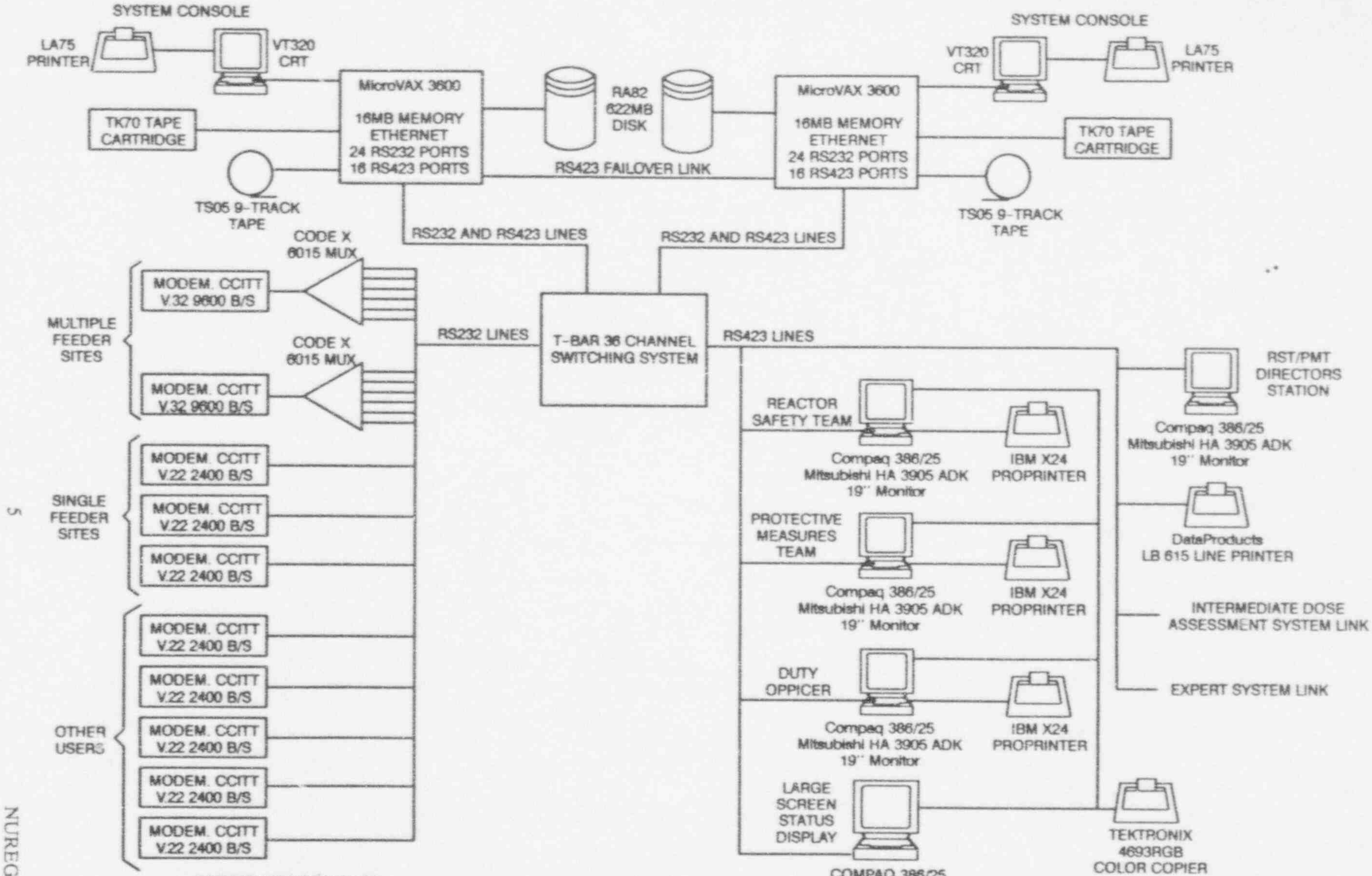
- With regard to the capability of the current hardware environment at the sites to support the generation of a data feed to ERDS, approximately 5 to 10 percent of the licensee systems are running at close to 100 percent processing capability now in the post trip or incident environment and approximately 10 to 15 percent of the licensee systems are hardware limited (e.g., no available output port for an ERDS connection). In many cases however, the licensees with hardware limitations are planning equipment upgrades in the near future for reasons other than supporting ERDS.

Primary Coolant System	Pressure Temperatures—Hot Leg Temperatures—Cold Leg Temperatures—Core Exit Thermocouples Subcooling Margin Pressurizer Level RCS Charging/Makeup Flow Reactor Vessel Level (When Available) Reactor Coolant Flow Reactor Power
Secondary Coolant System	Steam Generator Levels Steam Generator Pressures Main Feedwater Flows Auxiliary/Emergency Feedwater Flows
Safety Injection	High Pressure Safety Injection Flows Low Pressure Safety Injection Flows Safety Injection Flows (Westinghouse) Borated Water Storage Tank Level
Containment	Containment Pressure Containment Temperatures Hydrogen Concentration Containment Sump Levels
Radiation Monitoring System	Reactor Coolant Radioactivity Containment Radiation Level Condenser Air Removal Radiation Level Effluent Radiation Monitors Process Radiation Monitor Levels
Meteorological	Wind Speed Wind Direction Atmospheric Stability

Figure 1. PWR Parameter List

Primary Coolant System	Reactor Pressure Reactor Vessel Level Feedwater Flow Reactor Power
Safety Injection	RCIC Flow HPCI/HPCS Flow Core Spray Flow LPCI Flow Condensate Storage Tank Level
Containment	Drywell Pressure Drywell Temperatures Hydrogen and Oxygen Concentration Drywell Sump Levels Suppression Pool Temperature Suppression Pool Level
Radiation Monitoring System	Reactor Coolant Radioactivity Level Primary Containment Radiation Level Condenser Off-Gas Radiation Level Effluent Radiation Monitor Process Radiation Levels
Meteorological	Wind Speed Wind Direction Atmospheric Stability

Figure 2. BWR Parameter List



REFERENECE DRAWINGS

BF6-T003DD - ERDS HOST COMPUTER CONFIGURATION
BF6-T004DD - ERDS OPERATIONS CENTER EQUIPMENT CONFIGURATION
BF6-T010DD - ERDS COMMUNICATIONS INTERCONNECTIONS

FIGURE 3. NRC EMERGENCY RESPONSE DATA SYSTEM

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A1561

3. Implementation

3.1 ERDS Implementation Overview

As an ERDS participant, the licensee is expected to provide a real time data stream of data point values from an existing computer system(s) to NRC provided equipment. Since ERDS treats each reactor unit as an individual plant, a separate data stream is required for each reactor unit. The licensee is expected to provide the software to extract the data point engineering values from their system, organize them into a standard sequence, and to translate values from internal computer format into ASCII or EBCDIC. The data points to be included in the transmission are those which to the greatest possible extent satisfy the NRC desired parameter list. Any parameter which is not available to be electronically transmitted from a licensee system will not be backfit, but will instead be provided in verbal transmissions as needed during an emergency. In addition to the data point identifiers and values, the transmission should include the quality (validated, questionable, bad, etc.) of the data point value. The data will be transmitted to the NRC over dial-up telephone lines. The NRC is planning an upgrade of the Emergency Telecommunications System to a combination satellite and land lines network that would include ERDS, but the details of this upgrade have not been decided. In addition to the computer related aspects of ERDS implementation, administrative and quality assurance/configuration controls must be established. The steps necessary for a licensee to implement the ERDS program are outlined in the following sections.

3.2 ERDS Transmission/Reception Plan

The ERDS Transmission/Reception Plan (Appendix A) was developed by EI International, Inc., the NRC ERDS implementation contractor, to provide a procedure for licensees to follow in completing the computer application portions of the ERDS implementation. It establishes the sequence for correspondence, meetings, computer application development, and testing.

3.3 ERDS Communication Description And Survey Questionnaire

The ERDS Communications Description and Survey Questionnaire (Appendix B) was designed to provide the hardware, communications, data point, and administrative information necessary to design the ERDS system interface and data base for each reactor unit. When instructed to forward this questionnaire to the NRC in Appendix A, it should be forwarded to the NRC ERDS Project Manager with a copy to the NRC ERDS implementation contractor at the following addresses:

John R. Jolicoeur
ERDS Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop MNBB-3206
Washington, DC 20555

Tony P. LaRosa
EI International, Inc.
Post Office Box 50736
Idaho Falls, Idaho 83401

Also included in Appendix B is the description of the data communication methodology to be used in the ERDS implementation. Individual computer system limitations which prohibit the use of the generic communication protocol should be addressed in the questionnaire.

3.4 Data Point Library

The Data Point Library as described in Appendix C will be used to provide background information concerning each individual data point in the licensee data stream to better define the data point for the NRC technical teams. This provision was made to compensate for plant to plant differences in instrumentation. The data points outlined in the ERDS desired parameter list will be used to define generic displays for PWR and BWR units. Experience to date with early ERDS volunteers has shown a desire on the part of some licensees to send parameters not included in the desired list. The individual data bases for each unit will have a limited amount of additional space to allow for the addition of plant specific data points to the data stream. Plant specific data points which a licensee considers valuable to the assessment of critical safety functions may be submitted for consideration as possible additions to the data point library. Appendices D, E, F, G, H, and I provide amplifying information to be used to aid in computer point selection and Data Point Library completion.

3.5 System Isolation Requirements

While it is recognized that ERDS is not a safety system, it is conceivable that a licensee's ERDS interface could communicate with a safety system. In this case appropriate isolation devices would be required at these interfaces.

3.6 Administrative Implementation Requirements

ERDS implementation will entail a change in the way the licensees provide data to the NRC during a plant emergency. As such, Emergency Plan Implementing procedures should be modified to require ERDS to be activated upon notification of the NRC of the declaration of an Alert or higher emergency classification level.

Configuration management is an integral part of assuring the quality of a data network of this size. Part of the implementation plan must address procedures which will be followed to ensure the integrity of the ERDS hardware and software configuration at each reactor unit. These procedures should include provisions to allow NRC to review proposed system modifications which could affect the data communication protocol in advance of these changes to ensure that the changes are compatible with the ERDS. Changes to the Data Point Library should be submitted using the Data Point Library Reference File Form from Appendix C within thirty days of the change.

3.7 Periodic Testing

In order to verify system connectivity, periodic tests of the ERDS data link will be conducted with each licensee. The tests will be coordinated by the NRC and consist of operational tests of the licensee's ERDS data communications. The initial testing periodicity will be quarterly.

3.8 ERDS Questions And Answers

Appendix J provides answers to frequently asked questions concerning the ERDS implementation program.

3.9 Point Of Contact

Any questions concerning the ERDS implementation program should be referred to:

John R. Jolicoeur
ERDS Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop MNBB 3206
Washington, DC 20555

Tel: (301) 492-4155

4. References

1. U.S. Nuclear Regulatory Commission, "Report to Congress on NRC Emergency Communications", USNRC Report NUREG 0729, September 1980

2. U.S. Nuclear Regulatory Commission, "Report to Congress on the Acquisition of Reactor Data for the NRC Operations Center," USNRC Report NUREG 0730, September, 1980

3. U.S. Nuclear Regulatory Commission, "Emergency Response Data System Generic Letter No. 89-15," August 21, 1989

4. EI International, Inc., "Hardware Design Document," Report Number, NRC-201, July 1, 1988

Contents

APPENDIX A

EMERGENCY RESPONSE DATA SYSTEM (ERDS) TRANSMISSION/RECEPTION PLAN

EI International, Inc.

INTRODUCTION

The purpose of this document is to describe a plan which will allow the Nuclear Regulatory Commission (NRC) to survey and incorporate the utilities which have agreed to participate in the Emergency Response Data System (ERDS) program into the Emergency Response Data System.

SCOPE OF WORK

A significant portion of the work scope for the ERDS includes developing a communications link with each of the participating nuclear utilities. This link will establish a means for the utility's plant computer(s) to automatically transmit predefined data points to the ERDS computer at the request of the Nuclear Regulatory Commission.

To perform this function, both the ERDS and plant computers must be software and hardware compatible. This compatibility exists at the data transmission interface level.

HARDWARE REQUIREMENTS

Accomplishing the hardware interface for the ERDS is straightforward and consist of standard off-the-shelf components.

The hardware interface requires:

Single-feeder Sites:

- an RS-232C asynchronous modem control port and modem on each end of the communication line.

Multiple-feeder Sites:

- Multiple-feeder plants will require a multiplexer to be placed between the modems and computer(s) RS-232C ports.

SOFTWARE TASKS

The software tasks associated with the data interface are plant-specific with a data reception communications program residing on the ERDS computer. In certain situations limited custom software will be written for the ERDS.

The plant-specific software includes transmitting the actual data points to the Data Point Library (DPL) in the ERDS. These data points will essentially comprise a database (formally referred to as the DPL) which will reside on the ERDS and be made available to the users whenever a utility is transmitting data.

ESTABLISHING THE DPL and the PLANT ATTRIBUTE LIBRARY (PAL)

Since the focal point of the ERDS is the DPL, a concentrated effort must be put forth to ensure that the DPL for each utility is accurate and that the software protocol for transferring these values is known to the ERDS software.

The ERDS database, or DPL, contains specific information about each data point, i.e., point ID, description, engineering units, etc. Storing this information in the ERDS eliminates the necessity to transmit the information with each data set.

The Plant Attribute Library (PAL) contains the communications information necessary to communicate with each utility and remains on file within the system as a reference to establish the utility's software protocol requirements which the ERDS can expect to accommodate during data transmission. Without the PAl information, it would not be possible to communicate with the plant computer.

INCORPORATING THE UTILITY INTO THE ERDS

The plan for incorporating each utility into the ERDS consists of the steps outlined on pages three (3), four (4), and five (5) of this plan and are common among all the participating licensees.

In preparing this plan, the activities required to incorporate the utility into the ERDS were identified based on experience gained from the few site surveys that have been conducted to date. Understanding that not all utilities operate in the same manner, the steps described herein represent the basic or minimum effort required to incorporate the plant into the ERDS.

Depending on the utility's and NRC's schedule, tasks can be added or rearranged to accommodate the situation.

STEPS REQUIRED TO INCORPORATE THE PLANT INTO THE ERDS

Once a utility decides to participate in the ERDS program, the required activities are:

- 1) The NRC notifies the contractor, EI International, Inc. (EI), that a utility has received a site survey questionnaire.

This questionnaire consists of several enclosures which inquire about the plant computer capabilities and the available data points to be transmitted to the ERDS.

Identification of these data points is the most tedious effort required of the utility because the response essentially forms the ERDS database (the DPL) and, as described in previous sections, the DPL is the focal point of the ERDS. Efforts must be made to ensure the accuracy of the DPL and that the software protocol for transferring these values is known to the ERDS software.

- 2) After the utility has received the questionnaire, they will be contacted by EI.

EI personnel will contact the utility to discuss the items within the site survey questionnaire along with typical utility responses, to describe EI's involvement in the ERDS program, to answer general and specific questions regarding what is expected of both the utility and EI, and to convey EI's experiences and/or problems learned from other participating utilities. If the utility was not part of the pre-ERDS survey, an EI representative will assist the utility in selecting plant data points which fulfill the NRC's requested parameter list.

- 3) A site visit will be arranged.

A visit is not mandatory but should be conducted prior to the licensee's return of the DPL and PAL in an effort to minimize errors in answering the questionnaire. If necessary, the visit can occur after the DPL and PAL are submitted. In a very few circumstances, a visit may not be necessary; however, this is not recommended.

- 4) The NRC will install phone lines at the site.

- 5) The utility then answers and returns the site survey questionnaire containing the DPL and PAL information to the NRC.

Verbal communications between the utility's contact and EI personnel are ongoing during this phase in preparation for software development on both ends of the data link and establishment of the ERDS database.

- 6) If the plant's computer system requires customized ERDS reception software, specific ERDS code will be developed and implemented by EI.

This may not be required if the licensee's system can conform to the "generic" software protocols of the ERDS.

- 7) In parallel with EI software development, the utility will design and write their data transmission software.

During this phase, EI will continue to provide consulting assistance to the utility's programmers in preparation for a preliminary software test. Any required transmission equipment including modem(s) and, if necessary, multiplexer(s) will be shipped to the utility during this phase.

- 8) Preliminary software testing is the next step and is the first attempt at transferring data between the plant and ERDS computers. The preliminary software test performs initial data transmission testing of the utility's software and any custom code EI has developed. This is in actuality the software debugging period and problems are to be expected.

This step is complete when data can be transmitted by the utility's plant or development computer and the ERDS computer without error.

- 9) Following the preliminary software tests and the initial data transference between the plant and the ERDS computers, a formal test will be conducted at EI prior to adding the licensee to the ERDS.

Upon successful completion of this test, the DPL, PAL, and any special software routines will be incorporated into the ERDS production computer. At this time, the utility will be transmitting data from their plant computer and not their development system.

- 10) A formal test is then conducted on the ERDS computer at the Operations Center. This is the final test to demonstrate system functionality. Again, data transmission will be from the designated plant computer system.

- 11) The final step in the schedule has the utility on-line with all development and testing completed.

SUMMARY

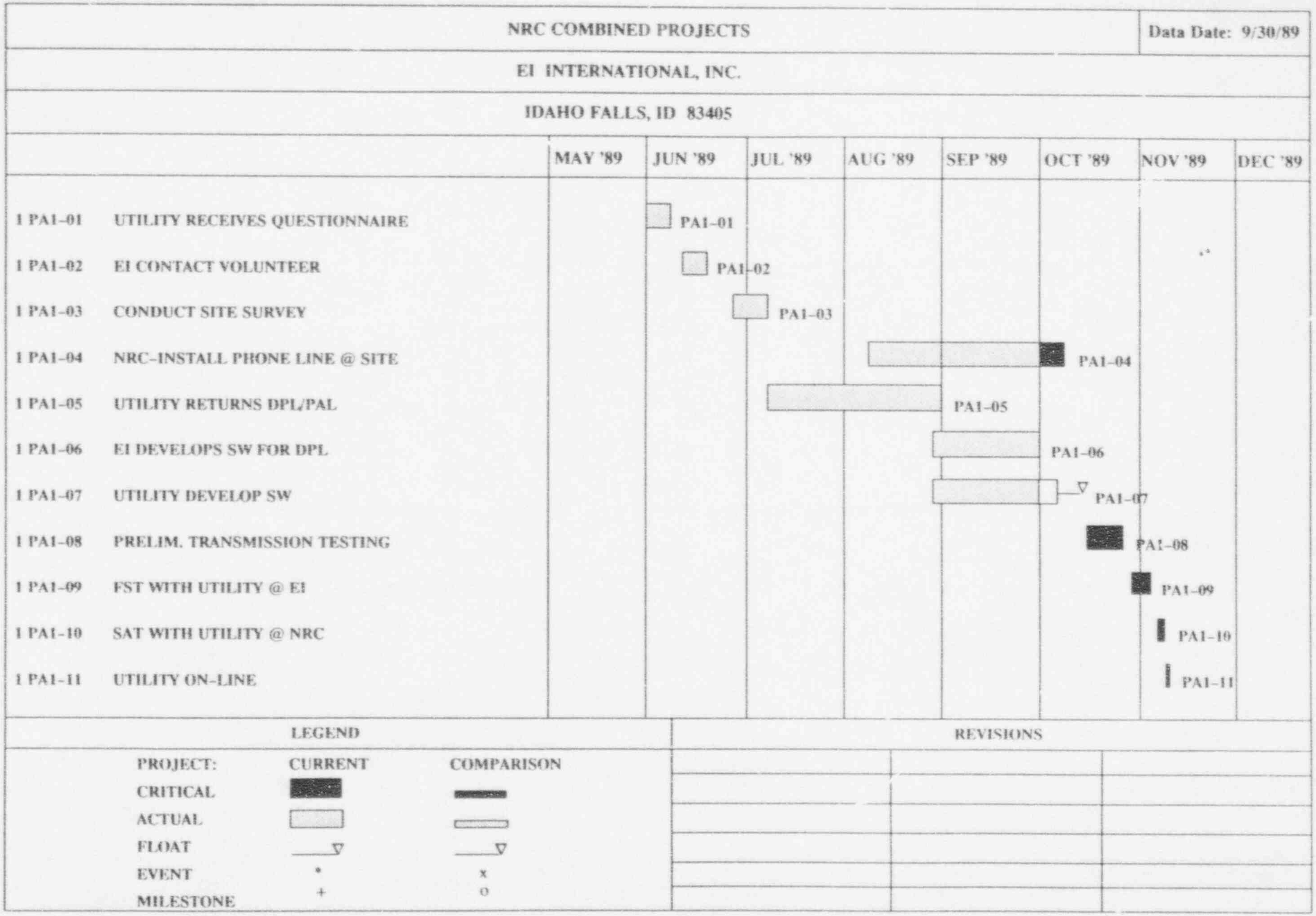
The eleven (11) steps as outlined on the previous pages are to be used as a guideline for scheauling and accomplishing the tasks required to incorporate the utilities into the ERDS. Again, understanding that not all utilities operate in the same manner, the steps as previously outlined represent only the basic approach to the efforts required. Tasks can be added or rearranged to accommodate each utility.

The most significant portion of the work scope of this plan is the development of the communications link with each of the participating utilities. While the hardware interface for the ERDS is straightforward, consisting of off-the-shelf hardware, the software tasks are plant-specific and require a dedicated effort in establishing the Data Point Library and the Plant Attribute Library. The ERDS Communications Description and Survey Questionnaire (site survey questionnaire) explains in detail the purpose of collecting this data, provides descriptions and examples of the data streams the ERDS is expecting to see transmitted over the communications lines, and provides samples of forms to be filled out and returned as part of implementing this Transmission/Reception Plan.

It is of vital importance that a dedicated effort be put forth to ensure the accuracy of the information in the questionnaire (the DPL) for each utility. The contractor's (EI's) personnel are available to assist the utility during all phases of this plan including the selection of hardware and software interfaces and, most importantly, during the selection of the required data points.

SCHEDULE

The attached sample schedule (Attachment A) presents a visual display of the milestones associated with the implementation of this plan and is an actual schedule of a participating plant. This schedule can be used as a guide for each utility to project schedules and testing dates. The scheduled milestones represent the eleven (11) steps as outlined in this plan and are scheduled for Palisades (PA1) plant.



ATTACHMENT A — Sample Schedule

APPENDIX B

ERDS COMMUNICATIONS DESCRIPTION AND SURVEY QUESTIONNAIRE

The following is a questionnaire pertaining to the Nuclear Regulatory Commission's (NRC) Emergency Response Data System (ERDS). It consists of a series of questions concerning plant I/0 points, software protocols, data formats, transmission frequencies, and other plant computer specific information to be used in the ERDS computer database files. Also, included here are descriptions and examples of data streams that the NRC is expecting to see transmitted over the communication line.

The purpose of collecting the data is to develop a plant-specific database that will be retrieved into the ERDS once the system is activated by a utility. It will also be used to design and implement ERDS software that can receive the utility's data transmission. In essence, this information will provide the basis for building a profile of the plant in the ERDS database.

In some cases, the I/0 point data may be distributed over several computers. The ERDS considers this situation a multi-feeder site and Section IV must be filled out for each feeder.

I. Contacts

Note: Please provide name, title, mailing address, and phone number.

A. Survey Coordinator (i.e., contact for later clarification of questionnaire answers):

B. Computer Hardware Specialist(s):

C. Systems Software Specialist(s):

D. Application-level Software Specialist(s):

E. Telephone Systems Specialist(s):

II. ERDS Communications Description

A. Hardware

The following hardware will be supplied:

- for a single-feeder site:

  Codex 2234 modem – V.22 2400 bps, asynchronous, auto-dialing, auto-answer, error-correcting, using the AT command set

- for a multiple-feeder site:

  Codex 6015 multiplexer,
  Codex 2260 modem – V.32 9600 bps, asynchronous, auto-dialing, auto-answer, error-correcting, using the AT command set

The modems are intended to be operated in the auto-reliable link mode (referred to as MNP in the modem manuals) with speed conversion and flow control enabled. Speed conversion allows the computer to communicate with the modem at a baud rate which is independent of the baud rate the modem is using to communicate with the remote modem. This feature is important because the modems have the ability to adjust their transmission rate downward if the remote modem is operating at a lower speed. However, in order to use speed conversion, the site computer must support some form of flow control. Three types of flow control are supported by the modems: XON/XOFF, RTS/CTS, and DTR/CTS. All of the above features are discussed in the modem manuals.

B. Software

1. Data Transmission

All transmissions, from both the site and the ERDS, will be terminated with a carriage return (<CR>).

a. Site will initiate a link request in ASCII using:

   - the three-character site designator,
   - the word LINK,
   - local site time and date in the format MM/DD/YY/HH:MM:SS, and
   - a <CR>.

   If the site does not receive a response from the ERDS within one minute, it should send another link request message and continue sending them at one-minute intervals. If more than five minutes elapses without a response, site personnel should notify the NRC before disconnecting the line.

b. ERDS will respond in ASCII with:

   - the three-character site designator,
   - the word ACCEPTED or DENIED, and
   - a <CR>.

   If the ERDS responds with the denied message, the site should wait one minute and then send a link request message and continue sending them at one-minute intervals. If more than five minutes elapses without a response, site personnel should notify the NRC before disconnecting the line.

c. When the ERDS is ready to receive data, it will send an initiate message in ASCII using:

- the three-character site designator,
- the word INITIATE, and
- a <CR>.

If the ERDS does not send an initiate message within one minute of the accept message, the site should send the link request message (described in Section II.B.1.a.).

d. Upon receipt of the initiate message, the plant begins transmission of data at a 15-second rate. The data string consists of:

- a header containing the three-character site designator and date and time in the format MM/DD/YY/HH:MM:SS,
- the data packet sequenced with point identifier, value, and quality tag,
- a trailer containing the checksum value of the data packet, and a <CR>.

e. When the site or ERDS wishes to terminate the connection, an ASCII message will be sent containing:

- the three-character site designator,
- the word TERMINATE, and
- a <CR>.

f. If a site is inadvertently terminated (due to loss of communications or receipt of terminate message) and the incident is still underway, the site should reconnect with the ERDS by redialing and using the reconnect link request message. This message is in ASCII and will contain:

- the three-character site designator,
- the word RECONNECT,
- local site time and date in the format MM/DD/YY/HH:MM:SS, and
- a <CR>.

Upon receipt of this message, the ERDS will respond with the accept and initiate messages as described in Sections II.B.1.b and II.B.1.c. If the ERDS responds with a link deny message (described in Section II.B.1.b), the site should stop trying to reconnect and send a link request message (described in Section II.B.1.a). If the ERDS does not respond to the site's reconnect request within one minute, the site should send another reconnect request and continue sending reconnect requests once a minute. If more than five minutes elapses without a response, site personnel should notify the NRC before disconnecting the line. It is the responsibility of the site to monitor the outgoing line for loss of communications.

2. Data Format

The following three delimiters have been identified:

(1) field delimiter (*),

(2) data set delimiter (\), and

(3) carriage return (<CR>).

Note: The length of the messages sent by the ERDS (e.g., ACCEPTED, DENIED, INITIATE, TERMINATE) are variable and it is recommended that the site software use the data set delimiter as the message delimiter for messages received from the ERDS.

a. Link requests will be in ASCII as described in II.B.1.a. with each field separated by a field delimiter and the request terminated with a data set delimiter. For example, PA1*LINK*01/12/89/11:48:50\<CR>.

b. The ERDS response will be in ASCII as described in II.B.1.b. with each field separated by a field delimiter and the response terminated with a data set delimiter. For example, PA1*ACCEPTED\<CR>.

c. When the ERDS is ready to receive data it will respond in ASCII as described in II.B.1.c with each field separated by a field delimiter and the response terminated with a data set delimiter. For example, PA1*INITIATE\<CR>.

d. Data streams will be in ASCII and will consist of three parts (header, data, and trailer) as described in II.B.1.d. with each field separated by a field delimiter and each of the three parts separated by a data set delimiter. For example,

Header: PA1*01/12/89/11:50:30\

Data: B21CP004*-0.1234E+00*3*.....(for each parameter)\

Trailer: 0000056000\<CR>

e. The point identifier may be up to 12 characters in length.

f. The value may be up to 20 characters in length.

g. The following quality tags will be accepted by the ERDS:

Good	= 0	Value is within range tolerance for discreet points or input points are within tolerance for composed points.
Off-scan	= 1	Point is currently out-of-service.
Suspect	= 2	Value is not bad yet should not be considered good. This quality will occur primarily on composed values when enough good inputs are present to allow the calculation to be made yet a bad quality on other inputs may make the result questionable.
Bad	= 3	Value is not within tolerance for discreet points or calculation of a composed point may not be made due to the qualities of its inputs.

Unknown	= 4	No quality indicator available.
Operator Entered	= 5	Value has been manually entered, overriding the discreet or composed value.
High Alarm	= 6	Value is in high alarm.
Low Alarm	= 7	Value is in low alarm.

h. The checksum which accompanies each update set will be an integer value calculated by summing each of the bytes of the transmission, up to and including the dataset delimiter following the body of the update set (the body of the update set being the portion containing the parameter, value, and quality indications). This integer checksum value will then be encoded into the update set as a 10-digit value, left-padded with zeros as required to fill the 10-digit field. The checksum is the sum of the transmitted bytes.

i. The reconnect link request message will be in ASCII as described in Section II.B.1.f with each field separated by a field delimiter and the request terminated with a data set delimiter. For example, PA1*RECONNECT*01/12/89/11:48:50\<CR>.

3. Protocol

a. ERDS will use XON/XOFF to stop, resume, or suspend data transmission for the site.

b. Communication parameters:

- eight data bits
- 1 stop bit
- parity = none

4. Exceptions

Please note any exceptions which must be taken to Section II and explain why.

III. Selection Of Data Feeders

A. How many data feeders are there (six maximum)?

B. Identify the selected data feeders and provide the following for each:

(1) a short description of the categories of data points it will provide (e.g., met, rad, or plant data points, by unit) and

(2) the rationale for selecting it if another system can also provide its categories of data points.

C. Which data feeder is the site time determining feeder? This should be the feeder which is providing the majority of the data points.

IV. Data Feeder Information

Note: A new Section IV must be filled out for each feeder system selected.

General Questions

1. Identification of Data Feeder

a. What is the name in local parlance given to this data feeder (e.g., Emergency Response Information System)? Please give both the acronym and the words forming it.

b. Is this the site time determining feeder?

c. What is the update frequency of this feeder (in seconds)?

2. Hardware/Software Environment

a. Identify the manufacturer and model number of the data feeder hardware.

b. Identify the operating system.

c. What method of timekeeping is implemented on this feeder system (Daylight Savings, Standard, Greenwich)?

d. In what time zone is this feeder located?

3. **Data Communication Details**

a. Can this data feeder provide asynchronous serial data communication (RS-232-C) with full-modem control?

b. Will this feeder transmit in ASCII or EBCDIC?

c. Can this feeder transmit at a serial baud rate of 2400 bps? If not, at what baud rate can it transmit?

d. Does the operating system support XON/XOFF flow control?

   1. Are any problems foreseen with the NRC using XON/XOFF to control the transmission of data?

e. If it is not feasible to reconfigure a serial port for the ERDS linkup (i.e., change the baud rate, parity, etc.), please explain why.

f. Can the serial port dedicated to the ERDS be configured so that the NRC need not emulate a specific brand of terminal (i.e., can it be configured to be a "vanilla" terminal)?

g. Do any ports currently exist for the ERDS linkup?

1. If not, is it possible to add additional ports?

2. If yes, will the port be used solely by the ERDS or shared with other non-emergency-time users? Give details.

4. Data Feeder Physical Environment and Management

a. Where is the data feeder located in terms of the TSC, EOF, and control room?

b. Is the data feeder protected from loss of supply of electricity?

c. Is there a human operator for this data feeder?

1. If so, how many hours a day is the feeder attended?

APPENDIX C
DATA POINT LIBRARY

The Data Point Library is a site-specific database residing on the ERDS computer which expands upon the basic information in a typical data point dictionary. The data being displayed at the NRC's Operations Center for the ERDS parameter will be the same as the plant's Emergency Response Team's data. That is, it will have the same value, timestamp, and be in the same engineering units. This requires that the Operations Center personnel adjust their thinking to accommodate the plant, functioning in terms of the plant's unique design and communicating with the plant's Response Team in the latter's unique engineering and operational "language". In order to do this, the Operations Center personnel need information which relates the data both to the plant's design and to the manner in which the plant's team utilizes and reacts to the data.

The types of information contained in the Data Point Library are the data point identifier, description, engineering units, range, alarms and/or technical specification limits and engineering system data. There will be one record in the plant's Data Point Library for each data point the plant will be sending to the ERDS.

Because the points selected for transmission to the ERDS are indicative of plant "health" and are associated with Critical Safety Functions, they are the indicators the plant's Response Team uses to determine the proper actions to take to mitigate an incident. Where required and useful, the Data Point Library will present textual information to the Operations Center user to provide information supplementing the point's value which will be useful in understanding how the plant team interprets the data. For instance, associated with a transmitted data point representing the reactor vessel level, the Data Point Library should contain the physical zero reference point, conversion factor for the height above the top of active fuel, type of detectors, effects of running reactor coolant pumps, effects of cold calibration, effects of elevated containment temperature, etc. Associated with a reactor water storage tank level transmitted as a percentage should be the capacity of that tank in gallons, number of reactor quality water storage tanks at the plant site, zero reference point conversion factor from percent to gallons, etc.

The Data Point Library will be particularly useful to the Operations Center user when evaluating the plant's action in predicting off-site radioactive releases. Associated with an effluent gaseous release data point expressed in CPM, the Data Point Library Reference Sheet should indicate the assumptions regarding isotopic mix, the current calibration factors of detectors, the discharge point or points for monitored releases, expected stack flow rates under various fan combinations, and any default values used by the plant team in their calculations.

Two examples of typical Data Point Library entries are included. The first is an example for a BWR and the second is an example for a PWR.

BWR DATA POINT LIBRARY REFERENCE FILE

Date:	06/05/89
Reactor Unit:	XYZ
Data Feeder:	N/A
NRC ERDS Parameter:	CST Level
Point ID:	C345Z04
Plant Spec Point Desc.:	CS TNK IA LVL
Generic/Cond Desc.:	Condensate Storage Tank A Level
Analog/Digital:	A
Engr Units/Dig States:	%
Engr Units Conversion:	Each 1% = 1692 Gallons
Minimum Instr Range:	0
Maximum Instr Range:	100
Zero Point Reference:	SEALEV
Reference Point Notes:	At 0% 245,000 Gals Remain In Tank
PROC or SENS:	P
Number of Sensors:	2
How Processed:	Average
Sensor Locations:	245,000 Gal Above Tank Bottom
Alarm/Trip Set Points:	Low Level At 12%
NI Detector Power Supply Cut-off Power Level:	N/A
NI Detector Power Supply Turn-on Power Level:	N/A
Instrument Failure Mode:	Low
Temperature Compensation For DP Transmitters:	N/A
Level Reference Leg:	N/A
Unique System Desc.:	This averaged sensor reading is for the normally used volume of the tank. The remaining 245,000 gallons are monitored by two discrete alarms at 150,000 and 50,000 gallons total remaining tank contents. Total tank volume is 414,200 gallons.

NOTE: A second identical tank normally dedicated to XYZ Unit 1 is available for cross-connecting to this tank at the bottom (ECCS) suction line.

PWR DATA POINT LIBRARY REFERENCE FILE

Date:	06/05/89
Reactor Unit:	ABC
Data Feeder:	ERIS
NRC ERDS Parameter:	AX FD FL 1/A
Point ID:	AF105A
Plant Spec Point Desc.:	AFW Flow SG 11 MTR
Generic/Cond Desc.:	AFW Flow SG 11 Frm Elec AFW Pump
Analog/Digital:	A
Engr Units/Dig States:	GPM
Engr Units Conversion:	N/A
Minimum Instr Range:	0
Maximum Instr Range:	500
Zero Point Reference:	N/A
Reference Point Notes:	N/A
PROC or SENS:	S
Number of Sensors:	1
How Processed:	N/A
Sensor Locations:	On Line To SG 11 Outside Containment
Alarm/Trip Set Points:	High Flow At 500 GPM
NI Detector Power Supply Cut-off Power Level:	N/A
NI Detector Power Supply Turn-on Power Level:	N/A
Instrument Failure Mode:	Low
Temperature Compensation For DP Transmitters:	N/A
Level Reference Leg:	N/A
Unique System Desc.:	There are one electric and two turbine-driven AFW pumps. The electric pump has dedicated discharge lines to each SG. The flow element for this point represents the last sensor prior to the line entering containment. The two turbine-driven pumps use separate piping to the SGs. Maximum rated flow for this pump is 450 GPM. Shutoff head is 1200 PSIG.

DATA POINT LIBRARY REFERENCE FILE

Date: __/ __/ __

Reactor Unit: ______________________

Data Feeder: ______________________

NRC ERDS Parameter: ______________________

Point ID: ______________________

Plant Spec Point Desc.: ______________________

Generic/Cond Desc.: ______________________

Analog/Digital: __________

Engr Units/Dig States: __________

Engr Units Conversion: ______________________

Minimum Instr Range: __________

Maximum Instr Range: __________

Zero Point Reference: __________

Reference Point Notes: ______________________

PROC or SENS: ______________________

Number of Sensors: __________

How Processed: __________

Sensor Locations: ______________________

Alarm/Trip Set Points: ______________________

NI Detector Power Supply
Cut-off Power Level: ______________________

NI Detector Power Supply
Turn-on Power Level: ______________________

Instrument Failure Mode: ______________________

Temperature Compensation
For DP Transmitters: ______________________

Level Reference Leg: ______________________

Unique System Desc.: ______________________

____

____

____

____

APPENDIX D

DATA POINT LIBRARY REFERENCE FILE DEFINITIONS

Date:	The date that this form is filled out or modified. (Eight characters)
Reactor Unit:	The nuclear power plant name and abbreviation from the enclosed list of sites. (Three characters)
Data Feeder:	If there is more than one data feeder for your system, enter the acronym for the data feeder from which the point comes. If there is only one data feeder, enter "N/A" in this field. (Ten characters)
NRC ERDS Parameter:	One of the parameters from the enclosed BWR or PWR parameter list. A single value should be transmitted for each parameter for each loop. If not on the list, insert "Not Listed" or "NL". (Twelve characters)
Point ID:	Alphanumeric point description used to label the point during transmission. (Twelve characters)
Plant-Specific Point Description:	Licensee computer point description for the transmitted point. (Forty characters).
Generic Or Condensed Description:	Parameter description from the enclosed list of points for a BWR or PWR. If not on the list, condense the plant-specific point description. (Thirty-two characters)
Analog/Digital:	"A" if the signal is analog or numerical or "D" if the signal is off/on. (One character)
Engineering Units Or Digital States:	Engineering units used by the licensee for display on licensee output devices. Use the engineering units abbreviations from the enclosed list when possible. When specifying pressure, use "PSIA" or "PSIG" rather than "PSI". For digital signals, give the "OFF" and "ON" state descriptors. (Twelve characters)
Engineering Units Conversion:	Notes about any special features of the A/D conversion and scaling. (Forty characters)
Minimum Instrument Range:	Engineering units value below which data cannot go (bottom-of-scale value). (Ten characters)
Maximum Instrument Range:	Engineering units value above which data cannot go (top of-scale value). (Ten characters)
Zero Reference Point:	Zero-point of engineering units scale, used primarily for levels or heights. Use the zero reference point abbreviations from the enclosed list when possible. (Six characters)

Reference Point Notes:	Notes about the reference point or other important and special features of the parameter. (Forty characters)
PROC or SENS:	Is the point formed by processing more than one signal, or is the source a single sensor ("P" or "S")? (One character)
Number of Sensors:	The number of signals processed in a full calculation assuming no bypassed or inoperative sensors. (Three characters)
How Processed:	The processing algorithm (sum, average, weighted average, highest, lowest, or a short description). (Forty characters)
Sensor Locations:	Description of the location(s) of the instrument(s) used. (Forty characters)
Alarm or Trip Setpoints:	The most important setpoints for the parameter. State whether the limit is high or low. (Forty characters)
NI Detector Power Supply Cut-off Power Level:	The power level at which the power supply for the NI detector switches off. (Fifteen characters)
NI Detector Power Supply Turn-on Power Level:	The power level at which the power supply for the NI detector switches on. (Fifteen characters)
Instrument Failure Mode:	The mode in which this instrument fails. Possible answers are HIGH, MEDIUM, or LOW. If available, provide the numeric value at which the instrument fails. (Thirty characters)
Temperature Compensation For DP Transmitters:	This question pertains to differential pressure transmitters. Possible answers are "YES" or "NO" ("Y" or "N"). If the answer is "NO", please attach a copy of the correction curve. (One character)
Level Reference Leg:	The type of level measurement (dry or wet) used on the level reference leg. (Three characters)
Unique System Description:	Additional important information which will assist the NRC Operations Center personnel in understanding how the plant team interprets the data. (600 characters)

APPENDIX E

CRITICAL SAFETY FUNCTION PARAMETERS

Critical Safety Function Parameters For Boiling Water Reactors

Reactivity Control:	Parameter Description	Typical Units
NI POWER RNG	Nuclear Instruments, Power Range	%
NI INTER RNG	Nuclear Instruments, Intermediate Range	AMP
NI SOURC RNG	Nuclear Instruments, Source Range	C/SEC
CORE COOLING		
REAC VES LEV	Reactor Vessel Water Level	IN
MAIN FD FLOW	Feedwater Flow into the Reactor System	%
RCIC FLOW	Reactor Core Isolation Cooling Flow	GPM
RCS INTEGRITY		
RCS PRESSURE	Reactor Coolant System Pressure	PSIG
HPCI FLOW	High Pressure Coolant Injection Flow	GPM
LPCI FLOW	Low Pressure Coolant Injection Flow	GPM
CR SPRAY FL	Core Spray Cooling System Flow	GPM
DW FD SMP LV	Drywell Floor Drain Sump Level	IN
RADIOACTIVITY CONTROL		
EFF GAS RAD	Radioactivity of Released Gasses	MCI/HR
EFF LIQ RAD	Radioactivity of Released Liquids	MCI/HR
CND A/E RAD	Condenser Air Ejector Radioactivity	C/MIN
DW RAD	Radiation Level in the Drywell	R/HR
MN STEAM RAD	Radiation Level of the Main Steam Line	MR/HR
CONTAINMENT CONDITIONS		
DW PRESS	Drywell Pressure	PSIG
DW TEMP	Drywell Temperature	F
SP TEMP	Suppression Pool Temperature	F
SP LEVEL	Suppression Pool Water Level	IN
H2 CONC	Drywell or Torus Hydrogen Concentration	%
O2 CONC	Drywell or Torus Oxygen Concentration	%
MISCELLANEOUS PARAMETERS		
CST LEVEL	Condensate Storage Tank Level	%
WIND SPEED	Wind Speed at the Reactor Site	MPH
WIND DIR	Wind Direction at the Reactor Site	DEG
STAB CLASS	Air Stability at the Reactor Site	

Critical Safety Function Parameters For Pressurized Water Reactors

Reactivity Control	Parameter Description	Typical Units
NI POWER RNG	Nuclear Instruments, Power Range	%
NI INTER RNG	Nuclear Instruments, Intermediate Range	AMP
NI SOURC RNG	Nuclear Instruments, Source Range	C/SEC
CORE COOLING		
REAC VES LEV	Reactor Vessel Water Level	IN
TEMP CORE EX	Highest Temperature at the Core Exit	F
SUB MARGIN	Saturation Temperature—Highest CET	F
CORE FLOW	Total Reactor Coolant Flow	MLB/HR
STEAM GENERATORS		
SG LEVEL 1/A	Steam Generator 1 (or A) Water Level	%
SG LEVEL 2/B	Steam Generator 2 (or B) Water Level	%
SG LEVEL 3/C	Steam Generator 3 (or C) Water Level	%
SG LEVEL 4/D	Steam Generator 4 (or D) Water Level	%
SG PRESS 1/A	Steam Generator 1 (or A) Pressure	PSIG
SG PRESS 2/B	Steam Generator 2 (or B) Pressure	PSIG
SG PRESS 3/C	Steam Generator 3 (or C) Pressure	PSIG
SG PRESS 4/D	Steam Generator 4 (or D) Pressure	PSIG
MN FD FL 1/A	Stm Gen 1 (or A) Main Feedwater Flow	LBM/HR
MN FD FL 2/B	Stm Gen 2 (or B) Main Feedwater Flow	LBM/HR
MN FD FL 3/C	Stm Gen 3 (or C) Main Feedwater Flow	LBM/HR
MN FD FL 4/D	Stm Gen 4 (or D) Main Feedwater Flow	LBM/HR
AX FD FL 1/A	Stm Gen 1 (or A) Auxiliary FW Flow	GPM
AX FD FL 2/B	Stm Gen 2 (or B) Auxiliary FW Flow	GPM
AX FD FL 3/C	Stm Gen 3 (or C) Auxiliary FW Flow	GPM
AX FD FL 4/D	Stm Gen 4 (or D) Auxiliary FW Flow	GPM
HL TEMP 1/A	Stm Gen 1 (or A) Inlet Temperature	F
HL TEMP 2/B	Stm Gen 2 (or B) Inlet Temperature	F
HL TEMP 3/C	Stm Gen 3 (or C) Inlet Temperature	F
HL TEMP 4/D	Stm Gen 4 (or D) Inlet Temperature	F
CL TEMP 1/A	Stm Gen 1 (or A) Outlet Temperature	F
CL TEMP 2/B	Stm Gen 2 (or B) Outlet Temperature	F
CL TEMP 3/C	Stm Gen 3 (or C) Outlet Temperature	F
CL TEMP 4/D	Stm Gen 4 (or D) Outlet Temperature	F

Critical Safety Function Parameters For Pressurized Water Reactors
(Cont'd)

Reactivity Control	Parameter Description	Typical Units
RCS INTEGRITY		
RCS PRESSURE	Reactor Coolant System Pressure	PSIG
PRZR LEVEL	Primary System Pressurizer Level	%
RCS CHG/MU	Primary System Charging or Makeup Flow	GPM
HP SI FLOW	High Pressure Safety Injection Flow	GPM
LP SI FLOW	Low Pressure Safety Injection Flow	GPM
CTMNT SMP NR	Containment Sump Narrow Range Level	IN
CTMNT SMP WR	Containment Sump Wide Range Level	IN
RADIOACTIVITY CONTROL		
EFF GAS RAD	Radioactivity of Released Gasses	MCI/HR
EFF LIQ RAD	Radioactivity of Released Liquids	MCI/HR
COND A/E RAD	Condenser Air Ejector Radioactivity	C/MIN
CNTMNT RAD	Radiation Level in the Containment	R/HR
RCS LTDN RAD	Rad Level of the RCS Letdown Line	C/SEC
MAIN SL 1/A	Stm Gen 1 (or A) Steam Line Rad Level	MR/HR
MAIN SL 2/B	Stm Gen 2 (or B) Steam Line Rad Level	MR/HR
MAIN SL 3/C	Stm Gen 3 (or C) Steam Line Rad Level	MR/HR
MAIN SL 4/D	Stm Gen 4 (or D) Steam Line Rad Level	MR/HR
SG BD RAD 1A	Stm Gen 1 (or A) Blowdown Rad Level	MR/HR
SG BD RAD 2B	Stm Gen 2 (or B) Blowdown Rad Level	MR/HR
SG BD RAD 3C	Stm Gen 3 (or C) Blowdown Rad Level	MR/HR
SG BD RAD 4D	Stm Gen 4 (or D) Blowdown Rad Level	MR/HR
CONTAINMENT CONDITIONS		
CTMNT PRESS	Containment Pressure	PSIG
CTMNT TEMP	Containment Temperature	F
H2 CONC	Containment Hydrogen Concentration	%
MISCELLANEOUS PARAMETERS		
BWST LEVEL	Borated Water Storage Tank Level	%
WIND SPEED	Wind Speed at the Reactor Site	MPH
WIND DIR	Wind Direction at the Reactor Site	DEG
STAB CLASS	Air Stability at the Reactor Site	

APPENDIX F

ENGINEERING UNITS CODING SCHEME

PSIG	=	Pounds per square inch gauge
PSIA	=	Pounds per square inch absolute
INH_2O	=	Inches of Water Pressure
%	=	Percent
INCHES		
FEET		
FT&IN	=	Feet and inches
FTDEC	=	Feet and decimal feet
GAL	=	Gallons
LB	=	Pounds or pounds mass
GPM	=	Gallons per minute
KGPM	=	Thousands of gallons per minute
LB/HR	=	Pounds per hour
KLB/HR	=	Thousands of pounds per hour
MLB/HR	=	Millions of pounds per hour
CPM	=	Counts per minute
CPS	=	Counts per second
AMPS		
MAMPS	=	Milliamps
μAMPS	=	Microamps
DEGF	=	Degrees Fahrenheit
DEGC	=	Degrees Centigrade
MR/HR	=	Millirem per hour
R/HR	=	Rem per hour
CI/CC	=	Curies per CC
CI/ML	=	Curies per ML
μCI/CC	=	Microcuries per CC
μCI/ML	=	Microcuries per ML
CI/S	=	Curies per second
μCI/S	=	Microcuries per second
DEGFR	=	Degrees true (for wind direction from)
DEGTO	=	Degrees true (for wind direction to)
DF/FT	=	Degrees Fahrenheit per foot
DC/M	=	Degrees Centigrade per meter
DC/HM	=	Degrees Centigrade per 100 meters
DF/HFT	=	Degrees Fahrenheit per 100 feet
STABA	=	Stability class in form of A - G
STABI	=	Stability class in form of integer, where A = 1, B = 2
MPH	=	Miles per hour
M/S	=	Meters per second

APPENDIX G

ZERO REFERENCE CODING SCHEME

This field applies to levels and heights only. Leave it blank for temperatures, pressure, and flows. Give the physical point represented by the number zero for the parameter from the choices below.

TAF	=	Top of active fuel
UPHEAD	=	Upper head
LWHEAD	=	Lower head
MSSKRT	=	Moisture separator skirt
TOPHTR	=	Top of pressurizer heater bank
SURGE	=	Surge line penetration
SPRAY	=	At the spray nozzle
UTUBES	=	Top of S/GU tubes
TUBSHT	=	At S/G tube sheet
TNKBOT	=	Bottom of tank sump (e.g., CST)
COMPLX	=	Reference too complex for database entry
CNTFLR	=	Containment floor
SEALEV	=	Mean sea level

APPENDIX H

CODING SCHEME
FOR UNIT NAME AND UNIT ID

ARKANSAS NUCLEAR ONE-1	AN1
ARKANSAS NUCLEAR ONE-2	AN2
BEAVER VALLEY-1	BV1
BEAVER VALLEY-2	BV2
BELLEFONTE-1	BE1
BELLEFONTE-2	BE2
BIG ROCK POINT	RP1
BRAIDWOOD-1	BR1
BRAIDWOOD-2	BR2
BROWNS FERRY-1	BF1
BROWNS FERRY-2	BF2
BROWNS FERRY-3	BF3
BRUNSWICK-1	BK1
BRUNSWICK-2	BK2
BYRON-1	BY1
BYRON-2	BY2
CALLOWAY-1	CW1
CALVERT CLIFFS-1	CC1
CALVERT CLIFFS-2	CC2
CATAWBA-1	CT1
CATAWBA-2	CT2
CLINTON-1	CL1
COMANCHE PEAK-1	CP1
COMANCHE PEAK-2	CP2
CONNECTICUT YANKEE	HN1
COOK-1	CK1
COOK-2	CK2
COOPER	CO1
CRYSTAL RIVER-3	CR3
DAVIS BESSE-1	DB1
DIABLO CANYON-1	DC1
DIABLO CANYON-2	DC2
DRESDEN-2	DN2
DRESDEN-3	DN3
DUANE ARNOLD	DA1
FARLEY-1	FA1
FARLEY-2	FA2
FERMI-2	FE2
FORT CALHOUN-1	FC1
FORT ST. VRAIN-1	FV1
GINNA	GI1
GRAND GULF-1	GG1
HATCH-1	HT1
HATCH-2	HT2
HOPE CREEK-1	HC1
INDIAN POINT-2	IP2
INDIAN POINT-3	IP3
JAMES A FITZPATRICK	FZ1
KEWAUNEE	KW1
LA CROSSE (GENOA-2)	LC1
LASALLE COUNTY-1	LS1
LASALLE COUNTY-2	LS2
LIMERICK-1	LM1
LIMERICK-2	LM2
MAINE YANKEE	MY1
MCGUIRE-1	MC1
MCGUIRE-2	MC2
MILLSTONE-1	MS1
MILLSTONE-2	MS2
MILLSTONE-3	MS3
MONTICELLO	MO1
NINE MILE POINT-1	NM1
NINE MILE POINT-2	NM2
NORTH ANNA-1	NA1
NORTH ANNA-2	NA2
OCONEE-1	OC1
OCONEE-2	OC2
OCONEE-3	OC3
OYSTER CREEK	OY1
PALISADES	PA1
PALO VERDE-1	PV1
PALO VERDE-2	PV2
PALO VERDE-3	PV3
PEACH BOTTOM-2	PE2
PEACH BOTTOM-3	PE3
PERRY-1	PY1
PERRY-2	PY2
PILGRIM-1	PG1
POINT BEACH-1	PB1
POINT BEACH-2	PB2
PRAIRIE ISLAND-1	PI1
PRAIRIE ISLAND-2	PI2
QUAD CITIES-1	QC1
QUAD CITIES-2	QC2
RANCHO SECO-1	RS1
RIVER BEND-1	RB1
ROBINSON-2	RO2
SALEM-1	SA1
SALEM-2	SA2
SAN ONOFRE-1	SO1
SAN ONOFRE-2	SO2
SAN ONOFRE-3	SO3
SEABROOK-1	SB1
SEQUOYAH-1	SE1
SEQUOYAH-2	SE2
SHEARON HARRIS-1	HR1
SHOREHAM	SH1
SOUTH TEXAS PROJECT-1	ST1
SOUTH TEXAS PROJECT-2	ST2
ST. LUCIE-1	SL1
ST. LUCIE-2	SL2
SURRY-1	SU1
SURRY-2	SU2
SUSQUEHANNA-1	SQ1
SUSQUEHANNA-2	SQ2
THREE MILE ISLAND-1	TM1
THREE MILE ISLAND-2	TM2
TROJAN	TR1
TURKEY POINT-3	TP3
TURKEY POINT-4	TP4
V. C. SUMMER	VS1
VERMONT YANKEE	VY1
VOGTLE-1	VO1
VOGTLE-2	VO2
WATERFORD-3	WF3
WATTS BAR-1	WB1
WATTS BAR-2	WB2
WNP-2	WP2
WOLF CREEK	WC1
YANKEE-ROWE	YR1
ZION-1	ZN1
ZION-2	ZN2

APPENDIX I

COMPUTER POINT SELECTION

The main theme of the computer point selection process is to identify the minimum set of computer points, available on the fewest (preferably one) number of feeders from a site, which fully describe each of the parameters on the ERDS Parameter List.

When multiple computer points exist to describe a certain parameter, there is usually one point or a small subset of points which meet the following desirability criteria:

- For fluids systems (e.g., HPCI, Building Ventilation, Main Feedwater, etc.) the points representing the farthest location downstream in the system are most desirable. Examples:
  - If the ventilation system exhausts from all buildings in the power block converge and ascend up a single plant vent stack, then only the effluent process radiation monitors on the plant stack need be described under "gaseous effluent" versus describing the individual effluent monitors which may exist for each of the exhaust lines which converge.
  - If an injection or feedwater system has a set of points available which include flows measured at the pump discharges, at a combined header and at the point in the system just prior to injection into the loops or steam generators, then the points which should be selected as potential ERDS feeds are the furthest downstream points (flow measured just prior to injection into loops or steam generators).
- Computer points which have undergone the maximum amount of range checking and other data point validation schemes should be selected. We are aware that many utilities are in the process of upgrading computer system validation techniques and that what exists now may be replaced at some future date.
- Computer points representing the widest expected range of the parameter should be selected. For example: If there is a choice of computer points for "Containment Pressure" with one representing the range -5 to +5 PSIG and another representing the range -5 to +100 PSIG, the wide-range -5 to +100 PSIG computer point should be selected; even though its accuracy may not be as great near the normally expected pressure of -1 to +1 PSIG.
- The point composed of the maximum number of inputs should be used. The desirable point may be composed (processed) within the feeder computer or may be composed by a separate microprocessor outside the feeder as in the case of PWR Reactor Vessel Level Indication (RVLIS), Subcooling Margin Monitors (SMM) and meteorological tower systems. The philosophy of selecting the most composed points should not be applied in the case of parameters associated with PWR coolant loops (e.g., T-hot, T-cold, S/G Pressure, S/G Level, Main Feedwater Flow, etc.) to the extent of selecting points such as "Average T-hot", because loop-specific parameters are preferable for use in coolant-loop-specific accidents such as Steam Generator Tube Breaks. Composed points such as "Average T-hot Loop 1", "Average T-hot Loop 2 , etc., should be selected.

APPENDIX J

ERDS QUESTIONS AND ANSWERS

1. Will the implementation of the ERDS affect the NRC response role or the way that role is fulfilled?

No. The NRC response role was defined and approved by the Commission and would not change due to the ERDS. Current response activities, including discussions with the licensee, will be done more quickly and efficiently due to ERDS implementation but would not materially change.

2. What is the current program schedule?

The NRC ERDS is scheduled to be delivered to the Operations Center in April, 1990. As of that date it is anticipated that ERDS will have been implemented at five reactor units. There are currently over forty reactor units committed to ERDS implementation. Implementation at all units is scheduled to be completed by the end of 1992.

3. Will the implementation of the ERDS require significant equipment modification or addition by licensees?

The only equipment requirements are for the hardware that is needed to provide a data stream for each unit from the current licensee equipment that processes the requested data on site. For those licensees where no new hardware is required, the costs per reactor unit are estimated in the range of $20K to $50K. This estimate includes labor costs associated with software development, design change notice documentation, testing, and procedure development. Approximately 5 to 10 percent of the licensee's systems are running at close to 100 percent processing capacity in the post trip or incident environment, and approximately 10 to 15 percent of the licensee systems are hardware limited (e.g., no available output port for an ERDS connection). At the upper end of the cost spectrum, the ERDS feasibility study revealed that two plant sites would require additional computer equipment to provide the necessary ERDS feed. The hardware costs were estimated at $150K plus licensee staff time required to set up a custom system development effort with the appropriate contractor.

4. Will the ERDS be considered safety grade or require redundant equipment?

No. The ERDS feed will be as reliable as the current licensee equipment providing data to the licensee's own TSC and EOF. The addition of new plant instrumentation or computer data points to provide ERDS data will not be required.

5. Will the current data list be expanded?

No. The issue has been well studied since the Nuclear Data Link was originally proposed after TMI. The development of the data list followed our determination of our role in an emergency and provides the information we need to perform that role. The data list is intended to be generic in nature. There is a limited amount of space in each unit's data base to accommodate plant specific data points which are not on the data list, but would be useful in assessing plant conditions. Experience from the implementation program to date has indicated that there are parameters that licensees would like to send as a part of the ERDS data stream. Licensee recommendations for additional data points will be considered for addition to individual unit data bases. Needed data not transmitted over ERDS will still be passed over the ENS.

6. **Must the ERDS be used to transmit drill data?**

That is not a design requirement. For those system configurations which only allow the transmission of real data, no modification will be expected. However, if the licensee system is used for drills and can provide the transmission of the drill data, we would like to use the capability for our drill participation.

7. **Will the ERDS be an LCO of Tech Spec item?**

No.

8. **How soon does the NRC expect the system to be initiated after an Alert declaration?**

The ERDS should be initiated when the licensee notifies the NRC of the declaration of an Alert or higher emergency classification.

9. **Will the transmission of data point values for times prior to the time of the ERDS activation be required?**

No. Only the data values from the time of the link initiation will be required over the ERDS. Information on initiating conditions and plant status will be provided over the verbal communication line as necessary.

10. **Once the ERDS is implemented, will continuous manning of the ENS (Red Phone) still be required?**

Yes. The ERDS will not eliminate the need for verbal transmission of information such as licensee actions, recommended protective actions, and supplemental event specific data not provided by ERDS. Emphasis will be given to producing no new impact on Control Room personnel due to the transmission of data over the ERDS.

11. **Will the ERDS data be provided to State authorities?**

Although the NRC is not soliciting or recommending State participation in the ERDS program, one provision of the system design is user ports for States within the 10 mile plume exposure EPZ. This provision was made to reduce the likelihood of different data being provided to the NRC and a State because of differing data sets where the State has decided to collect data. This provision is not expected to affect States that already have a data collection system. If a State expresses a desire to participate in the ERDS program, the NRC will provide ERDS data to that State under a specific Memorandum of Understanding. The purpose of this Memorandum of Understanding would be to specify communication protocols for clarification of ERDS data and data security requirements. The NRC would provide those States with contractor developed software and make one output port available to the State from the NRC Operations Center. The States would have to obtain compatible PC hardware and licensed software used in the ERDS system to receive data. The specifications for a State ERDS workstation is attached at the end of the Questions and

Answers for your information. These provisions will ensure that all parties involved are using the same data base for their analysis. Any request made by a state to set up the capability to receive ERDS data will be discussed with the utility.

12. Will the NRC require a periodic test of the ERDS, and if so how frequently?

The NRC does expect that periodic testing will be required to ensure system operability. Currently we expect that testing will be done quarterly. Should system reliability permit, the frequency of testing may be reduced. Testing of a State link portion of the system will be done with the NRC. Therefore, no licensee participation will be required for this test.

13. Will participation in the ERDS program remain voluntary?

The NRC has initiated rulemaking to require the implementation of ERDS at all nuclear power plants. It is anticipated that the provisions of the proposed rule would be the same as those of the voluntary implementation program currently in effect.

14. What will be the boundary of system maintenance responsibility?

The NRC will be responsible for maintenance of all parts of the ERDS system installed starting at the input port of the first ERDS-specific piece of hardware (e.g., modem for single feeder plants and multiplexer for multi-feeder plants.)

WORKSTATION DESCRIPTION FOR THE STATE'S INTERFACE TO THE NRC'S EMERGENCY RESPONSE DATA SYSTEM

Hardware

1. Compaq 386/25 with:

   40 MByte Hard Disk (Minimum)
   640K Memory (Minimum)
   5 1/4 inch and/or 3 1/2 inch floppy drive
   EGA/VGA Card (640 x 480 Resolution)
   Serial Communications Port
   Parallel Printer Port

2. EGA Monitor (640 X 480 Resolution)
3. Mouse or Trackball with Card and Windows Driver
4. Desk Top Printer
5. Codex 2240 Modem

Software

6. Microsoft Windows 286
7. Winterm 8820
8. DOS 3.3

NOTE: Items 2, 3, 5, 6, 7, and 8 are required components. A functional equivalent for item 1 is acceptable as long as the required items are supported. Item 4 is optional.

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NRCM 1102,
3201, 3202

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13. ABSTRACT (200 words or less)

The U.S. Nuclear Regulatory Commission has begun implementation of the Emergency Response Data System (ERDS) to upgrade its ability to acquire data from nuclear power plants in the event of an emergency at the plant. ERDS provides a direct real-time transfer of data from licensee plant computers to the NRC Operations Center. The system has been designed to be activated by the licensee during an emergency which has been classified at an ALERT or higher level. The NRC portion of ERDS will receive the data stream, sort and file the data. The users will include the NRC Operations Center, the NRC Regional Office of the affected plant, and if requested, the States which are within the ten mile EPZ of the site. The currently installed Emergency Notification System will be used to supplement ERDS data.

This report provides the minimum guidance for implementation of ERDS at licensee sites. It is intended to be used for planning implementation under the current voluntary program as well for providing the minimum standards for implementing the proposed ERDS rule.

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS
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NRC Operations Center
emergency
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Regulatory Analysis

Regulatory Analysis of the Proposed Rule Concerning the Emergency Response Data System

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(continued)

Executive SUMMARY

The NRC has issued Generic Letter 89-15, which requests voluntary cooperation from each nuclear power reactor licensee in implementing an Emergency Response Data System (ERDS) program on each of its operational nuclear power units. The ERDS program will supplement the currently installed voice-only Emergency Notification System (ENS) by providing the NRC Operations Center with a timely and accurate limited set of parameters from the installed on-site computer systems in the event of a site emergency.

The NRC recognizes the importance of the ERDS in enhancing its ability to fulfill its role in the event of an emergency and has placed a high priority on the implementation of the ERDS program by all operational nuclear power units. A proposed rule that would amend 10 CFR Part 50 to ensure compliance with the ERDS program has been developed. The rule would ensure participation in the ERDS program and would set a definite schedule for its implementation.

This report is a regulatory analysis of the ERDS Rule. Two alternatives were examined:

- Take no action at this time and rely on the Generic Letter to promote licensee voluntary participation in the ERDS program;
- Adopt the proposed rule.

The first alternative relies on voluntary participation. Based on industry responses to date, the NRC is unable to confidently predict that more than 50% of the licensed nuclear power plants will participate. Additionally, there is no firm schedule of implementation nor are there provisions for NRC verifications. The absence of a regulatory basis for the ERDS is less than ideal.

The rule would ensure that the ERDS program is implemented in a timely fashion on all* operational nuclear reactor units with minimum impact on the industry. Since many of the elements of the rule are currently implemented by the nuclear power industry and none of the elements require advancements of the state of the art in a technical or personnel oriented discipline, there are no barriers to its timely implementation. Additionally, codifying the ERDS requirements would result in a net positive benefit to both the licensees and the NRC during an emergency.

The principal benefit that will accrue from the rule is the increased assurance that the NRC will have the means for timely acquisition, review, and evaluation of critical parameters at any operating reactor in distress. This, in turn, would improve the NRC's understanding of an event and allow it to more effectively perform its role, including monitoring the licensee to ensure that appropriate recommendations are made for offsite protective actions, supporting the licensee with technical analysis and logistic support, supporting offsite authorities, keeping other Federal agencies and entities informed of the status of the event, and keeping the media informed of NRC's knowledge of the status of the event. Thus the adoption of the rule would result in an unquantifiable but significant increase in the level of protection provided to the health and safety of the public. Those licensees who have not volunteered to participate in the ERDS program will benefit in that during an emergency licensee resources now required to collect and transmit data and information via the existing Emergency Notification System (ENS) would be available to be directed to managing the emergency.

Based on the findings of this report, the staff recommended adoption of the rule.

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*Throughout the regulatory analysis the staff indicates that adoption of the rule will result in ERDS programs being in-place at all nuclear power reactors. Technically, the ERDS rulemaking will impact all nuclear power reactor facilities except Big Rock Point and those that are permanently or indefinitely closed.

1. STATEMENT OF THE PROBLEM

The United States Nuclear Regulatory Commission (NRC) has defined its primary role during an emergency at a licensed nuclear power facility as that of monitoring the licensee to ensure that appropriate recommendations are made with respect to offsite corrective action (Ref. 1). Currently, the licensee's required response to the NRC during an emergency is the activation of an open communication channel, usually the Emergency Notification System (ENS). Although the system is deemed acceptable by the NRC during emergency exercises, some problems have come to light. At times the voice-only reporting has required excessive time for routine transmission of data and for verification or correction of possibly questionable data. Errors have been attributed to transcribing and interpreting voice-transmitted data, and the frequency of update intervals at times has been irregular. To overcome these problems and supplement the ENS, the NRC initiated the implementation of an Emergency Response Data System (ERDS) with a group of utilities receptive to the concept. This initial program was discussed at several Commission meetings and resulted in the issuance of Generic Letter 89-15, dated August 21, 1989, (Ref. 2) requesting voluntary cooperation from the licensees in implementing the ERDS. Tests of the ERDS concept indicate that the ERDS will be highly valuable during an accident, since it will provide the NRC with more timely and accurate data on the condition of the reactor plant. The information made available through the ERDS will allow the NRC to improve its interaction with all parties concerned to minimize the adverse consequences of the accident. Therefore, to ensure 100% industry participation, the NRC is considering a rulemaking that would require all nuclear power reactor licensees to install an ERDS for each unit. The following sections synopsize the rationale behind the need for an ERDS rule; they also review certain attributes of the proposed rule that must be considered.

1.1 Need for Emergency Response Data System Rule

As a result of the March 28, 1979 accident at Three Mile Island Unit 2, the NRC and others realized that the NRC needed to improve its ability to acquire reliable data on plant conditions during an emergency. This led to the conceptual design of the Nuclear Data Link (NDL) by Sandia National Laboratory. The NDL was designed to be an on-site fully automated data acquisition system based on an extensive set of plant sensors in constant communication with the NRC Operations Center. The NDL would have allowed the NRC to determine the status of any licensed power plant at any time. However, in 1984 the United States Congress rejected the NRC's budget request for implementing the NDL concept. The NRC continued to examine the data collection issue during 1985 and 1986, conducting site surveys of existing hardware and software that could support off-site data transmission (Ref. 3). Congress also remained interested in the concept of obtaining data from the power plants for use during an emergency and drafted proposed legislation: HR 5192 in 1986 and HR 1570 in 1987. In general, these bills prescribed features for a data collection system that differed from the NRC's concepts. After congressional response and testimony by the NRC and others, these bills were tabled. In March of 1987, a report entitled "Emergency Response Data System Requirements Analysis Report" (Ref. 4) was prepared under contract to the NRC by Phoenix Associates and COMEX Corporation. This report forms the basis of the current ERDS concept, which uses available on-site data acquisition systems to transmit critical data when activated by the licensee during an emergency.

The NRC recognizes the value of more reliable data transmission in the event of an emergency and has placed a high priority on the implementation of the ERDS at all nuclear power units. In January of 1988, the NRC awarded a contract for the procurement of ERDS hardware and software, as well as for data transmission units to be installed at each nuclear power plant unit to tie into the NRC's Operations Center. ERDS hardware integration and software development are in progress with delivery of an operational system at the NRC Operations Center expected in early 1990. The current ERDS, however, relies on voluntary cooperation by the licensees for the implementation of their on-site portions. Generic Letter 89-15 urges voluntary participation in the ERDS program by all licensees. A voluntary program, however, would not ensure 100% participation. The proposed rule would require all nuclear power reactor licensees to establish and maintain an ERDS. The rule, when adopted, would thus provide a regulatory basis for the ERDS and ensure participation by all licensees.

1.2 Scope of Consideration

The proposed ERDS rule would require each licensee to establish and maintain an ERDS program designed to supplement the currently installed voice-only ENS for transmission of selected critical parameters to the NRC Operations Center upon declaration of an emergency condition. This proposed rule would require 100% participation of all nuclear plants, and would provide increased assurance that the NRC will have access to critical information during a reactor accident, regardless of which unit might be affected. The main features of the licensee's ERDS program would include:

- The **hardware link**, which will connect the on-site data acquisition system with the data transmission unit supplied by the NRC;
- The **software link**, which will extract and format the requisite data to be transmitted to the NRC Operations Center;

These two elements of the licensee's ERDS program are not separate, stand-alone elements. Rather, they are mutually reinforcing segments of the overall program. Together they will provide the increased assurance that the NRC will have accurate real-time data with which to adequately perform its role during an alert or higher emergency.

2. OBJECTIVES OF THE RULE

The primary objective of this rulemaking is to provide increased assurance that a reliable, effective communication system that will allow the NRC to monitor available critical parameters during an emergency is in place at <u>all</u> operating power reactors. With more timely and more accurate information, the NRC will be better able to fulfill its emergency response mission. Tests of the ERDS concept have demonstrated great value in using electronic data transmission. In tests performed to date, NRC response teams functioned more efficiently and their assessments were more timely. They noted major improvements in the ability to focus on significant factors and to predict the course of events (Ref. 1). Other aspects of NRC's role during an emergency at a licensed nuclear power reactor that will be enhanced by the ERDS include supporting the licensee with technical analysis and logistic support, supporting offsite authorities, keeping other Federal agencies and entities informed, and keeping the media informed.

A secondary objective of the ERDS rulemaking is to enhance the licensee's efficiency and effectiveness during emergency operations by redirecting the information reporting responsibilities of plant operating personnel. Currently, operating personnel in many cases must manually collect reactor parametric data and transmit it via voice communication to the NRC Operations Center. Using the ERDS to perform these functions automatically would allow the time spent collecting and transmitting this data to be used more effectively to focus on more substantive information on the management and operational aspects of the emergency.

Another objective of this rulemaking is to expedite the implementation of the ERDS program throughout the nuclear power industry. Licensees will be asked to supply an output port for NRC's use. Licensees will also provide the software necessary to assemble data to be transmitted. Data may be presented in either American Standard Code for Information Interchange (ASCII) or Extended Binary-Coded Decimal Interchange Code (EBCDIC). The parameters to be reported will not be excessive (approximately 65 to 100 points), and licensees will not be required to monitor more points than those currently residing on the plant computers. Both the rule and Generic Letter 89-15 indicate that the NRC will try to accommodate each licensee's system in order to minimize the burden to the licensees. Specifically, the objectives of the rule are to:

- Provide a reliable, effective communication system that will allow the NRC to monitor selected critical parameters during an emergency at an operating power reactor;
- Ensure the timely implementation of the ERDS at <u>all</u> licensed nuclear power reactors; and
- Codify the minimum requirements for the ERDS licensee program.

3. ALTERNATIVES TO THE RULE

Two possible alternatives were examined as part of this regul tory analysis:

- Take no action at this time and rely on the Generic Letter to promote licensees' voluntary participation in the ERDS program;
- Adopt the proposed rule.

Although only two alternatives are considered in this regulatory analysis, a number of other methods of transmitting data from a nuclear power reactor were briefly reviewed. These options included the use of trained radio communications operators on the current Emergency Notification System (ENS), telefax, and manually entered data on a dedicated on-site computer that would communicate with the NRC's Operations Center. These methods were disregarded since they did not meet the requirements for reliability and timeliness. Discussions of the options considered are provided in this section.

3.1 Option 1 - No Action by the NRC

The first option dictates no action at this time, relying on voluntary participation in the ERDS program by all utilities. Based on industr responses to date, the NRC is unable to confidently predict that more than 50 percent of the licensed nuclear power plants will participate. In addition, there would be no deadline for compliance. In short, the absence of a regulatory basis for the ERDS program may result in an incomplete system that would cover only a fraction of the reactor population. This approach does not ensure that the NRC will have the information it needs to respond in an effective manner to an actual reactor emergency.

3.2 Option 2 - Adopt the Rule

The second option is to adopt the proposed rule, the major requirements of which are summarized in this section. As previously stated, the NRC is considering a general rule to amend 10 CFR Part 50. It would require all licensees of nuclear power reactors to establish and maintain a hardware link, a software link, and a configuration control program for the ERDS. The proposed rule would also address such related items as when the ERDS should be activated and periodic testing requirements.

Since many nuclear power licensees already have most of the necessary hardware and software, any changes necessary to meet the requirements of the proposed rule are not expected to be substantive. Timely and complete implementation of the ERDS program would allow the NRC to more efficiently fulfill its responsibilities in the event of an emergency at a nuclear power reactor. The remainder of this section briefly summarizes the requirements of a licensee ERDS program that would be established by the proposed rule. With few exceptions, all nuclear power reactor licensees would be subject to the requirements of the final regulation if promulgated.

3.2.1 <u>Responsibility</u>

The licensees would be responsible for implementing an ERDS program and ensuring that the elements of the proposed rule are followed. The NRC would verify compliance with the rule and incorporate the ERDS in its emergency response planning.

3.2.2 <u>General Requirements For a Licensee ERDS Program</u>

Each licensee would implement an ERDS program at each nuclear power unit and would provide the personnel to implement the ERDS. Each licensee would also provide the software necessary to format the parameters for NRC use and an output port on an appropriate machine for NRC's use. Licensee personnel would be available for periodic testing of the ERDS and would be responsible for notifying the NRC of any changes made in the licensee's hardware, software, or monitoring program.

3.2.3 <u>ERDS Parameters</u>

The ERDS would transmit accurate real-time data on plant conditions in four areas:

- The reactor core and coolant system conditions (to assess the extent or likelihood of core damage);
- The conditions inside the containment building (to assess the likelihood of its failure);

- The radioactivity release rates (to assess the immediacy and degree of public danger);

- The data from the plant's meteorological tower (to assess the distribution of potential releases or actual impact on the public).

The data related to these conditions would be provided by the licensee's computer system to an NRC-supplied data transmission unit that will transmit the data to the NRC Operations Center. The licensee would activate the on-site ERDS and, at the same time, notify the NRC via the ENS of an emergency.

Table 3.1 depicts the required parameters data to be transmitted for a pressurized water reactor and Table 3.2 depicts those for a boiling water reactor. Should a licensee's present computer system not monitor some of the specified parameters, data for those parameters would not need to be part of the ERDS program. In such cases, the data values of those parameters, if available, would be transmitted over the ENS. However, if the licensee adds to its computer system the capability of monitoring such parameters, the NRC would expect to receive the data through the ERDS.

TABLE 3.1 PRESSURIZED WATER REACTOR PARAMETER LIST

Primary Coolant System	Pressure Temperatures -- Hot leg Temperatures -- Cold Leg Temperatures -- Core Exit Thermocouples Subcooling Margin Pressurizer Level Reactor Cooling System (RCS) Charging/Makeup Flow Reactor Vessel Level (when available) Reactor Coolant Flow Reactor Power
Secondary Coolant System	Steam Generator Levels Steam Generator Pressures Main Feedwater Flows Auxiliary/Emergency Feedwater Flows
Safety Injection	High-Pressure Safety Injection (HPSI) Flows Low-Pressure Safety Injection (LPSI) Flows Safety Injection Flows (Westinghouse) Borated Water Storage Tank Level
Containment	Containment Pressure Containment Temperatures Hydrogen Concentration Containment Sump Levels
Radiation Monitoring System	Reactor Coolant Radioactivity Containment Radiation Level Condenser Air Removal Radiation Level Effluent Radiation Monitors Process Radiation Monitor Levels
Meteorological	Wind Speed Wind Direction Atmospheric Stability

Table 3.2. BOILING WATER REACTOR PARAMETER LIST

Primary Coolant System	Reactor Pressure Reactor Vessel Level Feedwater Flow Reactor Power
Safety Injection	Reactor Core Isolation Cooling (RCIC) Flow High-Pressure Coolant Injection (HPCI)/High-Pressure Core Spray (HPCS) Flow Core Spray Flow Low-Pressure Coolant Injection (LPCI) Flow Condensate Storage Tank Level
Containment	Drywell Pressure Drywell Temperatures Hydrogen and Oxygen Concentration Drywell Sump Levels Suppression Pool Temperature Suppression Pool Level
Radiation Monitoring System	Reactor Coolant Radioactivity Level Primary Containment Radiation Level Condenser Off-Gas Radiation Level Effluent Radiation Monitor Process Radiation Levels
Meteorological	Wind Speed Wind Direction Atmospheric Stability

3.2.4 Hardware Link

Each licensee would provide the hardware necessary to interface with the NRC-supplied communications link. In most cases, this can be accomplished with already-installed equipment. The NRC would supply one (for a single unit site) or more (for a multiple unit site) transmitting device(s) which would be configured to accept the ready-to-send/clear-to-send (RTS/CTS) control signal of RS-232C interface standard "handshaking protocol" (i.e., initiating transmitted signal is linked and acknowledged by the receiving end). In the case of sites having the requisite ERDS parameters located on multiple computers for a single reactor unit, the NRC would furnish a multiplexer to serve the multi-feeder reactor unit. Software would be supplied by the licensee to work with the multiplexer.

3.2.5 Software Link

Each licensee would provide the necessary software to select the required parameter data for transmission. The ERDS will accept data in either the American Standard Code for Information Interchange (ASCII) or Extended Binary-Coded Decimal Interchange Code (EBCDIC). All link-control messages would be sent in ASCII. The data stream structure would comprise three sections: a fixed-length header, a set of self-identifying parametric values, and a fixed-length trailer. Each data point packet would contain 3 fields: an identifier, the value, and a quality tag.

3.2.6 Configuration Control Requirements

Each licensee would implement an ERDS configuration control program by which the NRC will be informed of any changes to the ERDS on-site hardware or software.

3.2.7 Periodic Testing

Nuclear power plant licensees would be required to conduct a test of the ERDS program periodically. Initial testing would be done on an quarterly basis. Should experience indicate a high degree of reliability with the system operation, the frequency of testing may be reduced. The testing would consist of transmitting to the NRC the equivalent of approximately 20 minutes of data using the established ERDS "handshake protocol." In addition to the quarterly schedule, testing would be required after major system modifications by the licensee.

4. Consequences

This section addresses the cost and the benefits associated with the identified options. Two alternatives were identified: (1) maintain the ERDS on a voluntary basis and (2) issue a rule. The determination of the consequences associated with the proposed rule was based on NRC technical reports and communications and discussions with commercial companies. Conservative engineering judgment was used where data could not be expeditiously obtained.

Adoption of the proposed rule would ensure 100% participation in the ERDS program. This increased participation would provide a better information base to the NRC. This, in turn, would help to ensure that NRC expertise would be available to assist in minimizing consequences to the public in case of an accident, thereby effectively increasing the protection to the public health and safety. These benefits could be substantial, whereas costs to utilities are minimal on an annual, per reactor basis, as discussed below. Moreover, substantial cost savings in averted adverse consequences are probable.

The incremental costs associated with adopting the proposed rule are low primarily because the development cost of the ERDS as well as costs of procuring the necessary communication terminals at the nuclear plant site and the ERDS computer system at the NRC Operations Center have already been incurred by the NRC. These already-borne costs are not considered to be incremental costs attributable to the proposed rule.

Implementation of the proposed rule would require all licensees to participate in the ERDS program. For most of those licensees who have voluntarily complied, it would cause minimal impact. There would be an impact for those who have not chosen to comply voluntarily. However, this codification of the ERDS requirements and its application to the entire reactor population would help to ensure an effective and reliable basis for the NRC to monitor and act in emergencies.

For the sake of thoroughness and completeness, the typical topics addressed in the preparation of a regulatory impact analysis are addressed in the following sections.

4.1 No Action by the NRC - Maintain the Voluntary ERDS Program

The current ERDS program assumes that the licensees will implement the on-site aspects of the program on a voluntary basis. As such, this option presents essentially a continuation of the status quo, which is comparable to no action. Incremental costs and benefits are not normally defined for a no-action decision. On the other hand, in the interest of comparing similar situations, it can be assumed that the costs and benefits associated with voluntary participation would be proportional to those assigned to the rule using the industry participation rate. To some extent, marginally higher, NRC costs could result from voluntary participation because the variety of hardware and software used by the licensees could be more burdensome on the Commission and because of the complications posed by the open-ended schedule attendant to this option. The main weakness of this option is that there is no assurance that all of the reactor units will participate. Thus the public in the vicinity of those units that are not part of the ERDS would be at some higher incremental risk since NRC's oversight role in the case of an emergency at these plants is not likely to be as effective as it would be for plants with ERDS.

4.2 Proposed Rule

The benefits derived from implementing the proposed ERDS rule directly address problems that arise from the no-action option and other briefly considered alternatives. Complete voluntary implementation could be complicated by variable interfaces at some licensees' facilities and lack of an enforceable timetable; the proposed rule would require standard interfaces at all licensees by a specific date. Other data collection systems considered required much new hardware and software and additional manpower from both licensees and the Commission; the proposed rule would use already-installed hardware, relatively minor software revisions, and minimal additional manpower. In short, implementation of the proposed ERDS rule would provide the greatest benefit for the least cost. The following sections present greater detail regarding specific benefits.

4.2.1 Benefits

The key objective of the proposed ERDS rule is to achieve a high degree of assurance that accurate near real-time data are made available to the NRC to use during emergency response. The NRC's primary role in an emergency was defined in the 1987 revision to NUREG-0728[1] as monitoring the licensee to ensure that appropriate recommendations are being made with respect to off-site protective actions. In addition, the NRC's role includes supporting the licensee with technical analysis and logistic support, supporting offsite authorities, keeping other Federal agencies and entities informed of the status of the incident, and keeping the media informed of the NRC's knowledge of the status of the incident. Currently, these NRC functions rely on data transmitted verbally through the Emergency Notification System (ENS). Testing of the ERDS has demonstrated improvements in reliability of the data transmitted. In addition, the time to acquire and transmit the data is faster, as is the time required for verification and validation of the data.

The implementation of the ERDS as a supplement to the ENS not only would improve the accuracy and timeliness of data transmission but also would enable the licensee to better use its time and resources to effectively and efficiently deal with the emergency. The combination of better and more timely assessments of licensee actions by the NRC, and the focusing of licensee resources to better deal with the emergency at hand together will reduce the overall risk to the public health and safety from an emergency.

While estimating the reduction in off-site radiation exposure to the general public attributable to the implementation of an ERDS is beyond the scope of this analysis, it is clear from the testing to date that implementation of the proposed ERDS rule would better provide the NRC with the information needed to fulfill its major role of monitoring an emergency to ensure that the licensee has recommended the appropriate corrective actions.

4.2.2 Occupational Radiation Exposure

The requirements of the proposed rule would have no effect on routine occupational radiation exposure. Therefore, no incremental impacts in this category, either positive or negative, are associated with this action.

4.2.3 Industry Costs and Savings

The major costs of implementing the ERDS program have been borne by the NRC in that the NRC has already established the ERDS computer system at the NRC Operations Center and has procured the necessary on-site communication hardware. Additionally, costs have already been incurred by the licensees voluntarily participating in the ERDS program.

Estimates of the cost for implementing and maintaining an ERDS program in accordance with the proposed rule were based on the following assumptions and bases:

- The ERDS program has a 30-year duration for each unit;
- A 5% discount rate was used in the present-value base calculations;
- A 10% discount rate was used in the present-value sensitivity calculations;
- There would be no cost impact to the NRC as a result of providing the on-site communications links since the on-site communication links have already been procured by the NRC and are not incremental costs;
- All costs are expressed in 1990 constant dollars;
- The ERDS actions are implemented at all plants in 1992.

Because of the diversity in the details of implementing the ERDS program at each reactor unit, a base set of characteristics for a typical ERDS program for a generic unit was established. This base set of characteristics included the following attributes:

- The average cost of ERDS-related hardware needed to link the on-site data acquisition units with the NRC-supplied communications link is $25,000;
- The average level of effort needed to establish the ERDS program and to develop the requisite software to provide the necessary parameter data from the licensee's computer system is 12 staff weeks;
- Every 5 years, $5,000 will be spent in upgrading the ERDS-related hardware (because of equipment obsolescence or upgrades to the plant data acquisition system);
- Every 5 years, 4 staff weeks will be expended to modify the software to conform to the hardware upgrades;
- Periodic testing will entail 1 staff day of effort 4 times per year;
- One staff week of effort will be expended each year in maintaining the on-site ERDS configuration control program.

1992 was selected as the reference year the proposed rule would be implemented. All historical cost data were escalated to 1990 constant dollars using appropriate escalation factors. All future costs are presented in 1990 constant dollars and present-valued based on a 5% real discount rate. The discount rate should be interpreted as the rate of return on invested funds over and above the rate of inflation. Recurring costs such as those for hardware and software upgrades were placed at the appropriate recurring intervals and the costs brought back to the 1992 datum year using standard present-value calculation methods. A cost impact analysis using a 10% discount rate was performed to determine the sensitivity of the costs to the discount rate.

The following are the point estimates of the costs of the elements required to implement and maintain an ERDS program at a nuclear power unit. The point cost estimate is derived from the bases and assumptions previously delineated and represent the most probable costs for each element. For any individual reactor site, costs could vary significantly from those estimated here. For example, at selected facilities the initial hardware setup could be in excess of $100,000 because additional computer equipment will be needed to provide the necessary ERDS feed. The estimates developed here, the sum of which equals $153,000, apply to a single generic unit, are based on a 5% discount rate to capture the costs over 30 years, and are rounded to the nearest 1,000 dollars.

Licensee Point Cost Estimates

ERDS-related hardware	$ 25,000
Establish ERDS program	28,000
Periodic hardware upgrades	13,000
Periodic software upgrades	24,000
Periodic testing	28,000
Configuration control	35,000
Total (1990 dollars):	$153,000

This cost of implementing and maintaining an ERDS program, on an annualized basis, amounts to only about $10,000 per year per plant. Based on a 10% real discount rate, the comparable estimates for a single generic reactor are a total of $113,000 (30 year present worth), and an equivalent annual cost of $12,000. These are trivial amounts compared to a typical nuclear plant's annual non-fuel operation and maintenance (O&M) budget, which typically ranges from $50 to $100 million dollars per year.

The total cost impact for 118 reactors is thus estimated at approximately $18 million and $13.3 million for a 5% and 10% real discount rate, respectively. These estimates capture the total industry cost over a 30 year program duration. It should be stressed that these estimates include resources already incurred or committed to ERDS on the part of voluntary participants. To the extent these costs exist independent of the decision on this rule, they are not incremental costs. Recognizing that the ERDS requirements among voluntary participants are comparable to the requirements of the rule, and assuming 50% of the power reactor units voluntarily participate, the total industry cost on an incremental cost basis could be viewed as 50% of that cited above.

4.2.4 NRC Costs

NRC costs are incurred from several activities associated with the implementation and maintenance of a formal licensing basis for the ERDS. During implementation, it is assumed that the NRC will perform an initial review for each reactor. The initial costs that would be incurred by the NRC can be estimated as follows:

NRC Implementation Costs

NRC Activity Plants	Cost per Plant	Total Cost for 118 Plants
Initial Review of Licensee's Submittal	$ 1,720	$203,000
Total	$1,720	$203,000

The new rule would also require the NRC to perform certain annually recurring activities (e.g., periodic testing). The estimate per unit is based on three staff days per year to maintain ERDS configuration control and one staff day four times a year for periodic system testing. The equivalent annual NRC cost is estimated as follows:

NRC Recurring Costs

Annual NRC Activity	Equivalent Annual NRC Cost per Plant	Total Annual Cost for 118 Plants
ERDS Configuration Control	$1,030	$121,500
Periodic Testing	1,370	161,700
Total	$2,400	$283,200

Using a 5% discount rate and amortizing over 30 years, the 1992 present worth of all recurring NRC costs per plant is $37,000 in 1990 dollars. For all 118 reactors, the 1992 present worth of recurring NRC costs is simply the product of $37,000 per plant multiplied by 118 plants or $4.4 million. If a 10% discount rate is used, the estimate for recurring costs for each unit becomes $23,000 in 1990 dollars.

The total cost to the NRC can now be estimated by the summation of the one-time implementation costs and the present worth of recurring costs as indicated in thė following table.

Total NRC Cost

NRC Activity	Total NRC Cost per Reactor (1990 $)	Total NRC Cost for 118 Reactors (1990 $)
Implementation	$ 1,720	$ 203,000
Recurring *	37,000	4,366,000
Total	$ 38,720	$4,569,000

Note: * 30 year present worth

Therefore, the 1992 present worth of total initial and recurring NRC costs to implement the proposed rule is estimated to be approximately $4.6 million in 1990 dollars.

Here too it can be argued that NRC's total cost could be lower on an incremental cost basis if one takes credit for the voluntary participants. Under the voluntary program, the NRC is committed to periodic testing and configuration cont ɔl over the full life of the ERDS program completely independent of the rule. Thus, assuming 50% of the reactors voluntarily participate, the total NRC cost on an incremental cost basis could be slightly less than 50% of the costs cited above.

5. DECISION RATIONALE

5.1 Regulatory Analysis

The ERDS program supplements the currently installed voice-only ENS by providing the NRC Operations Center with a more timely and accurate set of values of a limited number of parameters from the installed onsite computer systems in the event of a site emergency. The NRC recognizes the importance of the ERDS in enhancing its ability to fulfill its role in the event of an alert or higher emergency and thereby enhancing the public health and safety. Since many of the elements of the proposed rule are currently implemented by the nuclear power industry and none of the elements require advancement of the state of the art in a technical or personnel-oriented discipline, there are no barriers to its timely implementation. Adoption of the proposed rule is estimated to cost approximately $150,000 per reactor or about $18 million for the entire industry based on a reactor population of 118. These estimates are based on a 30 year program life and a 5% real discount rate. Based on the findings of this report, the staff recommends adoption of the proposed rule as the best means to accomplish the goals of providing the NRC with improved accurate real-time data during reactor emergencies.

5.2 Environmental Impact: Categorical Exclusion

The NRC has determined that this proposed rule is the type of action described in categorical exclusion 10 CFR 51.22(c)(3). Therefore neither an environmental impact statement nor an environmental assessment has been prepared for this proposed rule.

6. Implementation

6.1 Schedules

6.1.1 Emergency Response Data System Rule Development

The proposed rule is scheduled to go to the Commission by the end of July 1990, with anticipated publication in the Federal Register approximately one month later. The final rule will become effective 30 days after the final rule is published in the Federal Register. This is expected to occur early in mid-1991. The schedule for licensee compliance and the anticipated date of complete implementation is contained in Section 6.1.2 below.

6.1.2 Emergency Response Data System Rule Implementation

The proposed rule will provide a public comment period of 75 days after its publication in the Federal Register. The final rule will require that each licensee develop and submit an ERDS program plan for review by the NRC within 75 days after the rule has been published in the Federal Register, and implement their program within 18 months after approval by the NRC. With this schedule, the ERDS program should be fully implemented by the winter of 1992 or spring of 1993.

6.1.3 Regulatory Guidance Development

Guidance on implementation of the ERDS rule is provided in NUREG-1394.

6.2 NRC Staff Actions

NRC's major ERDS hardware and software procurements have been completed; however, the preparation and review of the proposed rule and the review of the public comments, as well as the preparation of the final rule, the implementing NUREG and the regulatory analysis combined with the ambitious schedule for finalizing the rule, will require constant staff attention. Various NRC review and coordination tasks at both the working level and the senior management level will be required to finalize the rule.

7. References

1. NUREG-0728-R02, "NRC Incident Response Plan," June 1987, available from the NRC Public Document Room, 2120 L Street NW (lower level), Washington, DC.

2. Generic Letter 89-15, August 21, 1989, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

3. "Upgrading the NRC Operations Center's Emergency Data Acquisition Capability," NRC Staff Paper, December 21, 1984.

4. "Emergency Response Data System Requirements Analysis Report," March 1987, NRC-04-85-163.

Enclosure 6

Draft Congressional Letter

Draft Congressional Letter

Dear Chairman:

Enclosed for your information is a Federal Register notice for publication of a rule to require licensed nuclear power plants to participate in the Emergency Response Data System (ERDS). The rule would apply to all operating nuclear power reactor facilities except Big Rock Point and those that are permanently or indefinitely shut down. It is anticipated that during an emergency the ERDS will improve the NRC's capability to fulfill its protective and advisory role. Specifically, through the more timely and accurate acquisition of information on plant conditions available with the ERDS, the agency will be able to both effectively monitor the nuclear power reactor licensee and promptly provide appropriate recommendations with respect to offsite protective actions.

The proposed rule on this subject was published in the Federal Register on October 9, 1990 (55 FR 41095). The NRC received 31 letters of comment with over 110 separate comments from a citizens group, individuals, licensees, industry organizations, and State authorities. The NRC staff has identified 21 separate topics, which were responded to in the Federal Register notice.

Revisions, mainly clarifying and editorial, have been made in the final rule as a result of the comments received.

Sincerely,

Eric S. Beckjord, Director

Office of Nuclear Regulatory Research

Enclosure:

Federal Register notice
of Final Rule

cc: Ranking Minority Members

Enclosure 7

Draft Public Announcement

DRAFT

NRC AMENDS REGULATIONS TO REQUIRE EMERGENCY RESPONSE DATA SYSTEM

AT LICENSED NUCLEAR POWER PLANTS

The Nuclear Regulatory Commission is amending its regulations to require licensed nuclear power plants to participate in an Emergency Response Data System (ERDS). The rule would apply to all operating reactor power reactor facilities except Big Rock Point (which is exempt because the plant configuration does not permit collection of sufficient data to effectively participate in ERDS) and those that are permanently or indefinitely shut down.

The system would be used to provide the NRC, during an emergency, with reliable, near real-time data on the following selected plant conditions: reactor core and coolant system conditions to assess the extent or likelihood of damage to the nuclear fuel; conditions inside the containment structures to assess the likelihood and consequences of its failure; radioactivity release rate to assess the immediacy and degree of danger to the public; and meteorological data to assess the likely patterns of potential or actual radiological impact on the public.

The NRC needs this system to supplement the existing voice-only Emergency Notification System (ENS) to carry out its primary role in the event of a nuclear power plant emergency which is to monitor licensee actions to ensure that recommendations are made with respect to offsite protective measures. In addition, the NRC is expected to provide technical analysis and logistical support to the licensee; support offsite authorities (including confirmation of a licensee's recommendation to these authorities); keep other Federal agencies informed of the status of the emergency; keep the media informed of the NRC's knowledge of the status of the emergency; coordinate with other public affairs groups.

The voice-only ENS, which has been in place since shortly after the 1979 accident at the Three Mile Island nuclear power plant, has demonstrated that excessive amounts of time are needed for routine transmission of data and for verification or correction of questionable data. In addition, errors have been attributed to the transcription and interpretation of voice-transmitted data.

The rule would require utility licensees to provide the necessary computer software to assemble the data and output communication port for each reactor unit in its on-site computer system. The required data on the plant conditions would be transmitted to the NRC Operations Center (NRCOC) in Bethesda, Maryland, via NRC-provided communication link hardware. The system would be activated in the event of an alert, site area emergency or general emergency at a licensed nuclear power plant. Licensees would be required to have the system operable within 18 months of the effective date of this final rule or before initial escalation to full power, whichever comes later.

Under the ERDS voluntary program, States have begun to request information concerning access to ERDS to obtain data during an emergency. The NRC staff is developing a Memorandum of Understanding which would provide a mechanism for the individual States to have access to the ERDS.

DRAFT

In August 1989, the NRC staff requested the voluntary participation of the licensees in the ERDS program. Currently, about half of licensed nuclear power plants have volunteered to participate in its. Over ten reactor units already are capable of transmitting ERDS data to the NRCOC. This rule will ensure an expeditious and successful implementation of the ERDS program at all units.

The revisions to Part 50 of the NRC's regulations will become effective on (date).

DOCKET NUMBER PR
PROPOSED RULE PR 50
(55 FR 41095)

DOCKETED
USNRC

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OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

NUCLEAR REGULATORY COMMISSION

10 CFR Part 50

RIN 3150 - AD32

Emergency Response Data System

AGENCY: Nuclear Regulatory Commission.

ACTION: Proposed rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) proposes to amend 10 CFR Part 50 of its regulations to require licensees to participate in the Emergency Response Data System (ERDS) program and to set a definite schedule for its implementation. The ERDS is a direct electronic data link between computer data systems used by licensees and the NRC Operations Center. The ERDS would supplement the voice transmission over currently installed Emergency Notification System (ENS). The ERDS would provide the NRC Operations Center with timely and accurate values of a limited set of parameters that describe selected plant conditions. The parameter values would be taken directly from data systems existing on a licensee's onsite computer. The ERDS would be activated by a licensee during the declaration of an alert or higher emergency classification at a licensed nuclear power facility. The NRC's response role in the event of an emergency at a licensed nuclear facility is primarily to monitor the licensee to ensure that appropriate recommendations are made by the licensee regarding off-site protective actions. The proposed rule is needed to

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improve the NRC's capability to fulfill its response role during an emergency by better assuring that it will receive accurate and timely information on plant conditions. This action will also allow the licensee to more effectively and efficiently utilize its time and resources in collecting and transferring data to the NRC. The proposed requirement would apply to all operating nuclear power reactor facilities except Big Rock Point and those that are permanently or indefinitely shut down. However, units shut down for maintenance, or authorized only for fuel loading and low power operations are required to report under ERDS. Big Rock Point is exempt because the configuration of the facility is such that the number of parameters available are not sufficient for effective participation in the ERDS program.

DATES: Comment period expires [75 days after date of publication in the Federal Register]. Comments received after this date will be considered if it is practical to do so, but assurance of consideration cannot be given except as to comments received on or before this date.

ADDRESSES: Mail written comments to: the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch.

Deliver comments to: 11555 Rockville Pike, Rockville, MD, between 7:45 a.m. and 4:15 p.m. on Federal workdays.

Copies of regulatory analysis, the environmental assessment and finding of no significant impact, the supporting statement submitted to OMB, and comments received may be examined at: The NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC.

FOR FURTHER INFORMATION CONTACT: M. L. Au, P.E., Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 492-3749.

SUPPLEMENTARY INFORMATION:

Background

As a result of the accident at Three Mile Island, Unit 2, on March 28, 1979, the Nuclear Regulatory Commission (NRC) and others recognized a need to substantially improve the NRC's abilitv to acquire accurate and timely data on plant conditions during emergencies. Before designing a system to accomplish this task, the NRC addressed several background issues dealing with its role during an accident, any changes necessary to enhance the response role to nuclear emergencies, and the information needed to support this role.

The NRC's role in the event of an emergency is primarily to monitor the licensee to ensure that appropriate recommendations are made with respect to offsite protective actions. Other aspects of the NRC's role include providing the licensee with technical analysis and logistic support, supporting offsite authorities (including confirming the licensee's recommendations to offsite authorities), keeping other Federal agencies and entities informed of the status of the incident, keeping the media informed of the NRC's knowledge of the status of the incident, and coordinating with other public affairs groups. Detailed study has determined that the Commission's statutory authority provides a sufficient basis for carrying out this defined emergency response role.

To fulfill this emergency response role, the NRC requires reliable real-time (actual time in which a process takes place) data on four types of selected plant conditions. These conditions are:

(1) Core and coolant system conditions -- needed to assess the extent or likelihood of core damage;

(2) Conditions inside the containment building -- needed to assess the likelihood and consequence of its failure;

(3) Radioactivity release rates -- needed to assess the immediacy and degree of public danger; and

(4) Data from the plant's meteorological tower -- needed to assess the likely patterns of potential or actual impact on the public.

Site surveys, conducted by the NRC in 1986, have shown that data relevant to these conditions are maintained in the plant computer systems by a majority of the licensees. Currently during an emergency, data on these conditions is transmitted to the NRC Operations Center by the licensee through the Emergency Notification System (ENS) via voice communication by telephone.

In SECY-84-481, "Upgrading the NRC Operations Center's Emergency Data Acquisition Capability," dated December 26, 1984, it was noted that experience with the ENS voice-only emergency communications link currently addressed in 10 CFR 50.72(a) demonstrated that excessive amounts of time are needed for routine transmission of data and for verification or correction of data that appears questionable. Errors were also attributed to transcribing and interpreting voice-transmitted data. This resulted in the NRC exploring improved methods to receive accurate and timely information it requires to perform its role during an alert or higher emergency.

After evaluating several options, the NRC selected the Emergency Response Data System (ERDS) as the most appropriate option to supplement the ENS. The staff conducted prototype ERDS testing with Duke Power and Commonwealth Edison reactor units. For example, data was transmitted and beneficially used via an ERDS prototype during the Zion Full Federal Exercise in June 1987. These tests demonstrated that there was great value in using electronic data transmission for obtaining a limited set of reliable, time tagged data. With this better and more timely data, the NRC response team functioned more efficiently and their assessments were more timely. Major improvements in the ability to focus on significant factors and to predict the course of events were noted. The questions directed from the NRC Operations Center to the licensee were focused on the overall event status and corrective actions being considered, rather than simple data requests, thereby reducing the volume of voice communications.

The NRC decided to implement the ERDS initially on a voluntary basis through the issuance of a generic letter while at the same time developing a rulemaking. On August 21, 1989, the NRC issued Generic Letter 89-15 to request the voluntary cooperation of each nuclear power reactor licensee in implementing an ERDS program at each of its operational nuclear power units. However, to date only about half of the operating nuclear power units have volunteered to participate in ERDS. The NRC recognizes the importance of the ERDS in enhancing its ability to fulfill its role in the event of an emergency and has placed a high priority on the implementation of the ERDS program by all operational nuclear power units. The staff has, therefore, developed the proposed rule that would amend Part 50 to require participation in the ERDS program and to set a definite schedule for its implementation.

Discussion

The ERDS would supplement the currently installed voice transmission ENS. The system will provide the NRC Operations Center with a timely and accurate limited set of parameters from the installed onsite computer systems in the event of an emergency at a nuclear power plant. Implementation of the ERDS would require each licensee to establish and maintain a computer information system which is designed to transmit a set of approximately 30 selected critical plant parameters. The ERDS would be activated by the licensee upon declaration of an alert or higher emergency condition at a licensed nuclear power reactor facility. Tests with the ERDS indicate that a computer-based transmission system is far more accurate and timely than the current practice of relaying information on plant conditions via telephone voice communication. Moreover, by automatically collecting and transmitting selected critical parameters to the NRC Operations Center, the ERDS would allow the licensee to redirect resources that now are required for voice communication of plant conditions to managing the emergency. Of course, the voice communication channel would remain available to permit needed dialogue between the licensee's facility and the NRC Operations Center during the emergency.

The proposed ERDS requirement would apply to all nuclear power reactor facilities except Big Rock Point and those that are permanently or indefinitely shut down. Big Rock Point is exempt because the facility has only five data points available for the ERDS program. Those units shut down for maintenance or authorized only for fuel loading and low power operations are required to report under ERDS.

The ERDS would become operational during (1) emergencies at the licensee's facilities and (2) emergency training exercises if the licensee's computer system has the capability to transmit the exercise data. The licensee would activate the ERDS to begin data transmission to the NRC Operations Center immediately after declaring an alert or a higher emergency classification.

The licensee would be required to provide the necessary software to assemble the data and an output communications port for each reactor unit in its in-plant computer system. The required emergency data would be transmitted to the NRC via NRC-furnished communication link hardware. The acquisition and transmission of data would not require human intervention after the system is activated, thereby ensuring uninterrupted transmission of real-time data. The data would be transmitted in a format compatible with the system at the NRC Operations Center. Guidance for format compatibility with the NRC receiving system is provided in NUREG-1394.

The two main features of the ERDS are:

o The software link, which will extract and format the requisite data to be transmitted to the NRC Operations Center; and

o The hardware link, which will connect the onsite data acquisition system of the licensee with the data transmission unit supplied by the

NRC. In most cases, implementing ERDS can be accomplished with already installed equipment at the licensee's facility.

The parameters to be included in the transmission are those that, to the greatest extent possible, describe the four selected plant conditions previously mentioned. The specific parameters desired by the NRC during an emergency are given in the proposed amendment to 10 CFR Part 50, Appendix E, Section VI, Paragraph 2. The units of these parameters are pre-established for each site and will be transmitted to the NRC Operations Center without any change. If the data for a selected plant condition parameter exists, but cannot be transmitted electronically from a licensee's system, then the licensee will continue to provide that data via the existing ENS.

With regard to the capability of the current hardware at the sites to support the generation of data required as input to ERDS, approximately 5 to 10 percent of the licensee computer systems are currently running at close to 100 percent processing capability in the post-trip or post-incident environment. Approximately 10 to 15 percent of the licensee systems are hardware limited (i.e., no available output port for an ERDS connection exists). However, in many of these cases, the licensees with hardware limitations were planning to upgrade their systems in the near future for reasons other than supporting ERDS.

Each licensee would establish and maintain an ERDS configuration control program which would ensure that the NRC is notified of any changes to the ERDS on-site hardware or software. Any hardware and software changes that affect the transmitted data points identified in the ERDS Data Point Library (data case) must be reported to the NRC within 30 days after changes are completed. Any changes that could affect the transmission format and communication protocol to the ERDS must be provided to the NRC, as soon as practicable, at least 30 days prior to the modification.

Other computer systems, such as the Nuclear Data Link (NDL) were considered; however, these would require new hardware and software as well as additional personnel for both licensees and the NRC.

Environmental Impact: Categorical Exclusion

The NRC has determined that this proposed regulation is the type of action described in categorical exclusion 10 CFR 51.22(c)(3)(iii). Therefore, neither an environmental impact statement nor an environmental assessment has been prepared for this proposed regulation.

Paperwork Reduction Act Statement

This proposed rule amends information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). This rule has been submitted to the Office of Management and Budget for review and approval of the paperwork requirements.

The regulatory analysis estimates an annual per reactor level of effort of 5 days for licensee staff and 3 days for NRC staff for the maintenance of the on-site ERDS configuration control program. An integral part of this activity is the preparation of configuration control reports by the licensee and their review by the NRC. This paperwork effort is estimated at less than one-third the overall configuration control level of effort. Thus, the reporting burden per reactor is estimated at less than 2 days per year, and the NRC's review effort is estimated at less than 1 day per reactor year. Send comments regarding this burden estimate or any aspect of this collection of information, including suggestions for reducing the burden, to the Information and Records Management Branch (MNBB-7714), U.S. Nuclear Regulatory Commission, Washington, DC 20555 and to the Paperwork Reduction Project (3150-0011), Office of Information and Regulatory Affairs (NEOB-3019), Office of Management and Budget, Washington, DC 20503.

Regulatory Analysis

The NRC has prepared a draft regulatory analysis on this proposed regulation. The analysis examines the costs and benefits of the alternatives considered by the NRC. The draft regulatory analysis is available for inspection in the NRC Public Document Room, 2120 L Street, NW. (Lower Level), Washington, DC. Single copies of the draft analysis may be obtained from M. L. Au, P.E., Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC 20555, telephone (301) 492-3749.

The NRC requests public comment on the draft regulatory analysis. Comments on the draft analysis may be submitted to the NRC as indicated under the ADDRESSES heading.

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act of 1980 (5 U.S.C. 605(b)), the Commission certifies that this rule will not, if promulgated, have a significant economic impact on a substantial number of small entities. This proposed rule affects only the licensing and operation of nuclear power plants. The companies that own these plants do not fall within the scope of the definition of "small entities" set forth in the Regulatory Flexibility Act or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR 121.

Backfit Analysis

As required by 10 CFR 50.109, the Commission has completed a backfit analysis for this proposed rule. The Commission concluded that the proposed rule will provide substantial increase in the overall protection of the public health and safety by ensuring far more accurate and timely flow of data for the NRC to fulfill its role during an alert or higher emergency. The direct and indirect costs estimated for the implementation of this rule are justified in view of this increased protection. Further, the implementation and maintenance requirements of the proposed rule will have no effect on occupational radiological exposure. The backfit analysis on which this determination is based is as follows:

Item 1: Statement of the specific objective that the proposed backfit is designed to achieve.

Response: The objective of the proposed ERDS rulemaking effort is to achieve a high degree of assurance that accurate real-time data is made available to the

NRC to evaluate critical parameters at any operating reactor facility during an alert or higher emergency. This in turn would improve the NRC's understanding of an event and allow the NRC to perform its role more effectively and efficiently which includes: (i) monitoring the licensee to ensure that appropriate recommendations are being made with respect to offsite protective actions; (ii) providing the licensee with technical analysis and logistic support; (iii) supporting offsite authorities; (iv) keeping other Federal agencies and entities informed of the status of the incident; and (v) keeping the media informed of the NRC's knowledge of the status of the incident.

In addition, the implementation of the ERDS would enable the licensee to better use its time and resources to effectively and efficiently deal with the emergency. The combination of better and more timely assessments of licensee actions by the NRC and the focusing of the licensee's resources to better deal with the emergency at hand together will reduce the overall risk to the public health and safety from an emergency.

Item 2: General description of the activity that would be required of the licensee or applicant in order to complete the backfit.

Response: All licensees or applicants would be required to install an NRC-supplied communication link, provide the software necessary to format available selected critical plant condition data for NRC use, provide the necessary hardware from the in-plant computer to interface with the NRC-supplied communication link, provide support for periodic testing of the ERDS, and report any configuration changes to the licensee's ERDS-related hardware and software. Initially, the ERDS will be tested quarterly, unless otherwise determined by NRC based on demonstrated system performance.

Item 3: Potential change in the risk to the public from the accidental offsite release of radioactive material.

Response: The implementation of the ERDS in all operating nuclear power reactors would provide the NRC with more accurate and timely data to fulfill its major role during an alert or higher emergency. The major role, as defined in the 1987 revision to NUREG-0728, is to monitor the licensee to ensure that appropriate recommendations are being made with respect to offsite protective actions. Currently, the NRC relies on data verbally transmitted through the Emergency Notification System (ENS) during an emergency. Although deemed adequate, this method of transmission has, on occasion, proven to be unreliable. In addition, data collection is time consuming since various instruments are read and their indications logged on a periodic basis for verbal communication via ENS. The implementation of the ERDS would improve the reliability and timeliness of data transmission and help ensure that any reactor unit in distress can be suitably monitored. Therefore, the NRC would be able to make better and more timely assessments of the licensee's actions regarding management of both emergency and protective actions. Although licensees will be required to maintain voice communication via the Emergency Notification System (ENS) with ERDS, the licensee resources that now are required to collect and relay data and information to the NRC will be available to deal with the emergency. The combination of better and more timely assessments of licensee actions by the NRC, and the focusing of licensee resources to better deal with the emergency at hand together will reduce the overall risk to the public health and safety from an emergency.

Item 4: Potential impact on radiological exposure of facility employees.

Response: The implementation of the proposed ERDS rule would have no effect on routine occupational radiological exposure and would not result in increased radiological exposure of facility employees.

Item 5: Installation and continuing costs associated with the backfit, including the cost of facility downtime or the cost of construction delay.

Response: The cost impact of the rule was estimated to be approximately $153,000 for one nuclear power reactor (one unit). This figure, expressed in 1990 dollars, represents the incremental worth of installing and operating ERDS for 30 years using a 5 percent discount rate. The overall industry cost of implementing the rule for 118 nuclear power reactor units was estimated at approximately $18 million. No downtime costs were considered in the cost impact estimates because the installation and operation of the ERDS should have no impact on the operation of a nuclear power plant.

Item 6: The potential safety impact of changes in plant or operational complexity, including the relationship to proposed and existing regulatory requirements.

Response: The proposed ERDS rule should have little or no impact on the operational complexity of the nuclear power reactor units since the required modifications to the hardware and software are minor. The redirection in the labor burden provided by the automatic collection and transmission of selected reactor data would increase the efficiency and effectiveness of nuclear power plant operating personnel during an emergency. The proposed rule is closely associated with Generic Letter 89-15 and complements the ENS that exists at every nuclear power reactor.

Item 7: The estimated resource burden on the NRC associated with the proposed backfit and availability of such resources.

Response: The impact on the NRC resulting from the implementation of the proposed ERDS rule is anticipated to be a one-time cost of about $200,000 for the current population of operational/licensed nuclear reactor units. This figure provides for initial reviews of licensees' implementation plan submittals. After implementation, the NRC cost is estimated to be approximately $4.3 million for 118 nuclear power reactor units. This figure represents the costs for periodic testing and configuration control expressed as the present worth in 1990 dollars and uses a 5 percent discount rate over 30 years.

Item 8: The potential impact of the differences in facility type, design, or age on the relevancy and practicality of the proposed backfit.

Response: The proposed rule is independent of the facility's type, design, or age. There are considerable variations in the instrumentation systems of the nuclear power plants, and the estimated cost impacts were based on an average value for current nuclear power plants to implement the ERDS. There will be no differences, however, in potential impacts between the various facilities on a yearly basis. The proposed rule does not require that licensees monitor more parameters than are presently monitored at each facility.

Item 9: Whether the proposed backfit is interim or final and, if interim, the justification for imposing the proposed backfit on an interim basis.

Response: Implementation of the ERDS in accordance with the proposed rule will require that all licensees develop and submit an ERDS implementation plan to the NRC within 60 days of the publication of the final rule in the Federal Register. The implementation plan should provide a schedule which identifies the earliest possible time frame for ERDS implementation by the licensee as well as proposed alternate implementation dates. The NRC will establish an industry wide ERDS implementation schedule which will take into account such factors as planned computer modifications and scheduled outages. The ERDS must be implemented within 18 months of the publication of the final rule in the Federal Register.

List of Subjects in 10 CFR Part 50

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

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 553, the NRC is proposing to adopt the following amendment to 10 CFR Part 50.

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

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

AUTHORITY: Secs. 102, 103, 104, 105, 161, 182, 183, 186, 189, 68 Stat. 936, 937, 938, 948, 953, 954, 955, 956, as amended, sec. 234, 83 Stat. 1244, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2201, 2232, 2233, 2236, 2239, 2282); secs. 201, as amended, 202, 206, 88 Stat. 1242, as amended, 1244, 1246, (42 U.S.C. 5841, 5842, 5846).

Section 50.7 also issued under Pub. L. 95-601, sec. 10, 92 Stat. 2951 (42 U.S.C. 5851). Section 50.10 also issued under secs. 101, 185, 68 Stat. 936, 955, as amended (42 U.S.C. 2131, 2235), sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.13, and 50.54(dd), also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Sections 50.23, 50.35, 50.55, and 50.56 also issued under sec. 185, 68 Stat. 955 (42 U.S.C. 2235). Sections 50.33a, 50.55a, and Appendix Q also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853 (42 U.S.C. 4332). Sections 50.34 and 50.54 also issued under sec. 204, 88 Stat. 1245 (42 U.S.C. 5844) Sections 50.58, 50.91, and 50.92 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Section 50.78 also issued under sec. 112, 68 Stat. 939 (42 U.S.C. 2152). Sections 50.80 through 50.81 also issued under sec. 184, 68 Stat. 954, as amended (42 U.S.C. 2234). Section 50.103 also issued under sec. 108, 68 Stat. 939, as amended (42 U.S.C. 2138). Appendix F also issued under sec. 187,
68 Stat. 955 (42 U.S.C. 2237).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273), §§ 50.46(a) and (b), and 50.54(c) are issued under sec. 161b, 68 Stat. 948, as amended (42 U.S.C. 2201(b)); §§ 50.7(a), 50.10(a)-(c), 50.34(a) and (e), 50.44(a)-(c), 50.46(a) and (b), 50.47(b), 50.48(a), (c), (d), and (e), 50.49(a), 50.54(a), (i), (i)(1), (l)-(n), (p), (q), (t), (v), and (y), 50.55(f), 50.55a(a), (c)-(e), (g), and (h), 50.59(c), 50.60(a), 50.62(c), 50.64(b), and 50.80(a) and (b) are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 50.49(d), (h), and (j), 50.54(w), (z), (bb), (cc), and (dd), 50.55(e), 50.59(b), 50.61(b), 50.62(b), 50.70(a), 50.71(a)-(c) and (e), 50.72(a), 50.73(a) and (b), 50.74, 50.78, and 50.90 are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

PART 50 - DOMESTIC LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

2. In § 50.72, paragraph (a)(4) is redesignated as paragraph (a)(5) and a new paragraph (a)(4) is added to read as follows:

§ 50.72 Immediate notification requirements for operating nuclear power reactors.

(a) * * *

(4) The licensee shall activate the Emergency Response Data System (ERDS)[5] for any condition that requires the declaration of an emergency class of alert, site area emergency, or general emergency at the time that the NRC Operations Center is notified of the emergency class declaration.

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5 Requirements for ERDS are addressed in Appendix E.

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3. Appendix E is amended by adding a new Section VI, Emergency Response Data System, to read as follows:

Appendix E - Emergency Planning and Preparedness for Production and Utilization Facilities

*   *   *   *   *

VI. Emergency Response Data System

1. The Emergency Response Data System (ERDS) is a direct real-time electronic data link between the licensee's onsite computer system and the NRC Operations Center which provides for the automated transmission of a limited data set of selected parameters. The ERDS supplements the existing voice transmission over the Emergency Notification System (ENS) by providing the NRC Operations Center with timely and accurate updates of a limited set of parameters from the licensee's installed onsite computer system in the event of an emergency. When selected plant data are not available on the licensee's onsite computer system, retrofitting of data points is not required. The licensee shall test the ERDS periodically to verify system availability and operability. The frequency of ERDS testing will be quarterly unless otherwise set by NRC based on demonstrated system performance.

2. Except for Big Rock Point and all nuclear power facilities that are shut down permanently or indefinitely, onsite hardware and software shall be provided at each unit by the licensee to interface with the NRC receiving system. The hardware and software must have the following characteristics:

a. Data points, if resident in the in-plant computer systems, must be transmitted for four selected types of plant conditions: reactor core and coolant system conditions; reactor containment conditions; radioactivity release rates; and plant meteorological tower data. A separate data feed is required for each reactor unit. While it is recognized that ERDS is not a safety system, it is conceivable that a licensee's ERDS interface could communicate with a safety system. In this case, appropriate isolation devices would be required at these interfaces.[6] The data points, identified in the following parameters will be transmitted:

(i) For pressurized water reactors (PWRs), the selected plant parameters are: (1) Primary coolant system: pressure, temperatures (hot leg, cold leg, and core exit thermocouples), subcooling margin, pressurizer level, reactor coolant charging/makeup flow, reactor vessel level (when available), reactor coolant flow, and reactor power; (2) Secondary coolant system: steam generator levels and pressures, main feedwater flows, and auxiliary and emergency feedwater flows; (3) Safety injection: high- and low-pressure safety injection flows, safety injection flows (Westinghouse), and borated water storage tank level; (4) Containment: pressure, temperatures, hydrogen concentration, and sump levels; (5) Radiation monitoring system: reactor coolant radioactivity,

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6 See 10 CFR 50.55a(h) Protection Systems.

containment radiation level, condenser air removal radiation level, effluent radiation monitors, and process radiation monitor levels; and (6) Meteorological data: wind speed, wind direction, and atmospheric stability.

(ii) For boiling water reactors (BWRs), the selected parameters are: (1) Reactor coolant system: reactor pressure, reactor vessel level, feedwater flow, and reactor power; (2) Safety injection: reactor core isolation cooling flow, high-pressure coolant injection/high-pressure core spray flow, core spray flow, low-pressure coolant injection flow, and condensate storage tank level; (3) Containment: drywell pressure, drywell temperatures, drywell sump levels, hydrogen and oxygen concentrations, suppression pool temperature, and suppression pool level; (4) Radiation monitoring system: reactor coolant radioactivity level, primary containment radiation level, condenser off-gas radiation level, effluent radiation monitor, and process radiation levels; and (5) Meteorological data: wind speed, wind direction, and atmospheric stability.

b. The above selected parameter sets must be transmitted at time intervals not less than 15 seconds or more than 60 seconds.

c. All link control and data transmission must be established in a format compatible with the NRC receiving system.[7]

3. Maintaining Emergency Response Data System

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7 Guidance is provided in NUREG-1394

a. Any hardware or software changes that affect the transmitted data points identified in the Emergency Response Data System Data Point Library (data base) must be submitted to the NRC within 30 days after changes are completed.

b. Hardware and software changes, with the exception of data point modifications, that could affect the transmission format and computer communication protocol to the ERDS must be provided to the NRC, as soon as practicable, at least 30 days prior to the modification.

4. Implementing Procedures for Emergency Response Data System

a. Each licensee shall develop and submit an ERDS implementation program plan to the NRC by [insert a date 75 days after publication of the final rule]. To ensure compatibility with the guidance provided for the Emergency Response Data System (ERDS), the ERDS implementation program plan must include, but not be limited to, information on the licensee's computer system configuration (i.e., hardware and software), interface, and procedures. Applicants for an operating license must comply with Appendix E, Section V of this part.

b. Each licensee shall complete implementation of the Emergency Response Data System by [insert a date eighteen months after the effective date of the final rule] or before initial escalation to full power, whichever comes later. Licensees with currently operational ERDS interfaces approved under the voluntary ERDS implementation program[8] will be considered to have met the requirements for ERDS under Appendix E, Sections VI.1, and 2 of this part.

Dated at Rockville, Maryland, this 2d day of October, 1990.

For the Nuclear Regulatory Commission.

Samuel J. Chilk,

Secretary.

8 See, NUREG-1394.

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

HELTEMES

JUN 23 1989

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

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

SUBJECT: WAIVER OF CRGR REVIEW OF PROPOSED REVISION OF 10 CFR 55 TO REQUIRE COMPLIANCE WITH FITNESS-FOR-DUTY PROGRAMS AND CONFORMING MODIFICATION TO COMMISSION'S ENFORCEMENT POLICY

Enclosed is a proposed revision of 10 CFR 55 to be sent to the Commission for review and approval for publication in the Federal Register. The Commission approved the "Final Rulemaking Fitness-for-Duty Programs (Part 26)" on March 22, 1989 and directed the staff to prepare a notice of proposed rulemaking to amend Part 55 so that the cutoff levels established pursuant to Part 26 become applicable to the licensed operator as a condition of their license. It further requested the staff to amend Part 2 Appendix C, to reflect appropriate enforcement sanctions for individual licensed operators. The existing 10 CFR 55.53 (d) requires the licensed operator to comply with all applicable rules, regulations and orders of the Commission. However, the Commission wanted it to be made clear to the licensed operator, what the penalty would be for violating the cutoff levels of Part 26 so that the operators have full notice of the gravity of any violation.

The proposed revision to Part 55 will only serve to provide full notice to licensed individuals of the conditions under which they are expected to perform their licensed duties and does not present any new staff position or constitute a new requirement; therefore we believe that CRGR review is not necessary.

Due to a Commission directive to prepare a notice of proposed rulemaking and return it to the Commission for review and approval, on a very short schedule, any comments concerning waiver of CRGR review are requested as soon as possible.

James H Sniezek

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

Enclosure:
As stated

~~8907120107~~ XA

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: PROPOSED REVISION OF 10 CFR PART 55 TO REQUIRE COMPLIANCE WITH FITNESS-FOR-DUTY PROGRAMS AND CONFORMING MODIFICATION TO COMMISSION'S ENFORCEMENT POLICY

Purpose: To obtain Commission approval to publish a notice of proposed rulemaking that revises §55.53 and §55.61 of 10 CFR Part 55 to require that compliance with the Fitness-for-Duty Programs (Part 26) is a condition of an operator or a senior operator license. A conforming modification to the Commission's enforcement policy, Appendix C to 10 CFR Part 2, is described.

Background: SECY-89-30, "Final Rulemaking - Fitness-for-Duty Programs (Part 26)," was approved by the Commission subject to the conditions stated in a staff requirements memorandum (SRM) of March 22, 1989. The SRM directed the staff to prepare a notice of proposed rulemaking to amend Part 55 so that the cutoff levels established pursuant to Part 26 become applicable to the licensed operators as a condition of their license. It further requested the staff to amend Part 2, Appendix C, to reflect appropriate enforcement sanctions for individual licensed operators.

Discussion: The Commission indicated in its SRM that it should be made clear what the penalty would be for violating the cutoff levels for substances described in Part 26, "Fitness-for-Duty Programs," so that the operators at nuclear power facilities will have full notice of the gravity of any violation and that 10 CFR Part 2 should be modified to reflect enforcement sanctions for individual operators who violate these cutoff levels. A summary of the staff's response to the SRM that indicates the changes to Part 55 and Part 2 is provided with the Notice of Proposed Rulemaking.

CONTACT:
David J. Lange, NRR
492-3172

Subpart G of 10 CFR, Part 55, "Modification and Revocation of Licenses," describes the circumstances when licenses may be modified or revoked, including for willful violation of or for failure to observe any of the terms, or conditions of a license. Subpart H, "Enforcement" indicates that civil penalties may be imposed for violation of a license issued under Section 107 ("Operators' Licenses") of the Atomic Energy Act. Therefore, making compliance with fitness-for-duty requirements a condition of an operator's license will provide a basis for issuing a notice of violation or civil penalty to operators who violate such provisions. This condition will be applicable to both power and non-power licensed operators.

It is the staff's position that the proposed amendment to Part 55 (§55.53, "Conditions of Licenses") will clearly describe the obligation of the licensed operator to comply with the conditions and cutoff levels established pursuant to 10 CFR Part 26, "Fitness-for-Duty Programs." Further, the proposed amendment to Part 55 (§ 55.61, "Modification and revocation of Licenses") will provide explicit notice for the terms or conditions under which a license may be revoked, suspended or modified.

The SRM also requested that the Enforcement Policy be amended by rulemaking along with the rulemaking of the changes to 10 CFR Part 55. The Commission in the past has not modified the Enforcement Policy by rulemaking, therefore the staff proposes to modify the Enforcement Policy in conjunction with the final rulemaking, as described in the Supplementary Information in the enclosed proposed amendment of 10 CFR Part 55. The Supplementary Information for the proposed rulemaking states that NRC intends to modify the Enforcement Policy as follows:

> In cases involving a licensed operator's failure to meet applicable fitness-for-duty requirements, the NRC may issue a notice of violation or a civil penalty to the Part 55 licensee including for the first time an individual fails a drug or alcohol test established to determine compliance with the cutoff levels of 10 CFR Part 26. The NRC may issue an order to suspend the Part 55 license for a period up to 3 years the second time an individual fails such a drug or alcohol test. In the event there are less than 3 years remaining in the term of the individual license, NRC may consider not renewing the individual license until the 3 year period is complete. The NRC may issue an order to revoke the Part 55 license the third time

an individual fails such a drug or alcohol test. A licensed operator or applicant who refuses to participate in the drug and alcohol testing programs established to determine compliance with the cutoff levels of 10 CFR Part 26 or involved in the sale, use, or possession of a controlled substance may be subject to license suspension, revocation, or denial. In addition, positive test results and failures to participate in drug and alcohol testing programs may be considered in making decisions concerning renewal of a Part 55 license. To assist in determining the severity levels of potential violations, Supplement I would be modified to provide an example at Severity Level I of a licensed operator performing duties while unfit and an example at Severity Level III of a licensed operator's initial failure of a drug or alcohol test.

The staff at that time will also modify the Enforcement Policy to state that civil penalty actions against licensed operators will require approval of the Commission in accordance with the Commission's direction in the Peach Bottom case (SECY-88-201).

Recommendations: That the Commission

Approve publication in the Federal Register of a notice of proposed rulemaking amending 10 CFR Part 55, Subpart F, to establish a new condition of an operator's license which would prohibit conduct of licensed activities while under the influence of any substance or mentally or physically impaired in any manner which could adversely affect performance of licensed duties and amend 10 CFR Part 55, Subpart G, to provide explicit additional notice for the terms or conditions under which a license may be revoked, suspended or modified.

Victor Stello, Jr.
Executive Director
for Operations

Enclosure:
Notice of Proposed Rulemaking

Nuclear Regulatory Commission
10 CFR Part 55
Operators' Licenses

Agency: Nuclear Regulatory Commission.

Action: Proposed Rule.

Summary: The Nuclear Regulatory Commission proposes to amend its regulations so that the conditions and cutoff levels established pursuant to 10 CFR Part 26, "Fitness-for-Duty Programs," become applicable to licensed operators as a condition of their license. The proposed rule will provide a basis for taking enforcement actions against licensed operators who use drugs or alcohol in a manner that would result in violation of the cutoff levels established pursuant to the Fitness-for-Duty rule.

On June 7, 1989, the Commission issued a new part to its regulations, Part 26, "Fitness-for-Duty Programs," requiring facility licensees authorized to operate nuclear power reactors to implement a Fitness-for-Duty Program that would provide reasonable assurance that nuclear power plant personnel are not under the influence of any legal or illegal substance that in any way would adversely affect their ability to safely perform their job duties.

The proposed revision to Part 55 will assure a safe operational environment for the performance of all licensed activities by taking into account the aspects of existing industry programs on fitness-for-duty and providing a clear understanding to licensed operators of the severity of violating requirements governing drug and alcohol use and of the impact of substance abuse.

Dates: The comment period expires [insert date 60 days from date of Federal Register publication]. Comments received after this date will be considered if it is practicable to do so, but assurance of consideration can be given only for comments filed on or before that date.

Addresses: Submit written comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555, ATTN: Docketing and Service Branch. Hand deliver comments to Docketing and Service Branch, One White Flint North, 11555 Rockville Pike, Rockville, MD, between 8:15 a.m. and 5:00 p.m.

Examine comments received at: The NRC Public Document Room, 2120 L Street NW., Washington, DC.

FOR FURTHER INFORMATION CONTACT: Kenneth E. Perkins, Jr., Chief, Operator Licensing Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: (301) 492-1031.

SUPPLEMENTARY INFORMATION

BACKGROUND

The Nuclear Regulatory Commission has issued its regulations to require licensees authorized to construct or operate nuclear power reactors to implement a fitness-for duty program. The general objective of this program is to provide reasonable assurance that nuclear power plant personnel are reliable, trustworthy, and not under the influence of any substance, legal or illegal, or mentally or physically impaired from any cause, which in any way adversely affects their ability to safely and competently perform their duties. A fitness-for-duty program developed under the requirements of this rule is intended to create an environment which is free of drugs and the effects of such substances.

The Commission has taken this action and is now proposing to add specific conditions to operator licenses to significantly increase assurance of public health and safety. The scientific evidence is conclusive that significant decrements in cognitive and physical task performance result from intoxication due to illicit drug abuse, as well as the use and misuse of legal substances. Given the addictive and impairing nature of certain drugs, while recognizing that the presence of drug metabolites does not necessarily relate directly to a current impaired state, the presence of drugs does strongly suggest the likelihood of past, present, or future impairment affecting job activities. In addition, the NRC believes that the reliability, integrity, and trustworthiness of persons working within nuclear power plants is important to assure public health and safety. Since there is an underlying assumption that workers will abide by the licensee's policies and procedures, any involvement with illegal drugs shows that the worker cannot be relied upon to obey laws of a health and safety nature, indicating that the individual may not scrupulously follow rigorous procedural requirements with the integrity required in the nuclear power industry to assure public health and safety.

The Commission considers unimpaired job performance by each licensed Operator or Senior Operator vital in assuring safe facility operation. The use of alcohol and drugs can directly impair job performance. Other causes of impairment include use of prescription and over-the-counter medications, emotional and mental stress, fatigue illness including allergies and physical psychological impairments. The effects of alcohol, which is a drug, are well known and documented, and therefore, are not repeated here. Drugs such as marijuana, sedatives, hallucinogens, and high doses of stimulants could adversely affect an employee's ability to correctly judge situations and make decisions (NUREG/CR-3196, "Drug and Alcohol Abuse: The Bases for Employee Assistance Programs in the Nuclear Industry," available from the National Technical Information Service). The greatest impairment occurs shortly after use or abuse, and the negative short-term effects on human performance (including subtle or marginal impairments that are difficult for a supervisor to detect) can last for several hours or days. The proposed amendment to 10 CFR Part 55 will establish a new condition of an operator's license which will prohibit conduct of licensed duties while under the influence of any legal or illegal substance or physically or mentally impaired in a manner which would adversely affect performance of licensed duties. The proposed amendment to Part 55 will be applicable to both power and nonpower reactor licensed operators. This rulemaking is not intended to backfit the provisions of 10 CFR Part 26 on nonpower facility licensees, but to make it clear to all licensed reactor operators (power and nonpower) through a condition of their license that use of drugs and alcohol in any manner which could adversely affect performance of licensed duties is prohibited.

As explained in the Commission's Enforcement Policy (see 53 FR 40027, Thursday, October 13, 1988), the Commission may take enforcement action where the conduct of the individual places in question the NRC's reasonable assurance that licensed activities will be properly conducted. The Commission may take enforcement action for reasons that would warrant refusal to issue a license on an original application. Accordingly, enforcement action may be taken regarding matters that raise issues of trustworthiness, reliability, use of sound judgment, integrity, competence, fitness-for-duty, or other matters that may not necessarily be a violation of specific Commission requirements.

Individuals licensed under 10 CFR Part 55 who are not reliable and trustworthy; who have been found, at any time, while employed by a licensed facility and having unescorted access, to have used drugs or alcohol in a manner which caused them to violate the cutoff levels established pursuant to 10 CFR Part 26; or who are mentally or physically impaired in any way that adversely affects their ability to safely and competently perform their duties will not be permitted to perform functions that could result in a risk to the health and safety of the public or other workers. Accordingly, the Commission proposes to amend Subpart F of 10 CFR Part 55 to establish as a condition of an operator's license a provision precluding performance of licensed duties while under the influence of drugs or alcohol or while mentally or physically impaired in any manner which could adversely affect performance. The Commission further proposes to amend Subpart G of 10 CFR Part 55 to provide explicit additional notice of the terms and conditions under which a license may be revoked, suspended or modified.

When the amended rule becomes effective, licensed operators will be subject to notices of violation, civil penalties or orders for violation of this condition. Therefore, in addition to amending the regulations to establish the Part 55 licensee's obligations, the Commission intends to modify the NRC Enforcement Policy in conjunction with the final rulemaking. It is the Commission's intention to modify the Enforcement Policy as follows:

> In cases involving a licensed operator's failure to meet applicable fitness for duty requirements, the NRC may issue an order, notice of violation or civil penalty to the Part 55 licensee including for the first time an individual fails a drug or alcohol test established to determine compliance with the cutoff levels of 10 CFR Part 26. The NRC may issue an order to suspend the Part 55 license for up to 3 years the second time an individual fails such a drug or alcohol test. In the event there are less than 3 years remaining in the term of the individual license, NRC

may consider not renewing the individual license or issuance of a new license until the 3 year period is complete. The NRC may issue an order to revoke the Part 55 license the third time an individual fails such a drug or alcohol test. A licensed operator or applicant who refuses to participate in the drug and alcohol testing programs established to determine compliance with the cutoff levels of 10 CFR Part 26 or who is involved in the sale, use, or possession of a controlled substance may be subject to license suspension, revocation, or denial. In addition, positive test results and failures to participate in drug and alcohol testing programs may be considered in making decisions concerning renewal of a Part 55 license.

To assist in determining the severity levels of potential violations, Supplement I would be modified to provide an example at Severity Level 1 of a licensed operator performing duties while unfit and an example at Severity Level III of a licensed operator's initial failure of a drug or alcohol test.

Environmental Impact: Categorical Exclusion

The NRC has determined that this proposed rule is the type of action described in categorical exclusion 10 CFR 51.22(c)(1). Therefore, neither an environmental impact statement nor an environmental assessment has been prepared for this proposed rule.

Paperwork Reduction Review

This proposed rule contains no information collection requirements and therefore is not subject to the requirements of the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.).

REGULATORY ANALYSIS

The regulations in 10 CFR Part 55 establish procedures and criteria for the issuance of licenses to Operators and Senior Operators of utilization facilities licensed pursuant to the Atomic Energy Act of 1954, as amended, or Section 202 of the Energy Reorganization Act of 1974, as amended, and 10 CFR Part 50. These established procedures provide for the terms and conditions upon which the Commission will issue, modify, maintain, and renew Operator and Senior Operator licenses.

Subpart F of Part 55, under §55.53 ("Conditions of Licenses"), sets forth the requirements and conditions for the maintenance of Operator and Senior Operator licenses.

Amending Subpart F to prohibit performance of licensed duties while under the influence of drug or alcohol or while mentally or physically impaired in any manner which could adversly affect safe and competent performance of licensed duties will provide notice to licensed individuals of the gravity of violating these requirements and will provide assurance that nuclear facilities are being operated safely.

Amending Subpart G to provide explicit additional notice to licensed operators of the terms and conditions under which a license may be revoked, suspended or modified will describe circumstances in which enforcement action will be taken and the penalty for violating the conditions or cutoff levels established pursuant to 10 CFR Part 26.

REGULATORY FLEXIBILITY CERTIFICATION

The proposed rule will not have a significant economic impact upon a substantial number of small entities. Many operator license applicants or operator licensees fall within the definition of small businesses found in Section 34 of the Small Business Act, 15 U.S.C. 632, or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121, or the NRC's size standards published December 9, 1985 (50 FR 50241). The proposed rule will only serve to provide notice to licensed individuals of the conditions under which they are expected to perform their licensed duties.

Thus, in accordance with the Regulatory Flexibility Act, 5 U.S.C. 605(b), the NRC hereby certifies that this rule, if promulgated, will not have a significant economic impact upon a substantial number of small entities.

BACKFIT ANALYSIS

This proposed rule does not modify or add to systems, structures, components, or design of a facility; the design approval or manufacturing license for a nuclear reactor facility; or the procedures or organization required to design, construct, or operate a facility. Accordingly, no backfit analysis pursuant to 10 CFR 50.109(c) is required for this proposed rule.

List of Subjects in 10 CFR Part 55

Manpower training programs, Nuclear power plants and reactors, Penalty, Reporting and record-keeping requirements

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 553, the NRC is proposing to adopt the following amendments to 10 CFR Part 55.

PART 55 - OPERATORS' LICENSES

1. The authority citation for Part 55 continues to read as follows:

AUTHORITY: Sec. 107, 161, 182, 68 Stat. 939, 948 953, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2137, 2201, 2232, 2282); secs. 201, as amended, 202, 88 Stat. 1242, as amended, 1244 (42 U.S.C. 5841, 5842).

Sections 55.41, 55.43, 55.45 and 55.59 also issued under sec. 306, Pub. L. 97-425, 96 Stat. 2262 (42 U.S.C. 10226). Section 55.61 also issued under secs. 186, 187, 68 Stat. 955 (42 U.S.C. 2236, 2237).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273); §§ 55.3, 55.21, 55.49 and 55.53 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 55.9, 55.23, 55.25, and 55.53(f) are i sued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

2. In §55.53, paragraph (j) is redesignated as paragraph (l) and new paragraphs (j) and (k) are added to read as follows:

55.53 Conditions of licenses.

* * * * *

(j) The licensee shall not perform activities authorized by a license issued under this part while under the influence of any legal or illegal substance or mentally or physically impaired from any cause which adversely affects his or her ability to safely and competently perform his or her duties.

(k) The licensee shall participate in the drug and alcohol testing programs established to determine compliance with the cutoff levels of 10 CFR Part 26.

3. In §55.61, a new paragraph (b)(5) is added to read as follows:

55.61 Modification and revocation of licenses.

(b)(5) For the sale, use or possession of a controlled substance, or a confirmed positive test for drugs, drug metabolites or alcohol in violation of the conditions and cutoff levels established pursuant to 10 CFR Part 26, or use of alcohol within the protected area, or a determination of unfitness for scheduled work due to the consumption of alcohol.

* * * * *

Dated at Rockville, Maryland, this _______________ day of _______________.

For the Nuclear Regulatory Commission,

____
Samuel J. Chilk
Secretary of the Commission.

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

MAR 2 9 1991

ACTION

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

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

SUBJECT: REQUEST FOR WAIVER OF CRGR REVIEW OF REVISION OF 10 CFR 55 TO REQUIRE COMPLIANCE WITH FITNESS-FOR-DUTY PROGRAMS AND CONFORMING MODIFICATION TO COMMISSION'S ENFORCEMENT POLICY

Enclosed are revisions of 10 CFR Parts 2 and 55 to be sent to the Commission for review and approval for publication in the Federal Register. The associated Commission paper is also enclosed. By memorandum dated March 22, 1989 (copy enclosed), the Commission approved the "Final Rulemaking - Fitness-for-Duty Programs (Part 26)" on March 22, 1989, and directed the staff to prepare a notice of proposed rulemaking to amend Part 55 so that the cutoff levels established pursuant to Part 26 become applicable to the licensed operators as a condition of their license. It further requested the staff to amend Part 2, Appendix C, to reflect appropriate enforcement sanctions for individual licensed operators.

The regulations in 10 CFR Part 55 establish procedures and criteria for the issuance of licenses to operators and senior operators of utilization facilities. These established procedures provide for the terms and conditions upon which the Commission will issue, modify, maintain, and renew operator and senior operator licenses. Subpart F of Part 55, under §55.53 ("Conditions of Licenses"), sets forth the requirements and conditions for the maintenance of operator and senior operator licenses. The existing 10 CFR 55.53(d) requires licensed operators and senior operators to comply with all applicable rules, regulations and orders of the Commission. This rule only serves to emphasize to the 10 CFR Part 55 operator and senior operator the conditions they are required to comply with under 10 CFR Part 26, "Fitness-for-Duty Program."

A regulatory analysis for compliance with the conditions and cutoff levels, that examined the costs and benefits of the alternatives considered by the Commission, was prepared for Part 26 and was therefore not repeated for these revisions to Parts 2 and 55. The staff has determined that the backfit rule, 10 CFR 50.109, does not apply to this rule, and therefore, that a backfit analysis is not required for this rule, because these amendments do not involve any provisions which would impose backfits as defined in 10 CFR 50.109(a)(1). Furthermore, these revisions to Parts 2 and 55 contain no information collection requirements. Should it become necessary to request any information in a particular case, the purpose of such request would be solely to verify compliance with the current licensing basis for the individual operator in question and would be requested under the existing provision of 10 CFR 55.31(b).

CONTACT:
Robert M. Gallo, NRR
49-21031

By memorandum dated July 18, 1989, CRGR review of this matter was deferred to the final rule stage. The enclosed final rule includes a summary of the public comments and describes the changes made as a result of these comments.

Because how well a plant is operated is a vital component of plant safety, the Commission wanted to clearly state what the penalty would be for violating the cutoff levels of Part 26 so that the operators have full notice of the gravity of any violation. These revisions to Parts 2 and 55 will only serve to provide full notice to licensed individuals of the conditions under which they are expected to perform their licensed duties. They do not present any new staff positions or constitute any new requirements. Therefore, we believe that CRGR review is not necessary and we request a waiver of CRGR review. Any comments concerning waiver of CRGR review are requested as soon as possible.

Frank J. Miraglia Jr.

Frank J. Miraglia, Jr., Deputy Director
Office of Nuclear Reactor Regulation

Enclosures:
As stated

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

OFFICE OF THE SECRETARY

ACTION - Murley, NRR/
Bernero, NMSS/
Beckjord, RES/
Norry, ADM

Cys: Stello
Taylor
Thompson
Blaha
Bird, OP
Scroggins, OC
Lieberman, OE
LBush, NRR
Meyer, ADM
Shelton, IRM

March 22, 1989

MEMORANDUM FOR: Victor Stello, Jr.
Executive Director for Operations

FROM: Samuel J. Chilk, Secretary

SUBJECT: SECY-89-30 - FINAL RULEMAKING - FITNESS-FOR-DUTY PROGRAMS

This is to advise you that the Commission (with all Commissioners agreeing) has approved the Fitness for Duty Rule subject to the following:

1. In regard to the frequency of random testing, the Commission has agreed to option 5 (the 100% testing rate). (Commissioner Carr would have preferred a 300% testing rate and the use of a lower cutoff (50 ng/ml) for initial screening for marijuana.)

2. In regard to alcohol testing, the Commission has agreed to option 2 (the .04% Blood Alcohol Content cutoff). The last line in subsection 26.24(g) which indicates that alcohol concentrations below the specified limit should be evaluated, should be deleted. Licensees have the general responsibility for evaluating the fitness of their personnel whether or not some specified limit is indicated for either drugs or alcohol.

3. The Commission has agreed to remove benzodiazepines and barbituates from the panel of drugs to be tested. (Commissioner Carr would have preferred to include them.)

4. The NRC Guidelines and the Statement of Consideration should make it clear that the list of substances and cutoff levels specified in the rule may be amended in the future in response to advances in technology, additional experience, or other factors identified by HHS or the NRC.

5. The Commission has agreed that the rule should specify, as a minimum, a four hour period of abstinence from consumption of alcoholic beverages that should precede all scheduled shifts of work and that there should be complete abstinence while on duty. This is conditioned on confirmation by the staff that this period of abstinence from consumption of alcoholic beverages is sufficient in most cases to eliminate the effects of moderate drinking. This requirement should clearly state that it does not preclude licensees from using

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individuals they need in responding to an emergency. (Chairman Zech and Commissioner Carr while approving a minimum of 4 hours, would have preferred an 8 hour period of abstinence. Commissioners Roberts and Curtiss disapproved, and would instead leave it to licensees to consider alcohol abstinence programs rather than requiring a particular prework abstinence period.)

~~(EDO)~~ (NRR) (SECY SUSPENSE: 4/9/89) 8700229

6. Testing should be randomized as to the time of day to assure deterrence against lunch-time drinking.

7. The staff should assure adequate data collection with a requirement that the information be provided to the NRC on a periodic basis so that the licensees programs can be analyzed and so that the Commission can assess the effectiveness of the rule and, if necessary, make appropriate improvements or changes. The method of collection adopted should assure that comparable data is supplied to NRC by its licensees in areas critical to ensuring compliance with the rule. Particular care should be taken to assure that licensees who use lower cutoffs for any drugs report the data in a manner consistent with the reporting protocol for other licensees.

8. Section 2.1(d) of the NRC Guidelines, as presently written, provides that specimens collected under NRC regulations requiring compliance with this part may only be designated or approved for testing as described in this part and shall not be used to conduct any other analysis or test without the permission of the tested individual. This is an important safeguard. The staff should be certain that all portions of the final published package reflect the fact that the Commission has included this language in our guidance. Specifically, the discussion in the response to the public comments on use of the samples for other purposes (page 54, Section 11.2.2) should reference Section 2.1(d).

9. The Guidelines should explicitly include the GC/MS test for 6-monoacetylomorphine (MAM) (unless the staff identifies a sound technical basis for not specifying it) in the testing profile, if the imunoassay screen is positive for morphine, in order to reduce the possibility of false positives. The guidelines should specify that, in the absence of GC/MS identified MAM, the licensee should take no action unless the Medical Review Officer identifies additional clinical evidence of opiate abuse or misuse. As an additional safeguard to be included in the NRC guidance, the guidance should specify the nature of the additional clinical evidence the Medical Review Officer will use in the interpretation of a positive finding.

10. The attached additions to the statement of consideration, recommended by OGC, should be added to the rule.

11. The attached editorial changes and clarifications should be made in the Federal Register Notice. The staff should carefully review the FRN and assure that any additional changes needed for clarification or internal consistency with the Commission's decision are made and that the format is consistent with the editorial requirements for publication in the Federal Register. The revised FRN should be returned for final Commission review, Affirmation, signature and publication in the Federal Register.
~~(EDO)~~ (NRR) (SECY SUSPENSE: 4/14/89) 8700229

12. The staff should revisit the need for changes to the final rule within 18 months following the implementation date of the rule.
~~(EDO)~~ (NRR) (SECY SUSPENSE: 3/91) 8900042

13. The staff should prepare a Notice of Proposed Rulemaking to:

    a. Amend Part 55 so that the cutoff limits of Part 26 become applicable to the licensed operators as a condition of their license. It should be made clear what the penalty for violating the cutoff limits will be, so that the operators have full notice of the gravity of any violation.

    b. Amend Part 2, Appendix C, to reflect appropriate enforcement sanctions for individual licensed operators.

    The Notice of Proposed Rulemaking should be returned to the Commission for review and approval.
    ~~(EDO)~~ (NRR) (SECY SUSPENSE: 6/1/89) 8900043

14. The staff should study the need to amend Part 26 to include materials licensees and fuel cycle facilities and how drugs and alcohol abuse affects their safety; especially the security of Category I facilities. The results of the study and, if appropriate a proposed rule, should be provided to the Commission.
~~(EDO)~~ (NMSS/RES) (SECY SUSPENSE: 12/89) 8900027

15. The staff should further explore the need to amend Part 26 to add benzodiazepines and barbituates to the testing protocol and lower the cutoff levels for marijuana and amphetamines based on information it has or receives. In this regard the staff should specifically request the Secretary of HHS to review and comment on the advantages and disadvantages of the NRC extending Part 26 to include these additional drugs and lower cutoff levels. To assist HHS, the staff should provide HHS with available information concerning industry experiences with their fitness-for-duty programs and the procedural modifications the NRC has made to further protect individual rights, as the HHS procedures

are applied by NRC to the nuclear power industry. The staff should request a prompt response from HHS. The staff should keep the Commission informed of the status of the HHS review. The staff should provide recommendations regarding a proposed rule to the Commission based upon information available. (Chairman Zech and Commissioner Carr would have preferred to publish for comment a Notice of Proposed Rulemaking to amend Part 26 to add benzodiazepines and barbituates to the testing protocol and lower the cutoff levels for marijuana and amphetamines and request HHS to comment on the proposed rule.)

(NRR) -~~(EDO)~~-Provide status of review (SECY SUSPENSE: 9/89)

(NRR) -~~(EDO)~~-Provide Recommendations regarding Proposed rule to Commission (SECY SUSPENSE: 12/89)

8900044

16. As a separate matter, the staff should request the Secretary of HHS to review the merits of adding benzodiazepines and barbituates to the classes of tested drugs and of lowering cutoff levels for marijuana and amphetamines for NRC and other federal programs. The staff should keep the Commission informed of the status of the HHS review.

-~~(EDO)~~ (ADM) (SECY SUSPENSE: 9/89)

8900045

cc: Chairman Zech
Commissioner Roberts
Commissioner Carr
Commissioner Rogers
Commissioner Curtiss
OGC

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: REVISION OF 10 CFR PART 55 TO REQUIRE COMPLIANCE WITH FITNESS-FOR-DUTY PROGRAMS AND OF THE COMMISSION'S ENFORCEMENT POLICY

Purpose: To obtain the Commission's approval to publish the final rule revising § 55.53 and § 55.61 of Title 10 of the Code of Federal Regulations (10 CFR) Part 55 to require that compliance with the conditions and cutoff levels of 10 CFR Part 26, "Fitness-for-Duty Programs," is a condition of an operator or a senior operator license. In addition, to reflect appropriate enforcement sanctions for individual licensed operators, a modification of the Commission's enforcement policy, Appendix C to 10 CFR Part 2, is also described.

Background: The Commission approved SECY 89-30, "Final Rulemaking - Fitness-for-Duty Programs (Part 26)," subject to the conditions stated in a staff requirements memorandum (SRM) of March 22, 1989. In the SRM, the Commission directed the staff to prepare a notice of proposed rulemaking to amend Part 55 so that the cutoff levels established pursuant to Part 26 become applicable to licensed operators as conditions of their license. The Commission further requested the staff to amend Part 2, Appendix C, to reflect appropriate enforcement sanctions for individual licensed operators. The proposed rulemaking was published in the Federal Register on April 17, 1990 (55 FR 14288).

Discussion: In its SRM the Commission indicated that the penalty should be clearly described for violating the cutoff levels for substances described in Part 26 so that licensed operators at nuclear power and non-power facilities will have full notice of the gravity of any violation. Therefore, 10 CFR Part 2 is being modified to reflect appropriate enforcement

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David J. Lange, NRR
492-3172

sanctions for individual licensed operators who violate applicable cutoff levels. A summary of the staff's response to the SRM, indicating the changes to Part 55 and Part 2, is provided herein; the final rule, including a summary of the public comments received on the proposed rule, is provided as an enclosure.

Subpart G of 10 CFR Part 55, "Modification and Revocation of Licenses," describes the circumstances under which licenses may be modified or revoked, including willful violation of or failure to observe any of the terms or conditions of a license. Subpart H, "Enforcement," indicates that civil penalties may be imposed for violation of a license issued under Section 107, "Operators' Licenses," of the Atomic Energy Act. Therefore, making compliance with fitness-for-duty (FFD) requirements a condition of an operator's license will provide a basis for issuing a notice of violation or civil penalty to licensed operators who violate such provisions. This condition will be applicable to both power and non-power licensed operators. The final rule is intended to make the 10 CFR Part 26 cutoff levels applicable to non-power reactor licensed operators (Part 55 licensees) not to the non-power facilities (Part 50 licensees).

It is the staff's position that this amendment to § 55.53, "Conditions of Licenses," will clearly describe the obligation of the licensed operator to comply with the FFD requirements for substance use or misuse (including applicable cutoff levels) contained in 10 CFR Part 26. Further, the amendment to § 55.61, "Modification and Revocation of Licenses," will provide explicit notice of the terms or conditions, including FFD standards, under which a license may be revoked, suspended, or modified.

Beyond making the Part 26 cutoff limits enforceable conditions of Part 55 operator licenses, the final rule contains a number of other provisions for ensuring that operator performance is not adversely affected in any manner by drugs or alcohol.

<u>First</u>, Part 26 explicitly imposes sanctions for use of illegal drugs. It does not explicitly impose sanctions for alcohol abuse although it requires facility licensees to impose sanctions sufficient to deter abuse. The staff agrees with this approach for other than licensed operators. For licensed operators, the staff believes it appropriate

that NRC specify sanctions for exceeding the alcohol cutoff levels and that such sanctions be the same as those for exceeding illegal drug cutoff levels. The staff believes that alcohol abuse before or during an operator's performance of licensed duties is a significant health and safety issue because of the critical duties of the operator to diagnose plant parameters and to perform immediate actions necessary to place the reactor in a safe shutdown condition.

Second, the final rule prohibits the operator from performing licensed duties while under the influence of any substance, legal or illegal, that could adversely affect his or her ability to safely and competently perform those duties. This standard will require the operator to comply with the Part 50 facility licensee's FFD program pertaining to the use or abuse of legal or illegal drugs. It is important for NRC to establish a standard for an operator's use of legal drugs because the licensed operators may be challenged to place the reactor in a safe shutdown condition and must be mentally alert and physically able to do so.

As pointed out in the supplementary information to Part 26, "the NRC believes that a licensee's policies regarding workers' use of legal drugs and alcohol is as important for ensuring public health and safety as the licensee's policy regarding illegal drug use." The revision to Part 55 will clearly establish an FFD standard that prohibits operators and senior operators from performing licensed activities while under the influence of legal or illegal drugs. This requirement is in addition to and not necessarily related to the Part 50 licensee's obligation to inform NRC if a licensed operator develops a physical or mental condition that causes him or her to fail the medical qualification requirements of Part 55.

The Part 50 licensee has a responsibility, under its FFD program, to establish and implement written policies and procedures that address the use and abuse of prescription and over-the-counter drugs. To be consistent with this rule it is expected that a Part 50 licensee will require licensed operators (Part 55 licensees) to comply with the facility's Part 26 program for reporting uses of prescription or over-the-counter drugs to a medical review officer or supervisory personnel. The facility's written policies and procedures must be in sufficient detail to provide affected individuals with information about what is expected of them and what consequences may result from lack of adherence to the facility's policies. The final rule will require, through a condition of the operator's license, that the operator comply with the facility licensee's established requirements addressing prescription and over-the-counter drugs. If only the sale, use, or possession of illegal substances are regulated, then the standard imposed on licensed operators will be significantly lowered and the primary objective of protecting the public health and safety will be compromised. The NRC must establish an FFD standard for licensed operators that recognizes that the use or misuse of legal over-the-counter and prescription drugs could cause physical and mental impairment just as well as the use of illegal drugs can.

Third, the final rule explicitly prohibits licensed operators from engaging in the sale, possession, or use of any illegal substance whether such sale is on site or off site. Although Part 26 only provides a specific sanction for on-site sale, possession, and use, the staff believes that the specific prohibition on licensed operators against the sale, possession, or use of illegal drugs on site or off site is consistent with the stated policy requirements of Part 26, to wit: "Individuals who are not reliable and trustworthy...shall not be licensed or permitted to perform responsible health and safety functions." (See 54 FR 24493.)

Fourth, the final rule places the responsibility of fitness for duty on the Part 55 licensed operator through a condition of his or her license. The operator is to be held personally accountable for the existing Part 55 medical requirements that govern his or her physical and mental condition and for the FFD standard established by the facility licensee. This requirement is consistent with the Part 50 licensee's obligation to inform the NRC if a licensed operator develops a physical or mental condition that causes the operator to fail the medical qualification standards established in ANSI/ANS-3.4-1983 and required by 10 CFR Part 55.

In the SRM the Commission also requested that the enforcement policy (Appendix C to 10 CFR Part 2) be amended by rulemaking along with the changes to 10 CFR Part 55. The staff is modifying the enforcement policy in conjunction with final rulemaking of 10 CFR Part 55, as described in the supplementary information for the enclosed amendment.

In cases involving a licensed operator's failure to meet applicable FFD requirements (10 CFR 55.53(j)), the NRC may issue a notice of violation or a civil penalty to a licensed operator, or an order to suspend, modify or revoke the license. These actions may be taken the first time a licensed operator fails a drug or alcohol test, that is, exceeds the cutoff levels of 10 CFR Part 26 or the facility licensee's cutoff levels, if lower. In addition, the NRC will at a minimum, issue an order to suspend the Part 55 license for up to 3 years the second time the individual exceeds those cutoff levels. If there are less than 3 years remaining in the term of the individual license, the NRC may consider not renewing the individual license or issuing a new license after the 3-year period is completed. The NRC will issue an order to revoke the Part 55 license the third time an individual exceeds applicable cutoff levels. A licensed operator or applicant who refuses to participate in the drug and alcohol testing programs established by the facility licensee or who is involved in the sale, use, or possession of an illegal drug may be subject to license suspension, revocation, or denial.

To assist in determining the severity levels of potential violations, 10 CFR Part 2, Appendix C, Supplement I, is modified to provide a Severity Level I example of a licensed operator performing duties while unfit, a Severity Level II example of a licensed operator involved in the sale, use, or possession of illegal drugs within the protected area, and a Severity Level III example of a licensed operator's initial failure of a drug or alcohol test.

The staff is also modifying the enforcement policy to state that civil penalty actions against licensed operators will require approval of the Commission in accordance with the Commission's direction in the Peach Bottom case (SECY 88-201).

The Office of General Counsel has reviewed this final rule and has no legal objection.

<u>Recommendations</u>: That the Commission:

(1) <u>Approve</u> publication in the <u>Federal Register</u> of final rulemaking amending 10 CFR Part 55, Subpart F, to clearly establish a condition of an operator's license that would prohibit conduct of licensed activities while under the influence of any substance that could adversely

affect performance of licensed duties, amending 10 CFR Part 55, Subpart G, to provide explicit additional notice for the terms or conditions under which a license may be revoked, suspended or modified, and amending 10 CFR Part 2, Appendix C to reflect appropriate enforcement sanctions for individual licensed operators.

(2) Certify that this rule does not have a significant economic impact on a substantial number of small entities in order to satisfy the requirements of the Regulatory Flexibility Act (5 U.S.C. 605(b)).

(3) Note:

(a) That the final rule will become effective 30 days after publication in the Federal Register.

(b) That a regulatory analysis has been prepared for this rulemaking action (Enclosure B).

(c) That neither an environmental impact statement nor an environmental assessment and finding of no significant impact has been prepared for this final rule because it meets the criteria for a categorical exclusion under § 51.22 (c)(1).

(d) That the Subcommittee on Nuclear Regulation of the Senate Committee on Environment and Public Works and the Subcommittee on Energy and Power of the House Committee on Energy and Commerce and the House Committee on Interior and Insular Affairs will be informed of this rulemaking action (Enclosure C).

(e) That the final rule does not contain new or amended information collection requirements subject to the Paperwork Reduction Act.

(f) That the Chief Counsel for Advocacy of the Small Business Administration will be informed of the certification and the reasons for it as required by the Regulatory Flexibility Act.

(g) That a public announcement will be issued (Enclosure D).

(h) That a copy of the final rule will be distributed to all affected licensees and other interested persons.

(i) That the Advisory Committee on Reactor Safeguards has reviewed the final rule.

(j) That the Committee to Review Generic Requirements waived their review of the final rule.

James M. Taylor
Executive Director
for Operations

Enclosure:
Final Rulemaking

NUCLEAR REGULATORY COMMISSION
10 CFR PART 55
RIN 3150-AD 55
Operators' Licenses

AGENCY: Nuclear Regulatory Commission.

ACTION: Final rule.

SUMMARY: The Nuclear Regulatory Commission (NRC) is amending its regulations to specify that the conditions and cutoff levels established pursuant to the Commission's Fitness-for-Duty Programs are applicable to licensed operators as conditions of their licenses. The final rule provides a basis for taking enforcement actions against licensed operators (1) who use drugs or alcohol in a manner that would exceed the cutoff levels contained in the fitness-for-duty rule, (2) who are determined by a facility medical review officer (MRO) to be under the influence of any prescription or over-the-counter drug that could adversely affect his or her ability to safely and competently perform licensed duties, or (3) who sell, use, or possess illegal drugs. The final rule will ensure a safe operational environment for the performance of all licensed activities by providing a clear understanding to licensed operators of the severity of violating requirements governing drug and alcohol use and substance abuse.

EFFECTIVE DATE: (30 days after publication in the Federal Register)

FOR FURTHER INFORMATION CONTACT: Robert M. Gallo, Chief, Operator Licensing Branch, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Telephone: (301) 492-1031.

SUPPLEMENTARY INFORMATION:

Background

On June 7, 1989 (54 FR 24468) the NRC issued a new 10 CFR Part 26, entitled "Fitness-for-Duty Programs," to require licensees authorized to construct or operate nuclear power reactors to implement a fitness-for-duty program. The general objective of this program is to provide reasonable assurance that nuclear power plant personnel will perform their tasks in a reliable and trustworthy manner, and not under the influence of any prescription, over-the-counter, or illegal substance that in any way adversely affects their ability to safely and competently perform their duties. A fitness-for-duty program, developed under the requirements of this rule, is intended to create a work environment that is free of drugs and alcohol and the effects of the use of these substances.

On April 17, 1990, (55 FR 14288), the NRC published in the Federal Register proposed amendments to 10 CFR Part 55 to specify that the conditions and cutoff levels established in 10 CFR Part 26, "Fitness-for-Duty Programs," are applicable to licensed operators as a condition of their licenses. These amendments also provide a basis for taking enforcement action against licensed operators who violate 10 CFR Part 26. The proposed rule also described contemplated changes to the NRC enforcement policy. The comment period ended on July 2, 1990.

The Commission is adding specific conditions to operator licenses issued under 10 CFR Part 55 to make fitness-for-duty requirements directly applicable to licensed operators. As pointed out in the supplementary information accompanying the promulgation of 10 CFR Part 26, the scientific evidence shows conclusively that significant decrements in cognitive and physical performance result from the use of illicit drugs as well as from the use and misuse of prescription and over-the-counter drugs. Given the addictive and impairing nature of certain drugs, while recognizing that the presence of drug metabolites does not necessarily relate directly to a current impaired state, the presence of drug metabolites in an individual's system strongly suggests the likelihood of past, present, or future impairment affecting job activities. More specifically, the Commission stated, "Individuals who are not reliable and trustworthy, under the influence of any substance, or mentally or physically impaired in any way that adversely affects their ability to safely and competently perform their duties, shall not be licensed or permitted to perform responsible health and safety functions." (See 54 FR 24468, June 7, 1989.) Although there is an underlying assumption that operators will abide by the licensees' policies and procedures, any involvement with illegal drugs, whether on site or off site, indicates that the operator cannot be relied upon to obey the law and therefore may not scrupulously follow rigorous procedural requirements with the integrity required to ensure public health and safety in the nuclear power industry.

The Commission believes strongly that licensed operators are a critical factor in ensuring the safe operation of the facility and consequently considers unimpaired job performance by each licensed operator or senior operator vital in ensuring safe facility operation. The NRC routinely denies Part 55 license applications or conditions operator and senior operator licenses if the applicant's medical condition and general health do not meet the minimum standards required for the safe performance of assigned job duties. Further, under § 55.25, if an operator develops, during the term of his or her license, a physical or mental condition that causes the operator to fail to meet the requirements for medical fitness, the facility licensee is required to notify the NRC. Any such condition may result in the operator's license being modified, suspended, or revoked.

The power reactor facility licensee is further required under § 26.20(a) to have written policies and procedures that address fitness-for-duty requirements on abuse of prescription and over-the-counter drugs and on other factors such as mental stress, fatigue, and illness that could affect fitness for duty. The Commission expects each licensed operator or senior operator to follow the licensee's written policies and procedures concerning the use and reporting requirements for prescription and over-the-counter drugs and other factors that the facility has determined could affect fitness for duty.

The use of alcohol and drugs can directly impair job performance. Other causes of impairment include use of prescription and over-the-counter medications, emotional and mental stress, fatigue, illness, and physical and psychological impairments. The effects of alcohol, which is a drug, are well known and documented and, therefore, are not repeated here. Drugs such as marijuana, sedatives, hallucinogens, and high doses of stimulants could adversely affect an employee's ability to correctly judge situations and make decisions (NUREG/CR-3196, "Drug and Alcohol Abuse: The Bases for Employee Assistance Programs in the Nuclear Industry," available from the National Technical Information Service). The greatest impairment occurs shortly after use or abuse, and the negative short-term effects on human performance (including subtle or marginal impairments that are difficult for a supervisor to detect) can last for several hours or days. The amendment to 10 CFR Part 55 will establish a condition of an operator's license that will prohibit conduct of licensed duties while under the influence of alcohol or any prescription, over-the-counter or illegal substance that would adversely affect performance of licensed duties as described by the facility's fitness-for-duty program. The amendment will be applicable to licensed operators of power and non-power reactors. This rulemaking is not intended to apply the provisions of 10 CFR Part 26 to non-power facility licensees, but to make it clear to all licensed operators (power and non-power) through conditions of their licenses that the use of drugs or alcohol in any manner that could adversely affect performance of licensed duties would subject them to enforcement action.

As explained in the Commission's enforcement policy (see 53 FR 40027, October 13, 1988), the Commission may take enforcement action if the conduct of an individual places in question the NRC's reasonable assurance that licensed activities will be properly conducted. The Commission may take enforcement action for reasons that would warrant refusal to issue a license on an original application. Accordingly, enforcement action may be taken regarding matters that raise issues of trustworthiness, reliability, use of sound judgment, integrity, competence, fitness for duty, or other matters that may not necessarily be a violation of specific Commission requirements.

The Commission is amending §55.53 to establish as a condition of an operator's license a provision precluding performance of licensed duties while under the influence of drugs or alcohol in any manner that could adversely affect performance. The Commission further amends § 55.61 to provide explicit additional notice of the terms and conditions under which an operator's license may be revoked, suspended, or modified. In addition, positive test results and failures to participate in drug and alcohol testing programs will be considered in making decisions concerning renewal of a Part 55 license. These provisions will apply to any fitness-for-duty program established by a facility licensee, whether or not required by Commission regulations, including programs that establish cutoff levels below those set by 10 CFR Part 26, Appendix A. The Commission notes, however, that it has the discretion to forego enforcement action against a licensed operator if the facility licensee established cutoff levels that are so low as to be unreasonable in terms of the uncertainties of testing. The Commission has reserved the right to review facility licensee programs against the performance objectives of 10 CFR Part 26, which require reasonable detection measures. The revised rule will not impose the provisions of 10 CFR Part 26 on non-power facility licensees. It is revised to make compliance with the cutoff levels and the policy and procedures regarding the use of legal and illegal drugs established pursuant to 10 CFR Part 26 a license condition for all holders of a 10 CFR Part 55 license.

Part 26 requires that facility licensees provide appropriate training to licensed operators, among others, to ensure that they understand the effect of prescription and over-the-counter drugs and dietary conditions on job performance and on chemical test results. The training also should indicate information about the roles of supervisors and the medical review officer in reporting an operator's current use of over-the-counter drugs or prescription drugs that may impair his or her performance. Licensed operators are required to follow their facility's policies and procedures regarding fitness-for-duty requirements.

Licensed operators will be subject to notices of violation, civil penalties, or orders for violation of their facility licensee's fitness-for-duty requirements. Therefore, in addition to amending the regulations to establish the 10 CFR Part 55 licensed operators' obligations, the Commission is modifying the NRC enforcement policy (Appendix C to 10 CFR Part 2) in conjunction with the final rulemaking as described below.

In cases involving a licensed operator's failure to meet applicable fitness-for-duty requirements (10 CFR 55.53(j)), the NRC may issue a notice of violation or a civil penalty to a licensed operator, or an order to suspend, modify or revoke the license. These actions may be taken the first time a licensed operator fails a drug or alcohol test, that is, exceeds the cutoff levels of 10 CFR Part 26 or the facility licensee's cutoff levels, if lower. In addition, the NRC will, at a minimum, issue an order to suspend the Part 55 license for up to three years the second time an individual exceeds those cutoff levels. If there are less than three years remaining in the term of the individual license, the NRC may consider not renewing the individual license or issuing a new license until the three-year period is completed. The NRC will issue an order to revoke the Part 55 license the third time an individual exceeds those cutoff levels. A licensed operator or applicant who refuses to participate in the drug and alcohol testing programs established by the facility licensee or who is involved in the sale, use, or possession of an illegal drug may be subject to license suspension, revocation, or denial.

To assist in determining the severity levels of potential violations, 10 CFR Part 2, Appendix C, Supplement I is modified to provide a Severity Level I example of a licensed operator performing duties while unfit, a Severity Level II example of a licensed operator involved in the sale, use, or possession of illegal drugs within the protected area, and a Severity Level III example of a licensed operator's initial failure of a drug or alcohol test.

Summary of Public Comments

Letters of comment were received from 39 respondents. One commenter wrote two letters, which brought the total number of responses to 40. Thirty-one of the commenters wrote that the rule is unnecessary because the regulations already exist to ensure that the reactor operators adhere to 10 CFR Part 26. The Commission agrees that the necessary regulations exist to have licensed operators comply with the provisions of Part 26. However, the Commission realizes that the licensed operator is one of the main components and possibly the most critical component of continued safe reactor operation. Therefore, it wants to emphasize to and clearly inform the operators that as conditions of their licenses they must comply with their facility's fitness-for-duty program. The Commission also wants to clarify the term "use" versus "consumption" of alcohol in protected reactor areas. The rule has been rewritten to indicate that the "use of alcohol" means consumption of alcoholic beverages. The rule does not prohibit the use of alcohol within the protected areas for other than ingestion, such as application to the body. The use of medicine that contains alcohol is allowed within the parameters of the facility's fitness-for-duty program. However, use of over-the-counter or prescription drugs containing alcohol must be within the prescribed limitations and in compliance with the facility's fitness-for-duty program.

Twenty-eight of the commenters wrote that this rule singles out licensed operators for special treatment to the detriment of their morale. This rule stresses to licensed operators that because of their critical role in the safe operation of their reactors, they must be singled out for special treatment to stress that their continuous unimpaired job performance is a highly necessary component of the overall safe operation of the reactors. The rule also stresses to licensed operators that their licenses are a privilege and not a right, and that refusal to participate in facility fitness-for-duty requirements can lead to enforcement action and/or licensing action. There has been no change to the rulemaking because of these comments.

Twenty commenters stated that it is an unnecessary burden that the proposed rule requires medical personnel to be available 24 hours a day to make judgments about prescription and over-the-counter drugs. Medical personnel are not required by Part 26 or Part 55 to be on duty 24 hours a day for prescription and over-the-counter drug evaluation. The intent of the rule is that licensed operators follow the facility fitness-for-duty program for supervisory notification of fitness-for-duty concerns about the use of legal drugs. The rulemaking has been clarified to more fully explain this intent.

There were two questions about the basis for the rulemaking -- (1) What is the basis or need for the rule change? (2) Is it an industrywide problem? First, the basis for the rule change was discussed above under the need for the rule (regulations already exist). Secondly, there is currently no conclusive evidence of an industrywide drug problem. However, the Commission can have nothing but a zero tolerance level for drug and alcohol use or abuse because of the critical nature of the industry. Therefore, the Commission deemed it necessary to stress compliance with facility fitness-for-duty programs as a condition of licensure. There is no change to the rulemaking as a result of these comments.

There was one question about the reporting of legal drugs. A licensed operator asked how operators who do not report medicinal use of drugs will be treated. Licensed operators are required to follow the fitness-for-duty program procedures and policies developed by their facility.

Two issues were specific to licensed operators at test and research reactor facilities. One was that formal drug testing programs should not be required for non-power facilities. These programs are not required by Part 26 or Part 55; however, if a fitness-for-duty program has been established at a non-power facility, licensed operators are required to participate. The second issue, regarding over-the-counter and prescription medication, was that medical review officers do not exist at non-power facilities. That statement is true; there are no requirements in either Part 26 or Part 55 that they do. No change to the rulemaking was required as a result of these comments.

Environmental Impact: Categorical Exclusion

The NRC has determined that this final rule is the type of action described in categorical exclusion 10 CFR 51.22(c)(1). Therefore, neither an environmental impact statement nor an environmental assessment has been prepared for this rule.

Paperwork Reduction Act Statement

This rule contains no information collection requirements and, therefore, is not subject to the requirements of the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.).

Regulatory Analysis

The regulations in 10 CFR Part 55 establish procedures and criteria for the issuance of licenses to operators and senior operators of utilization facilities licensed pursuant to the Atomic Energy Act of 1954, as amended, or Section 202 of the Energy Reorganization Act of 1974, as amended, and 10 CFR Part 50. These established procedures provide the terms and conditions upon which the Commission will issue, modify, maintain, and renew operator and senior operator licenses.

Subpart F of Part 55, under §55.53, "Conditions of Licenses," sets forth the requirements and conditions for the maintenance of operator and senior operator licenses.

This rule serves to emphasize to the holders of operator and senior operator licenses the conditions they are required to comply with under 10 CFR Part 26, "Fitness-for-Duty Program." A regulatory analysis has been prepared for the final rule resulting in the promulgation of Part 26 and is available for inspection in the NRC Public Document Room, 2120 L Street NW (Lower Level), Washington, D.C. This analysis examines the costs and benefits of the alternatives considered by the Commission for compliance with the conditions and cutoff levels. The Commission previously requested public comment on the regulatory analysis as part of the rulemaking proceeding that resulted in the adoption of Part 26.

Regulatory Flexibility Certification

In accordance with the Regulatory Flexibility Act, 5 U.S.C. 605(b), the NRC certifies that this rule will not have a significant economic effect on a substantial number of small entities. Many applicants or holders of operator licenses fall within the definition of small businesses found in Section 34 of the Small Business Act (15 U.S.C. 632) or the Small Business Size Standards set out in regulations issued by the Small Business Administration at 13 CFR Part 121 or the NRC's size standards published December 9, 1985 (50 FR 50241). However, the rule will only serve to provide notice to licensed individuals of the conditions under which they are expected to perform their licensed duties.

Backfit Analysis

The NRC has determined that the backfit rule, 10 CFR 50.109, does not apply to this final rule, and therefore, that a backfit analysis is not required for this rule because these amendments do not involve any provisions that would impose backfits as defined in 10 CFR 50.109(a)(1).

List of Subjects in 10 CFR Parts 2 and 55

Part 2 - Administrative practice and procedure, Antitrust, Byproduct material, Classified information, Civil penalty, Enforcement, Environmental protection, Nuclear materials, Nuclear power plants and reactors, Penalty, Sex discrimination, Source material, Special nuclear material, Violations, Waste treatment and disposal.

Part 55 - Criminal penalty, Manpower training programs, Nuclear power plants and reactors, Reporting and recordkeeping requirements.

For the reasons set out in the preamble and under the authority of the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974, as amended, and 5 U.S.C. 552 and 553, the NRC is proposing to adopt the following amendments to 10 CFR Part 2 and 10 CFR Part 55.

PART 2 - RULES OF PRACTICE FOR DOMESTIC LICENSING PROCEEDINGS

1. The authority citation for Part 2 continues to read as follows:

AUTHORITY: Secs. 161,181,68 Stat. 948, 953, as amended (42 U.S.C 2201, 2231); sec. 191, as amended, Pub. L. 87-615, 76 Stat. 409 (42 U.S.C. 2241); sec. 201, 88 Stat. 1242, as amended (42 U.S.C. 5841); 5. U.S.C. 552.

Section 2.101 also issued under secs. 53, 62, 63, 81, 103, 104, 105, 68 Stat. 930, 932, 933, 935, 936, 937, 938, as amended (42 U.S.C. 2073, 2092, 2093, 2111, 2133, 2134, 2135); sec. 114(f), Pub. L. 97-425, 96 Stat. 2213, as amended (42 U.S.C. 10134(f)); sec. 102, Pub. L. 91-190, 83 Stat. 853, as amended (42 U.S.C. 4332); sec. 301, 88 Stat. 1248 (42 U.S.C. 5871). Sections 2.102, 2.103, 2.104, 2.105, 2.721 also issued under secs. 102, 103, 104, 105, 183, 189, 68 Stat. 936, 937, 938, 954, 955, as amended (42 U.S.C. 2132, 2133, 2134, 2135, 2233, 2239). Section 2.105 also issued under Pub. L. 97-415, 96 Stat. 2073 (42 U.S.C. 2239). Sections 2.200-2.206 also issued under secs. 186, 234, 68 Stat. 955, 83 Stat. 444, as amended (42 U.S.C. 2236, 2282); sec. 206, 88 Stat. 1246 (42 U.S.C. 5846). Sections 2.600-2.606 also issued under sec. 102, Pub. L. 91-190, 83 Stat. 853, as amended (42 U.S.C. 4332). Sections 2.700a, 2.719 also issued under 5 U.S.C. 554. Sections 2.754, 2.760, 2.770, 2.780 also issued under 5 U.S.C. 557. Section 2.764 and Table 1A of Appendix C also issued under secs. 135, 141, Pub. L. 97-425, 96 Stat. 2232, 2241 (42 U.S.C. 10155, 10161). Section 2.790 also issued under sec. 103, 68 Stat. 936, as amended (42 U.S.C. 2133) and 5 U.S.C. 552. Sections 2.800 and 2.808 also issued under 5 U.S.C. 553. Section 2.809 also issued under 5 U.S.C. 553 and sec. 29, Pub. L. 85-256, 71 Stat. 579 as amended (42 U.S.C. 2039). Subpart K also issued under sec. 189, 68 Stat. 955 (42 U.S.C. 2239); sec. 134, Pub. L. 97-425, 96 Stat. 2230 (42 U.S.C. 10154). Subpart L also issued under sec. [illegible], 68 Stat. 955 (42 U.S.C. 2239). Appendix A also issued under sec. 6, Pub. L. 91-560, 84 Stat. 1473 (42 U.S.C. 2135). Appendix B also issued under sec. 10, Pub. L. 99-240, 99 Stat. 1842 (42 U.S.C. 2021b et seq.).

2. Appendix C to 10 CFR Part 2 is amended by --
   a. Adding an undesignated paragraph at the end of Section V. E,
   b. Adding paragraph (8) to Section VIII, and
   c. Adding paragraph A. 5., B. 3., and C. 9 to Supplement I to read as follows:

Appendix C - General Statement of Policy and Procedure for NRC Enforcement Actions

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V. Enforcement Actions

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E. <u>Enforcement Actions Involving Individuals</u>

In the case of a licensed operator's failure to meet applicable fitness-for-duty requirements (10 CFR 55.53(j)), the NRC may issue a notice of violation or a civil penalty to the Part 55 licensee, or an order to suspend, modify or revoke the license. These actions may be taken the first time a licensed operator fails a drug or alcohol test, that is, exceeds the cutoff levels of 10 CFR Part 26 or the facility licensee's cutoff levels, if lower. In addition, the NRC will at a minimum, issue an order to suspend the Part 55 license for up to three years the second time a licensed operator exceeds those cutoff levels. In the event there are less than three years remaining in the term of the individual's license, the NRC may consider not renewing the individual's license or issuing a new license after the three year period is completed. The NRC will issue an order to revoke the Part 55 license the third time a licensed operator exceeds those cutoff levels. A licensed operator or applicant who refuses to participate in the drug and alcohol testing programs established by the facility licensee or who is involved in the sale, use, or possession of an illegal drug may be subject to license suspension, revocation, or denial. For the purposes of applying the examples in Appendix C, Supplement I, a licensed operator is considered unfit for duty when she or he has exceeded the cutoff levels established by the utility's fitness-for-duty program and is clearly not able to perform assigned duties because of alcohol or drug use.

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VIII. Responsibilities

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(8) Any proposed enforcement action involving a civil penalty to a licensed operator.

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Supplement I - Severity Categories

A. Severity I * * *

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5. A licensed operator performing duties while unfit for duty.

B. Severity II * * *

3. A licensed operator involved in the use, sale, or possession of illegal drugs or alcohol within the protected area.

C. Severity III * * *

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9. A licensed operator's failure of a drug or alcohol test.

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PART 55 - OPERATORS' LICENSES

3. The authority citation for Part 55 is revised to read as follows:

AUTHORITY: Secs. 107, 161, 182, 68 Stat. 939, 948 953, as amended, sec. 234, 83 Stat. 444, as amended (42 U.S.C. 2137, 2201, 2232, 2282); secs. 201, as amended, 202, 88 Stat. 1242, as amended, 1244 (42 U.S.C. 5841, 5842).

Sections 55.41, 55.43, 55.45 and 55.59 also issued under sec. 306, Pub. L. 97-425, 96 Stat. 2262 (42 U.S.C. 10226). Section 55.61 also issued under secs. 186, 187, 68 Stat. 955 (42 U.S.C. 2236, 2237).

For the purposes of sec. 223, 68 Stat. 958, as amended (42 U.S.C. 2273); §§ 55.3, 55.21, 55.49 and 55.53 are issued under sec. 161i, 68 Stat. 949, as amended (42 U.S.C. 2201(i)); and §§ 55.9, 55.23, 55.25, and 55.53(f) are issued under sec. 161o, 68 Stat. 950, as amended (42 U.S.C. 2201(o)).

4. In § 55.53, paragraph (j) is redesignated as paragraph (l) and new paragraphs (j) and (k) are added to read as follows:

§ 55.53 Conditions of licenses.

* * * * *

(j) The licensee shall not consume or ingest alcoholic beverages within the protected area of power reactors, or the controlled access area of non-power reactors. The licensee shall not use, possess, or sell any illegal drugs. The licensee shall not perform activities authorized by a license issued under this part while under the influence of alcohol or any prescription, over-the-counter, or illegal substance that could adversely affect his or her ability to safely and competently perform his or her licensed duties. For the purpose of this paragraph, with respect to alcoholic beverages and illegal drugs, the term "under the influence" means the licensee exceeded the lower of the cutoff levels for drugs or alcohol contained in 10 CFR Part 26, Appendix A, of this chapter, or as established by the facility licensee. With respect to prescription and over-the-counter drugs, the term "under the influence" means the licensee could be mentally or physically impaired, as determined by a medical review officer or supervisor if there is no medical officer available, in such a manner as to adversely affect his or her ability to safely and competently perform licensed duties.

(k) Each licensee at power reactors shall participate in the drug and alcohol testing programs established pursuant to 10 CFR Part 26. Each licensee at non-power reactors shall participate in any drug and alcohol testing program that may be established for that non-power facility.

* * * * *

3. In § 55.61, a new paragraph (b)(5) is added to read as follows:

§ 55.61 Modification and revocation of licenses.

* * * * *

(b) * * *

(5) For the sale, use or possession of illegal drugs, or refusal to participate in the facility drug and alcohol testing program, or a confirmed positive test for drugs, drug metabolites, or alcohol in violation of the conditions and cutoff levels established by § 55.53(j) or the consumption of alcoholic beverages within the protected area of power reactors or the controlled access area of non-power reactors, or a determination of unfitness for scheduled work as a result of the consumption of alcoholic beverages.

Dated at Rockville, Maryland, this ______________ day of ____________, 1991.

For the Nuclear Regulatory Commission,

____

Samuel J. Chilk,
Secretary of the Commission.