NUREG-1022

# Licensee Event Report System
# Description of System and
# Guidelines for Reporting

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# Licensee Event Report System

## Description of System and Guidelines for Reporting

**U.S. Nuclear Regulatory Commission**

**Office for Analysis and Evaluation of Operational Data**

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# Licensee Event Report System

Description of System and
Guidelines for Reporting

Manuscript Completed: September 1983
Date Published: September 1983

Office for Analysis and Evaluation of Operational Data
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

## ABSTRACT

On July 26, 1983, the Commission published in the Federal Register a final rule (10 CFR 50.73) that modifies and codifies the Licensee Event Report (LER) system. The rule becomes effective on January 1, 1984. This NUREG provides supporting information and guidance that will be of interest to persons responsible for the preparation and review of LERs. The information contained in this NUREG includes: (1) a brief description of how LERs are analyzed by the NRC, (2) a restatement of the guidance contained in the Statement of Consideration that accompanied the publication of the LER rule, (3) a set of examples of potentially reportable events with staff comments on the actual reportability of each event, (4) guidance on how to prepare an LER, including the LER forms, and (5) guidance on submittal of LERs.

TABLE OF CONTENTS





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## I. INTRODUCTION

On July 26, 1983, the Commission published in the Federal Register a final rule (10 CFR 50.73) that modifies and codifies the Licensee Event Report (LER) system. The rule becomes effective on January 1, 1984.

This NUREG provides considerable information that will be of interest to persons responsible for the preparation and review of LERs. In addition, since the reporting criteria in 10 CFR 50.72, "Immediate Notification Requirements of Significant Events at Operating Nuclear Power Plants," and 10 CFR 50.73 are, in most cases either identical or very similar, this NUREG is also useful for clarifying the types of events that require immediate notification of the NRC in accordance with 10 CFR 50.72. The information contained in this NUREG includes:

1. The LER rule (10 CFR 50.73) (Section II).

2. A brief description of how LERs are analyzed by the NRC (Section III).

3. A restatement (Sections IV and V) of the guidance contained in the Statement of Consideration that accompanied the publication of the LER rule. This guidance explains the intent of the various criteria and requirements contained in the LER rule.

4. General and specific guidance (Section VI) on how to prepare an LER, including the LER forms (Appendix A). Tables of some of the codes needed to complete the LER form are also included in Appendix B.

5. A set of examples (Appendix C) of potentially reportable events with staff comments on the actual reportability of each event. The descriptions contained in Appendix C have been taken from actual operational events; however, reference to the plant at which the event occurred has been removed and on occasion the description of the actual event has been altered slightly to illustrate a specific point.

Background

In December 1980, the Commission decided that the requirements for the reporting of operational experience data needed major revision and approved the development of an Integrated Operational Experience Reporting (IOER) system. The IOER system would have combined, modified, and made mandatory the existing Licensee Event Report (LER) system and the industry supported, voluntary Nuclear Plant Reliability Data (NPRD) System.

As a result of the Commission's approval of the concept of an IOER system, the NRC published an advance notice of proposed rulemaking on January 15, 1981 (46 FR 3541). This advance notice explained why the NRC needed operational experience data and described the deficiencies in the existing LER and NPRD systems.

On June 8, 1981, the Institute of Nuclear Power Operations (INPO) announced that because of its role as an active user of NPRDS data it would assume responsibility for management and funding of the NPRD system. Further, INPO decided to develop criteria that would be used in its management audits of member utilities to assess the adequacy of participation in the NPRD system.

Since there was a likelihood that the NPRD system under INPO direction would meet the NRC's need for reliability data, it was no longer necessary to proceed with the IOERS. Hence, the collection of detailed technical descriptions of significant events could be addressed in a separate rulemaking to modify and codify the existing LER reporting requirements.

However, the Commission made it explicitly clear that it has modified the scope of the LER reporting requirements with the expectation that sufficient utility participation, cooperation, and support of the NPRD system will be forthcoming. If the NPRD system does not become operational at a satisfactory level in a reasonable time, remedial action by the Commission in the form of additional rulemaking may become necessary.

On October 6, 1981, the NRC published an advance notice (46 FR 49134) that deferred development of the IOER system and sought public comment on the scope and content of the LER system.

On May 6, 1982, the NRC published in the Federal Register (47 FR 19543) a Notice of Proposed Rulemaking that would modify and codify the existing LER system. Interested persons were invited to submit written comments to the Secretary of the Commission by July 6, 1982. Numerous comments were received. After consideration of the comments and other factors involved, the Commission amended the proposed requirements published for public comment by clarifying the scope and content of the requirements, particularly the criteria that define which operational events must be reported.

The LER rule identifies the types of reactor events and problems that are believed to be significant and useful to the NRC in its effort to identify and resolve threats to public safety. It is designed to provide the information necessary for engineering studies of operational anomalies and trends and patterns analysis of operational occurrences.

The Commission believes that the NRC should continue to seek improved operational data methods and systems that will maximize the value of operational data. Thus, improvements will continue to be sought in the reporting, assessment, and feedback of operational data of events and problem sequences identified in this rule, NPRDS data, and such other information as appropriate.

## II. THE LER RULE (10 CFR 50.73)

§50.73 "Licensee Event Report System" states:

(a) <u>Reportable events</u>

(1) The holder of an operating license for a nuclear power plant (licensee) shall submit a Licensee Event Report (LER) for any event of the type described in this paragraph within 30 days after the discovery of the event. Unless otherwise specified in this section, the licensee shall report an event regardless of the plant mode or power level, and regardless of the significance of the structure, system, or component that initiated the event.

(2) The licensee shall report:

(i) (A) The completion of any nuclear plant shutdown required by the plant's Technical Specifications; or

(B) Any operation or condition prohibited by the plant's Technical Specifications; or

(C) Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x) of this part.

(ii) Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear power plant being:

(A) In an unanalyzed condition that significantly compromised plant safety;

(B) In a condition that was outside the design basis of the plant; or

(C) In a condition not covered by the plant's operating and emergency procedures.

(iii) Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant.

(iv) Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). [illegible] actuation of an ESF, including the RPS, that resulted [illegible] part of the preplanned sequ[illegible] operation need not be reported[illegible]

(v) Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

(vi) Events covered in paragraph (a)(2)(v) of this section may include one or more procedural errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to this paragraph if redundant equipment in the same system was operable and available to perform the required safety function.

(vii) Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a single system designed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

(viii) (A) Any airborne radioactivity release that exceeded 2 times the applicable concentrations of the limits specified in Appendix B, Table II of Part 20 of this chapter in unrestricted areas, when averaged over a time period of one hour.

(B) Any liquid effluent release that exceeded 2 times the limiting combined Maximum Permissible Concentration (MPC) (see Note 1 of Appendix B to Part 20 of this chapter) at the point of entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour.

(ix) Reports submitted to the Commission in accordance with paragraph (a)(2)(viii) of this section also meet the effluent release reporting requirements of paragraph 20.405(a)(5) of Part 20 of this chapter.

(x) Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases.

(b) Contents. The Licensee Event Report shall contain:

(1) A brief abstract describing the major occurrences during the event, including all component or system failures that contributed to the event and significant corrective action taken or planned to prevent recurrence.

(2) (i) A clear, specific, narrative description of what occurred so that knowledgeable readers conversant with the design of commercial nuclear power plants, but not familiar with the details of a particular plant, can understand the complete event.

(ii) The narrative description must include the following specific information as appropriate for the particular event:

(A) Plant operating conditions before the event.

(B) Status of structures, components, or systems that were inoperable at the start of the event and that contributed to the event.

(C) Dates and approximate times of occurrences.

(D) The cause of each component or system failure or personnel error, if known.

(E) The failure mode, mechanism, and effect of each failed component, if known.

(F) The Energy Industry Identification System component function identifier and system name of each component or system referred to in the LER.

(1) The Energy Industry Identification System is defined in: IEEE Std 803-1983 (May 16, 1983) Recommended Practices for Unique Identification Plants and Related Facilities - Principles and Definitions.

(2) IEEE Std 803-1983 has been approved for incorporation by reference by the Director of the Federal Register. A notice of any changes made to the material incorporate by reference will be published in the Federal Register. Copies may be obtained from the Institute of Electrical and Electronics Engineers, 345 East 47th Street, New York, NY 10017. A copy is available for inspection and copying for a fee at the Commission's Public Document Room, 1717 H Street, NW, Washington, D.C. and at the Office of the Federal Register, 1100 L St. NW, Washington, D.C.

(G) For failures of components with multiple functions, include a list of systems or secondary functions that were also affected.

(H) For failure that rendered a train of a safety system inoperable, an estimate of the elapsed time from the discovery of the failure until the train was returned to service.

(I) The method of discovery of each component or system failure or procedural error.

(J) (1) Operator actions that affected the course of the event, including operator errors, procedural deficiencies, or both, that contributed to the event.

(2) For each personnel error, the licensee shall discuss:

(i) Whether the error was a cognitive error (e.g., failure to recognize the actual plant condition, failure to realize which systems should be functioning, failure to recognize the true nature of the event) or a procedural error;

(ii) Whether the error was contrary to an approved procedure, was a direct result of an error in an approved procedure, or was associated with an activity or task that was not covered by an approved procedure;

(iii) Any unusual characteristics of the work location (e.g., heat, noise) that directly contributed to the error; and

(iv) The type of personnel involved (i.e., contractor personnel, utility-licensed operator, utility nonlicensed operator, other utility personnel).

(K) Automatically and manually initiated safety system responses.

(L) The manufacturer and model number (or other identification) of each component that failed during the event.

(3) An assessment of the safety consequences and implications of the event. This assessment must include the availability of other systems or components that could have performed the same function as the components and systems that failed during the event.

(4) A description of any corrective actions planned as a result of the event, including those to reduce the probability of similar events occurring in the future.

(5) Reference to any previous similar events at the same plant that are known to the licensee.

(6) The name and telephone number of a person within the licensee's organization who is knowledgeable about the event and can provide additional information concerning the event and the plant's characteristics.

(c) Supplemental information. The Commission may require the licensee to submit specific additional information beyond that required by paragraph (b) of this section if the Commission finds that supplemental material is necessary for complete understanding of an unusually complex or significant event. These requests for supplemental information will be made in writing and the licensee shall submit the requested information as a supplement to the initial LER.

(d) Submission of reports. Licensee Event Reports must be prepared on Form NRC 366 and submitted within 30 days of discovery of a reportable event or situation to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555. The licensee shall also submit an additional copy to the appropriate NRC Regional Office listed in Appendix A to Part 73 of this chapter.

(e) Report legibility. The reports and copies that licensees are required to submit to the Commission under the provisions of this section must be of sufficient quality to permit legible reproduction and micrographic processing.

(f) Exemptions. Upon written request from a licensee including adequate justification or at the initiation of the NRC staff, the NRC Executive Director for Operations may, by a letter to the licensee, grant exemptions to the reporting requirements under this section.

(g) Reportable occurrences. The requirements contained in this section replace all existing requirements for licensees to report "Reportable Occurrences" as defined in individual plant Technical Specifications.

## III. LER ANALYSIS AND EVALUATION PROGRAM

The NRC, and particularly the Office for Analysis and Evaluation of Operational Data (AEOD) manually screens each LER (about 4,500 in 1982) to identify those individual events or generic situations that warrant additional analysis and evaluation. From this screening process, NRC determines (a) whether a special study should be initiated, (b) whether the event meets the criteria for reporting as an Abnormal Occurrence Report to Congress or for reporting to the European Nuclear Energy Agency (NEA), and (d) whether the event should be included in the bimonthly publication, Power Reactor Events.

There are two fundamental objectives associated with the LER analysis process. First, to identify and isolate "precursor events" and secondly to identify emerging trends or patterns of potential safety significance. A precursor is considered to be an event that could have been serious if plant conditions, personnel action, or the extent of equipment failure or faulting had been slightly different. Such events are evaluated individually and corrective action is initiated, where appropriate.

The NRC staff also reviews the operational experience to identify repetitive events and failures (i.e., trend and pattern analysis). It attempts to identify situations where the frequency or the combined significance of reported events may be cause for concern. If such a situation is identified, past operating history is searched for similar events and a generic study is initiated to focus upon the nature, cause, consequences and possible corrective actions for the particular situation or concern.

This trends and pattern analysis usually applies to incidents of low individual significance for which repetition or, more accurately, frequency is the element which lends significance.

Each incident has certain classifications associated with it, which if all are specified, make it unique. For example, an incident can be described as:

> "A failure of a Target Rock Relief Valve to remain closed at Brunswick 1 due to pilot valve leakage on April 19, 1983."

Any subset of the underlined items (i.e., classifications) can be used to specify what is meant by a generic situation, and then how this situation is distributed across the unspecified categories (e.g., failures of Target Rock relief valves to remain closed distributed across plants) can be examined.

There are a number of ways that a trend or pattern is identified. In its simplest form, identification of a pattern or trend originates with a single engineer reviewing an individual LER. This occurs as a result of (1) the engineer reading the description of an event and recalling from memory similar events in other reports, and/or the LER identifying previous occurrences. In either of these instances, the engineer reviews the additional reports to ascertain the true extent of the pattern or trend.

Another way to identify a pattern or trend is by the a priori postulation of a concern. This concern could be entirely hypothetical (e.g., "I wonder what the experience has been with Target Rock relief valves") or could be based on nonspecific recall of information reviewed over a period of time (e.g., "It seems to me that we've seen a lot of failures of Target Rock relief valves recently"). The engineer collects and reviews data to identify the events where the concern has occurred.

When the engineer has identified all the occurrences of interest and understands the surrounding circumstances, he/she then evaluates the safety significance of the pattern or trend. This determination will involve (1) determination of where this type of occurrence could happen in the future (i.e., the "generic" nature of a problem); (2) comparison of the frequency of the occurrence against some standard; and (3) consideration of the potential impact of each occurrence. As noted above, the results of this analysis and evaluation are reported in generic reports and are used as input to various NRC programs (e.g., the Systematic Assessment of Licensee Performance (SALP) Program).

A primary objective of a more statistical trends and patterns program is to establish a review process which is not dependent on the prior formulation of a particular concern. Rather it is driven by the data, allowing the data to point to imbalances, non-uniformities, and to increasing frequency of occurrence which are then investigated more closely. Such a review is conducted at periodic intervals, determined only by the rate at which data are accumulated.

To accomplish this objective requires a computer data base which permits consistent retrieval of data because of the large amount of data to be looked at simultaneously. The development and implementation of the Sequence Coding and Search System (SCSS) was undertaken in part to satisfy this requirement. SCSS will, for the first time, allow the LER information to be stored, coded and retrieved in a satisfactory manner for a statistically-based pattern and trend analysis.

The NRC has recently initiated a program to perform a more statistical analysis of trends and patterns. The initial results of this analysis will be published in the near future and will be incorporated into various NRC programs (e.g., SALP).

## IV. OVERVIEW OF THE LER SYSTEM

The LER includes a detailed narrative description of potentially significant safety events. By describing in detail the event and the planned corrective action, the LER will provide the basis for the careful study of events or conditions that might lead to serious accidents. If the NRC staff decides that the event is especially significant from the standpoint of safety, the staff may request that the licensee provide additional information and data associated with the event.

The licensee will prepare an LER for those events or conditions that meet one or more of the criteria contained in §50.73(a). The criteria are based primarily on the nature, course, and consequences of the event. Therefore, the LER rule requires that events which meet the criteria be reported regardless of the plant operating mode or power level, and regardless of the safety significance of the components, systems, or structures involved. In trying to develop criteria for the identification of events reportable as LERs, the Commission has concentrated on the potential consequences of the event as the measure of significance. Therefore, the reporting criteria, in general, do not specifically address classes of initiating events or causes of the event. For example, there is no requirement that all personnel errors be reported. However, many reportable events will involve or will have been initiated by personnel errors.

Finally, licensees are permitted and encouraged to report any event that does not meet the criteria contained in §50.73(a), if the licensee believes that the event might be of safety significance, might be of generic interest or concern, or contains a lesson to be learned. Reporting requirements aside, assurance of safe operation of all plants depends on accurate and complete reporting by each licensee of all events having potential safety significance.

## V. PARAGRAPH-BY-PARAGRAPH EXPLANATION OF THE LER RULE

The significant provisions of the LER rule are explained below. In addition, specific examples of potentially reportable events are described and discussed in Appendix C.

Paragraph 50.73(a)(2)(i) requires reporting of:

"(A) The completion of any nuclear plant shutdown required by the plant's Technical Specifications; or

(B) Any operation prohibited by the plant's Technical Specifications; or

(C) Any deviation from the plant's Technical Specifications authorized pursuant to §50.54(x) of this part."

This paragraph requires events to be reported where the licensee is required to shut down the plant because the requirements of the Technical Specifications were not met. For the purpose of this paragraph, "shutdown" is defined as the point in time where the Technical Specifications require that the plant be in the first shutdown condition required by a Limiting Condition for Operation [e.g., hot standby (Mode 3) for PWRs with the Standard Technical Specifications]. If the condition is corrected before the time limit for being shutdown (i.e., before completion of the shutdown), the event need not be reported.

In addition, if a condition that was prohibited by the Technical Specifications existed (i.e., the plant was in a degraded mode allowed by the Technical Specifications) for a period of time longer than that permitted by the Technical Specifications, it must be reported even if the condition was not discovered until after the allowable time had elapsed and the condition was rectified immediately after discovery.

Paragraph 50.73(a)(2)(ii) requires reporting of:

"Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded, or that resulted in the nuclear power plant being:

(A) In an unanalyzed condition that significantly compromised plant safety;

(B) In a condition that was outside the design basis of the plant; or

(C) In a condition not covered by the plant's operating and emergency procedures."

This paragraph requires events to be reported where the plant, including its principal safety barriers, was seriously degraded or in an unanalyzed condition. For example, small voids in systems designed to remove heat from the reactor core which have been previously shown through analysis not to be safety significant need not be reported. However, the accumulation of voids that could inhibit the ability to adequately remove heat from the reactor core, particularly under natural circulation conditions, would constitute an unanalyzed condition and must be reported. In addition, voiding in instrument lines that results in an erroneous indication causing the operator to significantly misunderstand the true condition of the plant is also an unanalyzed condition and must be reported.

The Commission recognizes that the licensee may use engineering judgment and experience to determine whether an unanalyzed condition existed. It is not intended that this paragraph apply to minor variations in individual parameters, or to problems concerning single pieces of equipment. For example, at any time, one or more safety-related components may be out of service due to testing, maintenance, or a fault that has not yet been repaired. Any trivial single failure or minor error in performing surveillance tests could produce a situation in which two or more often unrelated, safety-related components are out-of-service. Technically, this is an unanalyzed condition. However, these events should be reported only if they involve functionally related components or if they significantly compromise plant safety.

Finally, this paragraph also includes material (e.g., metallurgical, chemical) problems that cause abnormal degradation of the principal safety barriers (i.e., the fuel cladding, reactor coolant system pressure boundary, or the containment).

Additional examples of situations included in this paragraph are:

(a) Fuel cladding failures in the reactor or in the storage pool that exceed expected values, that are unique or widespread, or that resulted from unexpected factors.

(b) Reactor coolant radioactivity levels that exceeded Technical Specification limits for iodine spikes or, radioactivity levels at a BWR air ejector monitor that exceeded the Technical Specification limits.

(c) Cracks and breaks in piping, the reactor vessel, or major components in the primary coolant circuit that have safety relevance (steam generators, reactor coolant pumps, valves, etc.)

(d) Significant welding or material defects in the primary coolant system.

(e) Serious temperature or pressure transients (e.g., transients that violate the plant's Technical Specifications).

(f) Loss of relief and/or safety valve operability during test or operation (such that the number of operable valves is less than required by the Technical Specifications).

(g) Loss of containment function or integrity (e.g., containment leakage rates exceeding the authorized limits).

Paragraph 50.73(a)(2)(iii) requires reporting of:

> "Any natural phenomenon or other external condition that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant."

This paragraph applies only to acts of nature (e.g., tornadoes) and external hazards (e.g., railroad tank car explosion). References to acts of sabotage are covered by §73.71.

Threats to personnel from internal hazards (e.g., radioactivity releases) are covered by a separate paragraph [§50.73(a)(2)(x)].

This paragraph requires those events to be reported where there is an actual threat to the plant from an external condition or natural phenomenon, and where the threat or damage challenges the ability of the plant to continue to operate in a safe manner (including the orderly shutdown and maintenance of shutdown conditions).

The licensee is to decide if a phenomenon or condition actually threatened the plant. For example, a minor brush fire in a remote area of the site that was quickly controlled by fire fighting personnel and, as a result, did not present a threat to the plant need not be reported. However, a major forest fire, large-scale flood, or major earthquake that presents a clear threat to the plant must be reported. Industrial or transportation accidents that occurred near the site and created a plant safety concern must also be reported.

<u>Paragraph 50.73(a)(2)(iv)</u> requires reporting of:

> "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including the RPS, that resulted from and was part of the preplanned sequence during testing or reactor operation need not be reported."

This paragraph requires events to be reported whenever an ESF actuates either manually or automatically, regardless of plant status. It is based on the premise that the ESFs are provided to mitigate the consequences of a significant event and, therefore, (1) they should work properly when called upon and (2) they should not be challenged frequently or unnecessarily. The Commission is interested both in events where an ESF was needed to mitigate the consequences (whether or not the equipment performed properly) and events where an ESF operated unnecessarily.

"Actuation" of multichannel ESF Actuation Systems is defined as actuation of enough channels to complete the minimum actuation logic (i.e., activation of sufficient channels to cause activation of the ESF Actuation System). Therefore, single channel actuations, whether caused by failures or otherwise, are not reportable if they do not complete the minimum actuation logic.

Operation of an ESF as part of a planned operational procedure or test (e.g., startup testing) need not be reported. However, if during the planned operating procedure or test, the ESF actuates in a way that is not part of the planned procedure, that actuation must be reported. For example, if the normal reactor shutdown procedure requires that the control rods be inserted by a manual reactor trip, the reactor trip need not be reported. However, if conditions develop during the shutdown that require an automatic reactor trip, such a reactor trip must be reported.

The fact that the safety analysis assumes that an ESF will actuate automatically during certain plant conditions does not eliminate the need to report that actuation. Actuations that need not be reported are those initiated for reasons other than to mitigate the consequences of an event (e.g., at the discretion of the licensee as part of a planned procedure or evolution).

Paragraphs 50.73(a)(2)(v) and (vi) require reporting of:

"(v) Any event or condition that alone could have prevented the fulfillment of the safety function of structures or systems that are needed to:

(A) Shut down the reactor and maintain it in a safe shutdown condition;

(B) Remove residual heat;

(C) Control the release of radioactive material; or

(D) Mitigate the consequences of an accident.

(vi) Events covered in paragraph (a)(2)(v) of this section may include one or more personnel errors, equipment failures, and/or discovery of design, analysis, fabrication, construction, and/or procedural inadequacies. However, individual component failures need not be reported pursuant to this paragraph if redundant equipment in the same system was operable and available to perform the required safety function."

The intent of thess paragraphs is to capture those events where there would have been a failure of a safety system to properly complete a safety function, regardless of when the failures were discovered or whether the system was needed at the time.

These paragraphs are also based on the assumption that safety-related systems and structures are intended to mitigate the consequences of an accident. While §50.73(a)(2)(iv) of this rule applies to actual actuations of an ESF, §50.73(a)(2)(v) covers an event or condition where redundant structures, components, or trains of a safety system could have failed to perform their intended function because of: one or more personnel errors, including procedure violations; equipment failures; or design, analysis, fabrication, construction, or procedural deficiencies. The event must be reported regardless of the situation or condition that caused the structure or systems to be unavailable, and regardless of whether or not an alternate safety system could have been used to perform the safety function (e.g., High Pressure Core Cooling failed, but feed-and-bleed or Low Pressure Core Cooling were available to provide the safety function of core cooling).

The applicability of these paragraphs includes those safety systems designed to mitigate the consequences of an accident (e.g., containment isolation, emergency filtration). Hence, minor operational events involving a specific component such as valve packing leaks, which could be considered a lack of control of radioactive material, should not be reported under this paragraph. System leaks or other similar events may, however, be reportable under other paragraphs.

It should be noted that there are a limited number of single-train systems that perform safety functions (e.g., the High Pressure Coolant Injection System in BWRs). For such systems, loss of the single train would prevent the fulfillment of the safety function of that system and, therefore, must be reported even though the plant Technical Specifications may allow such a condition to exist for a specified limited length of time.

It should also be noted that, if a potentially serious human error is made that could have prevented fulfillment of a safety function, but recovery factors resulted in the error being corrected, the error is still reportable.

The Commission recognizes that the application of this and other paragraphs of this section involves the use of engineering judgment on the part of licensees. In this case, a technical judgment must be made whether a failure or operator action that did actually disable one train of a safety system, could have, but did not, affect a redundant train within the ESF system. If so, this would constitute an event that "could have prevented" the fulfillment of a safety function, and, accordingly, must be reported.

If a component fails by an apparently random mechanism it may or may not be reportable if the functionally redundant component could fail by the same mechanism. Reporting is required if the failure constitutes a condition where there is reasonable doubt that the functionally redundant train or channel would remain operational until it completed its safety function or is repaired. For example, if a pump in one train of an ESF system fails because of improper lubrication, and engineering judgment indicates that there is a reasonable expectation that the functionally redundant pump in the other train, which was also improperly lubricated, would have also failed before it completed its safety function, then the actual failure is reportable and the potential failure of the functionally redundant pump must be discussed in the LER.

For safety systems that include three or more trains, the failure of two or more trains should be reported if, in the judgment of the licensee, the functional capability of the overall system was jeopardized.

Interaction between systems, particularly a safety system and a non-safety system, is also included in this criterion. For example, the Commission is increasingly concerned about the effect of a loss or degradation of what had been assumed to be non-essential inputs to safety systems. Therefore, this paragraph also includes those cases where a service (e.g., heating, ventilation, and cooling) or input (e.g., compressed air) which is necessary for reliable or long-term operation of a safety system is lost or degraded. Such loss or degradation is reportable if the proper fulfillment of the safety function is not or cannot be assured. Failures that affect inputs or services to systems that have no safety function need not be reported.

Finally the Commission recognizes that the licensee may also use engineering judgment to decide when personnel actions could have prevented fulfillment of a safety function. For example, when an individual improperly operates or maintains a component, he might conceivably have made the same error for all of the functionally redundant components (e.g., if he incorrectly calibrates one bistable amplifier in the Reactor Protection System, he could conceivably incorrectly calibrate all bistable amplifiers). However, for an event to be reportable it is necessary that the actions actually affect or involve components in more than one train or channel of a safety system, and the result of the actions must be undesirable from the perspective of protecting the health and safety of the public. The components can be functionally redundant (e.g., two pumps in different trains) or not functionally redundant (e.g., the operator correctly stops a pump in Train "A" and, instead of shutting the pump discharge valve in Train "A," he mistakenly shuts the pump discharge valve in Train "B").

Paragraph 50.73(a)(2)(vii) requires the reporting of:

> "Any event where a single cause or condition caused at least one independent train or channel to become inoperable in multiple systems or two independent trains or channels to become inoperable in a system designed to:
>
> (A) Shut down the reactor and maintain it in a safe shutdown condition,
>
> (B) Remove residual heat,
>
> (C) Control the release of radioactive material; or
>
> (D) Mitigate the consequences of an accident."

This paragraph requires those events to be reported where a single cause produced a component or group of components to become inoperable in redundant or independent portions (i.e., trains or channels) of one or more systems having a safety function. These events can identify previously unrecognized common cause failures and systems interactions. Such failures can be simultaneous failures which occur because of a single initiating cause (i.e., the single cause or mechanism serves as a common input to the failures); or the failures can be sequential (i.e., cascade failures), such as the case where a single component failure results in the failure of one or more additional components.

To be reportable, however, the event or failure must result in or involve the failure of independent portions of more than one train or channel in the same or different systems. For example, if a cause or condition caused components in Train "A" and "B" of a single system to become inoperable, even if additional trains (e.g., Train "C") were still available, the event must be reported. In addition, if the cause or condition caused components in Train "A" of one system and in Train "B" of another system (i.e., a train that is assumed in the safety analysis to be independent) to become inoperable, the event must be reported. However, if a cause or condition caused components in Train "A" of one system and Train "A" of another system (i.e., trains that are not assumed in the safety analysis to be independent), the event need not be reported unless it meets one or more of the other criteria in this section.

This paragraph does not include those cases where a train of a system or a component was removed from service as part of a planned evolution, in accordance with an approved procedure, and in accordance with the plant's Technical Specifications. For example, if the licensee removes part of a system from service to perform maintenance, and the Technical Specifications permit th resulting configuration, and the system or component is returned to service within the time limit specified in the Technical Specifications, the action need not be reported under this paragraph. However, if, while the train or component is out of service, the licensee identifies a condition that could have prevented the whole system from performing its intended function (e.g., the licensee finds a set of relays that is wired incorrectly), that condition must be reported.

Paragraphs 50.73(a)(2)(viii) and (ix) require reporting of:

"(viii)(A) Any airborne radioactivity release that exceeded 2 times the applicable concentrations of the limits specified in Table II of Appendix B to Part 20 of this chapter in unrestricted areas, when averaged over a time period of one hour.

(B) Any liquid effluent release that exceeded 2 times the limiting combined Maximum Permissible Concentration (MPC) (see Note 1 of Appendix B to Part 20 of this chapter) at the point of entry into the receiving water (i.e., unrestricted area) for all radionuclides except tritium and dissolved noble gases, when averaged over a time period of one hour.

(ix) Reports submitted to the Commission in accordance with paragraphs (a)(2)(viii) of this section also meet the effluent release reporting requirements of paragraph 20.405(a)(5) of Part 20 of this chapter."

Paragraph (viii) is similar to §20.405 but places a lower threshold for reporting events at commercial power reactors. The lower threshold is based on the significance of the breakdown of the licensee's program necessary to have a release of this size, rather than on the significance of the impact of the actual release. Reports of events covered by §50.73(a)(2)(viii) are to be made in lieu of reporting noble gas releases that exceed 10 times the instantaneous release rate, without averaging over a time period, as implied by the requirement of §20.405(a)(5).

Paragraph 50.73(a)(2)(x) requires reporting of:

"Any event that posed an actual threat to the safety of the nuclear power plant or significantly hampered site personnel in the performance of duties necessary for the safe operation of the nuclear power plant including fires, toxic gas releases, or radioactive releases."

This paragraph includes physical hazards (internal to the plant) to personnel (e.g., electrical fires). In addition, the hazard must hamper the ability of site personnel to perform safety-related activities affecting plant safety.

In-plant releases must be reported if they require evacuation of rooms or buildings containing systems important to safety and, as a result, the ability of the operators to perform necessary safety functions is significantly hampered. Precautionary evacuations of rooms and buildings that subsequent evaluation determines were not required need not be reported.

Paragraph 50.73(b) describes the format and content of the LER. It requires that the licensee prepare the LER in sufficient depth so that knowledgeable readers conversant with the design of commerical nuclear power plants, but not familiar with the details of a particular plant, can understand the complete event (i.e., the cause of the event, the plant status before the event, and the sequence of occurrences during the event).

Paragraph 50.73(b)(1) requires that the licensee provide a brief abstract (i.e., no more than 1400 typewritten characters, including spaces) describing the major occurrences during the event. The abstract should include all actual component or system failures that contributed to the event, all relevant operator errors, violations of procedures, and any significant corrective action taken or planned as a result of the event. This paragraph is needed to give LER data base users a brief description of the event in order to identify events of interest.

Paragraph 50.73(b)(2) requires that the licensee include in the LER a clear, specific narrative statement of exactly what happened during the entire event so that readers not familiar with the details of a particular

plant can understand the event. The licensee should emphasize how systems, components, and operating pers[illegible] performed. Specific hardware problems should not be covered in ex[illegible] detail. Characteristics of a plant that are unique and that infl[illegible]d the event (favorably or unfavorably) must be described. Terms, [illegible], or acronyms only in local use should be avoided or, as a minimum [illegible]ly defined.

The narrative must also d[illegible] the event from the perspective of the operator (e.g., what the op[illegible]ator saw, d[illegible], perceived, understood, or misunderstood).

Paragraph 50.73(b)(2)(ii)(F) requires that the Energy Industry Identification System (EIIS) component function identifier and system name for each component or system referred to in the LER be included in the LER. The "system name" may be either the full name (e.g., reactor coolant system) or the two letter system code (e.g., AB). When the name is long (e.g., residual heat removal low pressure coolant injection system (BWR)) the code should be used. The EIIS component function identifier and/or system name (i.e., two letter code) should be included in parenthesis following the first reference to a system or component in the text of the LER. The component function identifier and system name need not be repeated with each subsequence reference to the same component or system. In addition, EIIS component function identifiers and system names should not be included in the abstract section of the LER.

Whenever an uncertainty arises concerning the interpretation of a system boundary, the boundary should be defined consistent with the comparable system descriptions and interpretations contained in the NPRDS Reportable System and Component Manual for those systems included in the NPRDS reportable scope.

Paragraph 50.73(b)(3) requires that the LER include a summary assessment of the actual and potential safety consequences and implications of the event, including the criterion or criteria for reporting or other basis for submitting the report. This assessment may be based on the conditions existing at the time of the event. The evaluation must be carried out to the extent necessary to fully assess the safety consequences and safety margins associated with the event. An assessment of the event under alternative conditions must be included if the incident would have been more severe (e.g., the plant would have been in a condition not analyzed in the Safety Analysis Report) under reasonable and credible alternative conditions, such as power level or operating mode. For example, if an event occurred while the plant was at 15% power and the same event could have occurred while the plant was at 100% power, and, as a result, the consequences would have been considerably more serious, the licensee must assess and report those consequences.

Paragraph 50.73(b)(4) requires that the licensee describe in the LER any corrective actions planned as a result of the event that are known at the time the LER is submitted, including actions to reduce the probability of similar events occurring in the future. In addition, the licensee should describe corrective actions on similar or related components that were done as a direct result of the event (e.g., Pump #1 fails during an event and required corrective maintenance, however, the maintenance was also done on Pump #2).

The licensee should reference any previous similar events or failures, particularly if they were reported as LERs, and discuss why prior corrective action did not prevent recurrence (i.e., any earlier events which in retrospect are significant in relation to the subject event). After the initial LER is submitted, only substantial changes in the corrective action need be reported as a supplemental LER.

Paragraph 50.73(c) authorizes the NRC staff to require the licensee to submit specific supplemental information beyond that required by §50.73(b), if requested. Such information may be required if the staff finds that supplemental material is necessary for complete understanding of an unusually complex or significant event. Such requests for supplemental information must be made in writing, and the licensee must submit the requested information as a supplement to the initial LER within a time period mutually agreed upon by the NRC staff and the licensee.

Paragraph 50.73(f) gives the NRC's Executive Director for Operations the authority to grant case-by-case exemptions to the reporting requirements contained in the LER rule. This exemption could be used to limit the collection of certain data in those cases where full participation would be unduly difficult because of a plant's unique design or circumstances.

Paragraph 50.73(g) states that the reporting requirements contained in §50.73 replace the reporting requirements in all nuclear power plant Technical Specifications that are typically associated with Reportable Occurrences. The reporting requirements superseded by §50.73 are those contained in the Technical Specification sections that are usually titled "Prompt Notification with Written Followup" (Section 6.9.1.8) and "Thirty Day Written Reports" (Section 6.9.1.9). The reporting requirements that have been superseded are also described in Regulatory Guide 1.16, Revision 4, "Reporting of Operating Information-Appendix A Technical Specification," Paragraph 2, "Reportable Occurrences." The special reports typically described in Section 6.9.2, "Special Reports," of the Technical Specifications are still required.

## VI. INSTRUCTIONS FOR COMPLETING THE LER FORMS

General Instructions

1. Entries should be made for all items (facility name, etc.), as noted in the Specific Instructions below.

2. In the textual areas on the form (see Appendix A) (e.g., Items 16 and 17), there are certain limitations on the use of symbols. These limitations are:

   A. "Degree" should be spelled out.

   B. "Less than or equal to" use </=.

   C. "Greater than or equal to" use >/=.

   D. "Plus or minus" use +/-.

   E. Delta and any other Greek letter should be spelled.

   F. Exponents are not acceptable. A number should either be expressed as a decimal, spelled out, or preferably (to save space) designated in E field format. For example, $4.2 \times 10^{-6}$ could be expressed as 4.2 E-6, 0.0000042, or 4.2 x 10(-6).

3. Errors discovered in an LER should be corrected in a revised report. A revision of a prior LER should be identified in the LER report number, as described in Item 6 of the Specific Instructions. The revised report will replace the previous report in the computer file. Therefore, the update should be a complete entity and not contain only supplementary or revised information to the previously submitted report.

   Revisions should only be used to provide additional or corrected information about a previously reported event. A revision should not be used to report subsequent failures of the same or like component.

   Some plants have in the past used revisions to report new events that were discovered months after the original event but were loosely related to the original event (e.g., were discovered in response to the same IE Bulletin). These revisions had different event dates and discussed new, although similar, events. Events of this type should be reported as new LERs and should not be reported as revisions to previous LERs.

   Only substantial information that would significantly change a reader's perception of the course or consequences of an event, or substantial changes in the corrective action planned by the licensee need be reported as a revised LER.

4. A cover letter should be used to forward the LER to the NRC. The cover letter should include, as a minimum; (1) the signature of the company official submitting the report, (2) to whom copies were sent, and (3) the date issued (the data issued and the report date (see item 7 below) should be the same).

## Specific Instructions

### Item 1: Facility Name

Enter the name of the facility (e.g., Indian Point, Unit 1) at which the event occurred. If the event involved more than one unit at a station, enter the name of the nuclear facility with the lowest nuclear unit number (e.g., Three Mile Island, Unit 1).

### Item 2: Docket Number

Enter the docket number assigned to the unit. Examples of the proper manner of entering the docket number are as follows:

| Licensee | Docket Number |
|---|---|
| Yankee-Rowe | 05000029 |
| Sequoyah 2 | 05000328 |

Note: Use zeros as noted in the examples.

### Item 3: Page Number

Enter the total number of pages included (including figures and tables that are attached to the text description) in the LER package. For continuation sheets, number the pages consecutively beginning with "2" (the LER form that includes the abstract and other data is prenumbered on the form as Page 1).

### Item 4: Event Title

Enter a concise description of the event which defines the principal problem or issue associated with the event (e.g., "Inoperable Diesel Generators," "Reactor Trip," "Failure of the Reactor Trip Breakers"). The title, "Licensee Event Report" should <u>not</u> be used.

### Item 5: Event Date

Enter the date on which the event occurred in the six spaces provided (month, day, and year). Examples of the proper manner of entering the event date are as follows:

For "June 1, 1977" Enter "060177"

For " October 2, 1977" Enter "100277"

Note: Use leading zeros in the first and third spaces when appropriate as noted in the first example.

Item 6: Report Number

The Licensee Event Report (LER) number consists of three parts, (1) the last two digits of the event year (based on event date, not report date), the sequential report number, and (3) a revision number. The numbering system is shown in the diagram below:

| Event Year | Sequential Report Number | Revision Number |
|---|---|---|
| _ _ - | _ _ _ - | _ |

A. Event Year: Enter the last two digits of the year in which the event occurred. For example, for events occurring in 1984 enter 84 in the spaces provided.

B. Sequential Report Number: As each reportable event occurs and is reported for an individual unit during the year, it is assigned a sequential number. For example, for the first, fifteenth, and thirty-third events to occur and be reported in a given year at a given unit, enter 001, 015, and 033 in the spaces provided, respectively.

   The following criteria should be followed to insure consistency in the sequential numbering of reports:

   - Each nuclear unit should have its own set of sequential report numbers. Nuclear units at multi-unit sites should not share a set of sequential report numbers.

   - The sequential number should begin with 001 for the first event that occurred in each calendar year.

   - Use leading zeros for sequential numbers less than 100.

   - For an event common to both units of a two unit site, assign the sequential number to the lowest numbered nuclear unit.

   - If a sequential number is assigned to an event, and it is subsequently determined that the event is not reportable, a "hole" in the series of LERs would result. The licensee should write a brief letter to the NRC noting that "LER number xxx for docket 05000YYY will not be used."

C. Revision Number: The revision number of the original LER submitted is 0. The revision number for the first revision submitted should be 1. Subsequent revisions should be numbered sequentially (e.g., 2, 3, 4).

Item 7: Report Date

Enter the date of the report to the NRC in the six spaces provided, as described in Item 5 above.

Item 8: Other Facilities

Enter the facility name and docket number (see items 1 and 2 for format) of any other facilities at that site that were directly affected by the event (e.g., the event included shared components, the LER describes a tornado that threatened both units of a two unit plant).

Item 9: Operating Mode

Enter the operating mode (as defined in the plant's Technical Specifications) of the unit at the time of the event in the single space provided. If a plant's Technical Specifications do not specifically define operating modes, the letter "N" should be entered in this field.

Item 10: Power Level

Enter the percent of licensed thermal power at which the reactor was operating when the event occurred. For events occurring during shutdown conditions, enter 000. For all other operating conditions, enter the correct numerical value (estimate power level if it is not known precisely). Significant deviations in the operating power in the balance of plant should be clarified in the text. Leading zeros should be used (e.g., 009 for 9% power; 072 for 72% power).

Item 11: Reporting Requirements

Check one or more blocks depending on the reporting requirements that were met by the event. A single event can meet more than one reporting criteria. For example: if as a result of sabotage (reportable under §73.71(b)) a safety system failed to function (reportable under §50.73(a)(2)(v)) and the net result was a release of radioactive material in a restricted area that exceeded the applicable license limit (reportable under §20.405(a)(1)(iii)); the licensee should prepare a single LER and check the three boxes for §73.71(b), 50.73(a)(2)(v), and §20.405(a)(1)(iii). In addition, an event can be reportable as an LER even if it does not meet any of the criteria in 50.73 [e.g., a case of attempted sabotage (73.71(b)) that does not result in any consequences that meet the criteria in 50.73].

The "Other" block should be used if a reporting requirement is met that is not specified in Item 11. The reporting requirement should be specifically described in the abstract and text.

Item 12: Licensee Contact

Enter the Name, Position Title, and work Telephone Number of a person within the licensee's organization who is knowledgeable about the specific event described in the LER and who can provide additional information and clarification concerning the event and the plant's characteristics.

Item 13: Component Failures

Enter the appropriate data for each component failure described in the event.

A failure is defined as the termination of the ability of an item to perform its required function. Failures may be unannounced and not detected until the next test (unannounced failure), or they may be announced and detected by any number of methods at the instant of occurrence (announced failure). For vessels, piping, pipe fittings and penetration assemblies, a failure is any condition that permits detectable leakage of the contained fluid through the actual pressure boundary.

If multiple components failed and all of the information in Item 13 (e.g., cause, system, component) is identical for each component, then only a single entry is required in Item 13. The number of components that failed should be clearly defined in the abstract and text.

If more than four failures need to be coded, use one or more Failure Continuation Sheets (NRC-366B).

a. Cause - Enter the cause code from Appendix B. If more than one cause code is applicable, enter the cause code that most closely describes the root cause of the failure.

b. System - Enter the two letter system code from IEEE Standard 805-1983,* "Recommended Practices for System Identification in Nuclear Power Plants and Related Facilities."

c. Component - Enter the applicable component code from IEEE Standard 803A-1983,* "Recommended Practice for Unique Identification in Power Plants and Related Facilities - Component Function Identifiers."

d. Component Manufacturer - Enter the four character alphanumeric reference code for the manufacturer of the component as listed in Table 9 of the NPRDS Reporting Procedures Manual. Manufacturers that are not included in the list should be designated X999.

e. Reportable to NPRDS - Enter a "Y" if the failure is reportable to the NPRD System. Enter an "N" if the failure is not reportable to NPRDS.

Failure Continuation Sheet (NRC-366B)

If necessary, additional component failures may be coded on one or more failure continuation sheets. The entries in Items 1, 2, 3, and 6 of the failure continuation sheet should be coded in the same manner as the entries in Items 1, 2, 3, and 6 of the initial page of the LER. Item 13 should be completed in the same manner as Item 13 on the basic LER form (NRC-366). Failures coded on the LER form (NRC-366) should not be repeated on the failure continuation sheet. Any failure continuation sheets should follow any text continuation sheets.

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* Copies may be obtained form the Institute of Electrical and Electronics Engineers, 345 East 47th Street, New York, NY 10017. A copy is available for inspection and copying for a fee at the Commission's Public Document Room, 1717 H Street, NW, Washington, D.C. and at the Office of the Federal Register, 1100 L St. NW, Washington, D.C.

Item 14: Supplemental Report

Check "Yes" if the licensee plans to submit a follow-up report (e.g., a failed component had been returned to the manufacturer for additional testing, and the results of the test were not yet available when the LER was submitted).

Item 15: Expected Submission Date of Supplemental Report

Enter the expected date of submission of the supplemental LER, if applicable. See Item 5 for the proper date format.

Item 16: Abstract

Provide a brief abstract describing the major occurrences during the event, including all actual component or system failures that contributed to the event, all relevant operator errors or violations of procedures, and any significant corrective action taken or planned as a result of the event.

The abstract is needed to give LER data base users a brief description of the event in order to identify events of interest. The abstract should be limited to 1400 characters (including spaces) which is approximately fifteen lines of single-spaced typewritten text.

Item 17: Text

Enter the text of the LER. The LER should be written in sufficient depth so that knowledgeable readers conversant with the design of commercial nuclear power plants, but not familiar with the details of a particular plant, can understand the complete event (i.e., the cause of the event, the plant status before the event, and the sequence of occurrences during the event). The licensee should emphasize how systems, components, and operating personnel performed. Specific hardware problems should not be covered in excessive detail. Characteristics of a plant that are unique and that influenced the event (favorably or unfavorably) should be described. The text should also describe the event from the perspective of the operator (e.g., what the operator saw, did, perceived, understood, or misunderstood). Specific information that should be included, as appropriate for the particular event is described in Paragraphs 50.73(b)(2)(ii) 50.73(b)(3), 50.73(b)(4), and 50.73(b)(5) of the rule (see Section II).

There is no prescribed format for the LER text. The text should be written in the format that most clearly describes the event. Although 50.73(b) defines the information that should be included, as appropriate, for a particular event, it is not intended as an outline of the text format. After the text is written, however, the appropriate sections of 50.73(b) should be reviewed to insure that applicable subject have been adequately addressed in the text.

Text Contination Sheet (NRC-366A)

If necessary, the text may be continued on one or more additional text continuation sheets. There is no limit on the number of continuation sheets that may be included.

If one or more contination sheets are used, the entries in Items 1, 2, 3, and 6 should be coded in the same manner as the entries in Items 1, 2, 3, and 6 of the initial page of the LER.

APPENDIX A
LICENSEE EVENT REPORT FORMS

NRC Form 366 (9-83)

U.S. NUCLEAR REGULATORY COMMISSION

APPROVED OMB NO. 3150-0104
EXPIRES: 8/31/85

# LICENSEE EVENT REPORT (LER)

| FACILITY NAME (1) | DOCKET NUMBER (2) | PAGE (3) |
|---|---|---|
| | 0 5 0 0 0 | 1 OF |

TITLE (4)

| EVENT DATE (5) | | | LER NUMBER (6) | | | REPORT DATE (7) | | | OTHER FACILITIES INVOLVED (8) | |
|---|---|---|---|---|---|---|---|---|---|---|
| MONTH | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | MONTH | DAY | YEAR | FACILITY NAMES | DOCKET NUMBER(S) |
| | | | | — | — | | | | | 0 5 0 0 0 |
| | | | | | | | | | | 0 5 0 0 0 |

OPERATING MODE (9)

POWER LEVEL (10)

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check one or more of the following) (11)

| | | | |
|---|---|---|---|
| ☐ 20.402(b) | ☐ 20.405(c) | ☐ 50.73(a)(2)(iv) | ☐ 73.71(b) |
| ☐ 20.405(a)(1)(i) | ☐ 50.36(c)(1) | ☐ 50.73(a)(2)(v) | ☐ 73.71(c) |
| ☐ 20.405(a)(1)(ii) | ☐ 50.36(c)(2) | ☐ 50.73(a)(2)(vii) | ☐ OTHER (Specify in Abstract below and in Text, NRC Form 366A) |
| ☐ 20.405(a)(1)(iii) | ☐ 50.73(a)(2)(i) | ☐ 50.73(a)(2)(viii)(A) | |
| ☐ 20.405(a)(1)(iv) | ☐ 50.73(a)(2)(ii) | ☐ 50.73(a)(2)(viii)(B) | |
| ☐ 20.405(a)(1)(v) | ☐ 50.73(a)(2)(iii) | ☐ 50.73(a)(2)(x) | |

LICENSEE CONTACT FOR THIS LER (12)

| NAME | TELEPHONE NUMBER | |
|---|---|---|
| | AREA CODE | |
| | | |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO NPRDS | | CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO NPRDS | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | | | | | |
| | | | | | | | | | | | |

SUPPLEMENTAL REPORT EXPECTED (14)

☐ YES (If yes, complete EXPECTED SUBMISSION DATE) ☐ NO

| EXPECTED SUBMISSION DATE (15) | MONTH | DAY | YEAR |
|---|---|---|---|
| | | | |

ABSTRACT (Limit to 1400 spaces, i.e., approximately fifteen single-space typewritten lines) (16)



NRC Form 366 (9-83)

NRC Form 366A
(9-83)

U.S. NUCLEAR REGULATORY COMMISSION

# LICENSEE EVENT REPORT (LER) TEXT CONTINUATION

APPROVED OMB NO. 3150-0104
EXPIRES 8/31/85

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) | | | | | PAGE (3) | | |
|---|---|---|---|---|---|---|---|---|---|
| | | YEAR | | SEQUENTIAL NUMBER | | REVISION NUMBER | | | |
| | 0 5 0 0 0 | | — | | — | | | OF | |

TEXT (If more space is required, use additional NRC Form 366A's) (17)

NRC FORM 366A

NRC Form 366A
(9-83)

# LICENSEE EVENT REPORT (LER) FAILURE CONTINUATION

U.S. NUCLEAR REGULATORY COMMISSION
APPROVED OMB NO. 3150–0104
EXPIRES 8/31/85

| FACILITY NAME (1) | DOCKET NUMBER (2) | LER NUMBER (6) | | | PAGE (3) |
|---|---|---|---|---|---|
| | | YEAR | SEQUENTIAL NUMBER | REVISION NUMBER | |
| | 0 5 0 0 0 | | — | — | OF |

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

| CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO NPRDS | | CAUSE | SYSTEM | COMPONENT | MANUFAC-TURER | REPORTABLE TO NPRDS | |
|---|---|---|---|---|---|---|---|---|---|---|---|
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NRC FORM 366A

## APPENDIX B
## CAUSE CODES

| Cause Code | Meaning |
| --- | --- |
| A | Personnel Error |
| B | Design, Manufacturing, Construction/Installation |
| C | External Cause |
| D | Defective Procedure |
| E | Management/Quality Assurance Deficiency |
| X | Other |

The general definitions of these cause classifications are:

A. Personnel Error - This classification is assigned to failures attributed to human errors. When errors were made as a result of following incorrect written procedures, the occurrence should be entered under defective procedures (see Paragraph D below). When errors were made because written procedures were not followed or because personnel did not perform in accordance with accepted or approved practice, the occurrence should be classified under personnel error.

B. Design, Manufacturing, Construction/Installation - This classification is assigned to failures reasonably attributed to design, manufacture, construction or installation of a system, component or structure. For example, failures that were traced to such things as defective materials, significant breakdown in the quality assurance program or components otherwise unable to meet the specified functional requirements or performance specifications should be included in this classification.

C. External Cause - This classification is assigned to failures attributed to natural phenomena. A typical example includes failure resulting from a lightning strike, tornado, or flood. This classification is also assigned to man-made external causes that originate off-site (e.g., a industrial accident at a near-by industrial facility).

D. Defective Procedure - This classification is assigned to failures caused by inadequate or incomplete written procedures (see Paragraph A above) or instructions.

E. Management/Quality Assurance Deficiency - This classification is assigned to failures caused by failure of management or management systems (e.g., major breakdowns in the licensees administrative controls, preventive maintenance program, surveillance program, or quality assurance controls).

X. Other - This classification shall be assigned to failures for which the proximate cause cannot be identified or which cannot be assigned to one of the classifications noted above.

APPENDIX C

SAMPLE POTENTIALLY REPORTABLE EVENTS

TITLE: REACTOR TRIP WITH SAFETY INJECTION SYSTEM ACTUATION

As a result of a high stator temperature alarm for the main generator, an automatic turbine runback from 100% power to 4% power was initiated. The reactor did not trip since steam dump and steam generator relief capacity was sufficient. Reactor Coolant System (RCS) pressure increased to about 2340 psig and low steam generator level signals alarmed. After verification of clearing of the stator temperature alarm, an attempt was made to reload the main turbine generator; but, the reactor tripped on high steam generator level resulting from the reducing steam pressure.

Following the reactor trip, the cooldown effect from the open steam dump and steam generator relief valves caused the RCS pressure to momentarily drop to the Safety Injection Systems (SIS) trip point, but, pressure remained above the SIS pump shutoff head preventing injection of borated SIS water. The operators secured the SIS pumps after verification of recovery of RCS pressure. All safety systems performed normally.

Comments:

1. The event is reportable because the reactor tripped [50.73(a)(2)(iv)].

2. The event is reportable because an Engineered Safety Feature (the Safety Injection System) actuated. The event is reportable even though the SIS did not actually inject into the Reactor Coolant System [50.73(a)(2)(iv)].

TITLE: REACTOR SCRAM WITH SAFETY RELIEF VALVE FAILURE

As a result of a personnel valving error in the condenser circulating water system, a turbine trip, reactor scram, and a Group 1 isolation (closing the Main Steam Isolation Valves) occurred. About 20 minutes into the event, reactor pressure was increasing and a safety relief valve was manually opened to reduce pressure. The valve stuck open for 3 to 5 minutes and reseated at a reactor pressure of approximately 320 psig. The High Pressure Coolant Injectio (HPCI) System and Reactor Core Isolation Cooling (RCIC) System initiated as designed to increase water level. The HPCI discharge injection valve failed to open automatically and was manually opened. RCIC performed as designed. Water level remained well above the top of the fuel throughout the event.

Comments:

1. The event is reportable because the reactor scrammed [50.73(a)(2)(iv)].

2 The event is reportable because several Engineered Safety Features (Group 1 isolation, HPCI, and, in some cases, RCIC) actuated [50.73(a)(2)(iv)].

3. The event is reportable because the HPCI failed to fulfill its safety function [50.73(a)(2)(v)].

4. The event is reportable if the failure of the safety relief valve or the HPCI discharge injection valve had common mode or generic implications [50.73(a)(2)(ii)(A)].

TITLE: ELECTRICAL FAULT IN MAIN GENERATOR EXCITATION CIRCUIT AND IDENTIFIED PROBLEM WITH UPPER STAGE OF A REACTOR COOLANT PUMP SEAL

Unit 1 tripped from 96% power caused by a turbine trip. The main generator output breaker opened due to an electrical fault in the generator excitation circuit that led to a loss of main generator output. The resulting voltage transient on the on-site electrical distribution system caused a 4160 volt ESF bus to shed its loads. An emergency diesel engine automatically started. However, the diesel generator was found to be inoperable due to an apparent electrical problem in the voltage control system. In addition, a Reactor Coolant System (RCS) pressure transient occurred as a result of the reactor trip. This pressure transient appears to have caused one of the Reactor Coolant Pumps upper stage seals to fail.

The three Reactor Coolant Pump seals were subsequently replaced during the outage. One seal was replaced because the upper stage seal failed. The other two seals had indicated some erratic behavior and were changed as a preventative measure.

Comments:

1. The event is reportable because the reactor tripped [50.73(a)(2)(iv)].

2. The event is reportable because an Engineered Safety Feature (i.e., a diesel) automatically actuated [50.73(a)(2)(iv)].

3. The failure of the diesel generator and the failure of the reactor coolant pump seal do not make the event reportable as LERs unless the failures had common mode or generic implir tions (e.g., potential for failure of other RCP seals due to the pressure transient) [50.73(a)(2)(v)].

TITLE: DROPPED CONTROL ROD WITH SUBSEQUENT REACTOR TRIP RESULTING IN AN UNSCHEDULED SHUTDOWN

While at 100% power, the unit experienced a dropped control rod with a subsequent automatic load reduction to 70%. Rod recovery procedures were unsuccessful and flux tilt parameters required operator action to further reduce unit load. At about 400 MWe, a full runback occurred resulting in an increase in system pressure and a high pressure reactor trip. The No. 23 Reactor Coolant Pump failed to transfer from the auxiliary transformer to the station transformer and tripped. Upon attempting restart of the RCP, the licensee detected high vibration on the motor's lower radial bearing. Additionally, No. 22 Control Rod Drive MG Set wiped its inboard generator bearing and Steam Generator (SG) water chemistry samples indicated a primary to secondary leak in the No. 23 SG. The leak rate and activity measured in the SG did not exceed the Technical Specification limits.

Comments:

1. The event is reportable because the reactor tripped [50.73(a)(2)(iv)].

2. The event is not reportable because of the high activity in the Steam Generator. The activity level did not exeed the Technical Specification limit and a single steam generator tube failure is an analyzed situation that is within the design basis of the plant [50.73(a)(2)(ii)].

3. The fact that a control rod was dropped and could not be recovered does not make the event reportable unless the drop resulted from serious or generic material problems with common mode or generic implications [50.73(a)(2)(v)].

4. The problems with the No. 23 Reactor Coolant Pumps and the No. 22 Control Rod Drive MG Set do not make the event reportable unless they had common mode or generic implications [50.73(a)(2)(v)].

TITLE: RHR INOPERABLE

When the circuit breaker for the motor operated inlet isolation valve was closed the valve immediately shut. A "shut" control signal was being transmitted to the valve operator controller as a result of Channel B wide range pressure instrumentation maintenance action. When motive power was provided to the motor by closing its power supply breaker it functioned to shut the valve. System low flow alarms occurred in the Control Room and an operator was dispatched to open the valve by hand. Flow was subsequently restored and the system was declared operable.

Redundant trains of the Residual Heat Removal (RHR) System are supplied through a common inlet line from loop 3. The inlet line contains two essential motor-operated isolation valves in series. Shutting either valve renders the RHR trains inoperable. Therefore, both trains of the RHR were declared inoperable when the inlet isolation valve was inadvertently closed.

Comment:

The event is reportable because failure of a single valve caused the RHR system to be inoperable [50.73(a)(2)(v)].

TITLE: POTENTIAL LOSS OF HPCI

During normal refueling leak testing of the upstream containment isolation check valve on the High Pressure Coolant Injection (HPCI) steam exhaust, the disc of the non-containment isolation check valve was found lodged in downstream piping. This might have prevented HPCI from functioning if the disc had blocked the line. HPCI system was operable with the disc lodged in non-blocking position. The event was caused by fatigue failure of a disc pin.

Comments:

1. The event is reportable because of the potential failure of the HPCI to perform its safety function [50.73(a)(2)(v)].

2. The event would be reportable if the fatigue failure of the disc pin is indicative of a common mode failure [50.73(a)(2)(v)].

TITLE: RCIC AUTOMATIC ISOLATION

Reactor Core Isolation Cooling (RCIC) System Automatic Isolation and RCIC High Steam Flow signals were received by the unit operator. The spurious High Steam Flow signal was due to a micro switch in one of two differential pressure cells which failed in the closed position. The RCIC Automatic Isolation closed valves that isolated the steam supply to the RCIC turbine. (The RCIC High Steam Flow Isolation is designed to isolate the RCIC steam line in case of a RCIC pipe break outside of the Primary Containment. This feature functioned as designed.) The RCIC system was declared inoperable. Technical Specifications require the RCIC system to be operable whenever reactor pressure is greater than 150 psi. A similar event occurred in 1979 when RCIC Automatic Isolation tripped on a high steam flow signal. In accordance with Technical Specifications preparations were made to demonstrate the operability of the High Pressure Coolant Injection (HPCI) System. The consequences of this occurrence were minimized due to the short outage of the RCIC System (2 hours 10 minutes), and because safe plant operation was maintained with the HPCI System being available.

Comments:

1. The event would be reportable if the plant safety analysis took credit for operation of RCIC, and the RCIC was unable to perform its intended function [50.73(a)(2)(v)].

2. The event is reportable if RCIC, which actuated, was considered to be an Engineered Safety Feature [50.73(a)(2)(iv)].

TITLE: GENERIC SETPOINT DRIFT

With the plant in steady state operation at 2170 MWt and while performing a Main Steam Line Pressure Instrument Functional Test and Calibration, a switch was found to actuate at 853 psig. The Tech Specs limit is 825 +15 psig head correction. The redundant switches were operable. The cause of the occurrence was setpoint drift. The switch was recalibrated and tested successfully per HNP-2-5279, Barksdale Pressure Switch Calibration, and returned to service.

This is a repetitive event as reported in one previous LER. A generic review revealed that these type switches are used on other safety systems and that this type switch is subject to drift. An investigation will continue as to why these switches drift, and if necessary, they will be replaced.

<u>Comments</u>:

1. The event is not reportable due to the drift of a single pressure switch.

2. The event is reportable if it is indicative of a generic and/or repetitive problem with this type of switch which is used in several safety systems [50.73(a)(2)(v) or (vii)].

TITLE: EMERGENCY FEEDWATER PUMP OVERSPEED

During a period of 25 months, the turbine driven emergency feed pump failed on overspeed a total of fourteen (14) times. These failures occurred during startup and were more likely to occur the longer the interval (idle time) between starts. After the trips, restarts were generally successful.

Comment:

The event is reportable if it were determined that the failure on overspeed was a potential generic or common cause problem that could affect other safety related equipment in the plant. If the failures do not have generic implications, particularly if there are diverse (e.g., motor driven) pumps in the system, the event would not be reportable [50.73(a)(2)(v) or (vii)].

TITLE: INOPERABLE SNUBBERS

Eleven inoperable snubbers were found during periodic testing. All snubbers failed to lock-up in tension and/or compression. These failures did not render their respective systems inoperable. Failure of these snubbers to lock-up within accepted velocity limits was due to improper lockup settings and/or excessive seal bypass. All affected snubbers were overhauled, tested and reinstalled. Additional snubbers of similar type and service were tested satisfactorily per Technical Specifications. These snubbers are designed for low probability seismic events which did not occur. Numerous previous similar events have been reported by this licensee.

Comments:

The event is reportable if it were indicative of a generic problem that could cause numerous multiple independent trains in one or more safety systems to fail to fulfill their safety function following a seismic event [50.73(a)(2)(vii)].

TITLE: UNSEALED HALF-INCH HOLES THROUGH FIRE WALLS

The NRC Site Inspector reported finding four ½ inch unsealed holes through the Control Room Equipment Room east fire wall. The holes open into the stairway east of the Control Room and appear to have been left by the removal of a plate from the wall. The holes were immediately filled with Kaowool.

Subsequently, the NRC Site Inspector found two ½ inch unsealed holes in the north wall of room 427B, Low Voltage Switchgear Room. A firewatch was established until the holes were packed with Kaowool.

In each case, the plant had entered the Action Statement of the Technical Specifications. The Technical Specification requires all penetration fire barriers protecting safety-related areas to be functional at all times. The Action Statement requires that with a penetration fire barrier inoperable, the licensee must establish a continuous fire watch on at least one side of the affected penetration within 1 hour.

The cause of this situation was personnel error as the holes should have been filled with Kaowool when they were first opened, logged in the penetration log, and permanently filled with grout.

Comment:

The event is reportable because the licensee did not meet an Action Statement of the Technical Specification. The event is reportable even though the condition was not discovered until after the time allowed in the Action Statement had elapsed and the condition was rectified immediately after discovery [50.73(a)(2)(i)(B)].

TITLE: HIGH FAILURE RECURRENCE RATE - PROCESS RADIATION MONITORING

With the reactor operating at full power the operator noticed a hi-low flow alarm for the reactor building vent sample system. Laboratory personnel were sent to investigate. The sample pump had failed, and the lab personnel noted that there was a blown fuse for the pump. The fuse was replaced and the pump failed to start. Another fuse was inserted and the back-up pump was switched on but failed to start.

The cause of the pumps not operating was swelled carbon vanes. This was probably caused by moisture. The same pumps failed previously for identical causes. The same type of pumps, utilized for monitoring another radiological effluent pathway failed for the same cause on another occasion.

<u>Comment</u>:

1 The event is not reportable if the Process Radiation Monitoring system is used only to warn (i.e., alarm) the operator that high gaseous radioactivity levels exist in vent gases and no credit is taken for it in any safety analysis and it does not directly control the release of radioactive material [50.73(a)(2)(v)].

2 The event is reportable if similar pumps were used in safety-related systems because a single condition could have caused failures in multiple independent trains of a system that is required to control the release of radioactive material [50.73(a)(2)(vii)].

TITLE: FAILURE OF RADIATION MONITOR PUMPS

During the implementation of a design change, it was found that sediment had accumulated in the service water radiation monitor pumps for the four recirculation spray heat exchangers, causing the pumps to become inoperable. The four pump failures occurred during a period of seven days. These events were caused by sediment in the service water system settling in the pump internals. This sedimentation results in the pump becoming bound and no flow is then available to the applicable service water radiation monitor. The source and flowpath of the sediment is under investigation

Following an accident, each service water radiation monitoring pump would take suction from the service water discharge of one of the four recirculation spray heat exchangers, directing the flow through a radiation monitor in order to detect and identify a leaking heat exchanger. Failure of a radiation monitoring pump would not affect the performance of the associated heat exchanger.

The immediate corrective actions have been to declare the affected pump inoperable and verify the operability of the pump serving the parallel heat exchanger. In the instance when this second pump was also found to be inoperable, both pumps were repaired and verified operable within the 6 hour limitation as stated in the Technical Specifications.

Comment:

1. The event is reportable if the licensee did not meet an Action Statement of the Technical Specification. [50.73(a)(2)(i)(B)].

2. The event is reportable because a single failure caused independent trains in a system to become inoperable [50.73(a)(2)(vii)].

TITLE: OVERSIZED BREAKER WIRING LUGS

During testing of 480 volt safety-related breakers, one breaker would not trip electrically. Investigation revealed that one wire of the pigtail on the trip coil, although still in its lug, was so loose that there was no electrical connection. The loose connection was due to the fact that the pigtail lug was too large (No. 14-16 AWG), whereas the pigtail wire was No. 20 AWG. A No. 18-22 lug is the acceptable industry standard for a No. 20 AWG wire.

Since the trip coils were supplied pre-wired, all safety-related breakers utilizing the trip coil were inspected. All other breakers inspected had 14-16 AWG lugs. No lugs were found with loose electrical connections. Nevertheless, all No. 14-16 AWG lugs were replaced with acceptable industry Standard No. 18-22 AWG lugs.

<u>Comment</u>:

The event is reportable because the incompatible pigtails and lugs could have caused one or more safety systems to fail to perform their intended function [50.73(a)(2)(v)].

TITLE: DIESEL GENERATOR FAILURES

During the annual inspection of standby diesel generator 1G-31, the lower crankshaft thrust bearing (#13) and adjacent main bearing (#12) were found wiped on the journal surface. The #13 bearing was also found to have a small crack from the main oil supply line (located in the center of the journal surface) across the journal surface (approximately 2 inches) to the thrust surface. The depth of the crack in the #13 bearing extended from the journal surface down the oil supply port to the thrust surface (approximately 3/8 inches). The redundant standby diesel generator 1G-21 annual inspection revealed similar problems. Although both diesel generators were operable at the time of the surveillance testing, extended operation without corrective action could have resulted in bearing failure.

<u>Comment</u>:

The event is reportable because, although both diesel generators were operable, there is reasonable doubt that either diesel would have remained operable until it had completed its safety functions [50.73(a)(2)(v)].

TITLE: CONTAINMENT INSTRUMENT ISOLATION VALVES LEFT SHUT

While operating at 15% power, it was found that two containment instrument isolation valves required to be in an open position were in a closed position. Closure of these valves isolated a drywell high pressure switch associated with Emergency Core Cooling System (ECCS) and Reactor Protection System (RPS) initiation. Upon further investigation, it was also identified that the isolation valve to pressure instrumentation that bypasses certain RPS scrams at low pressure was also closed.

The instrument penetrations at which the valves in question are located were used for drywell pressure monitoring during the Integrated Leak Rate Test (ILRT). When the temporary ILRT instrumentation was removed, the penetration instrument isolation valves were inadvertently left in the closed position.

Instruments that were isolated as a result of closure of the valves in question are used as inputs to the following ECCS and RPS circuits:

High Drywell Pressure Scram
Containment Isolation
ECCS Initiation
Containment Spray Interlock
Automatic Pressure Relief
Condenser Low Vacuum (600 psig Interlock)

Comment:

The event would be reportable if the operator actions caused independent trains in more than one system to be inoperable (i.e., the operator erroneously operated two components in more than one train of more than one safety system) [50.73(a)(2)(vii)].

TITLE: CORROSION OF REACTOR COOLANT SYSTEM PIPING

During routine in-service inspection, wastage of the Reactor Coolant System (RCS) piping at a carbon steel to stainless steel (inconel filler) weld (directly below the Reactor Coolant Pump (RCP) casing on the cold leg suction) was noted. One eighth to one quarter inch wastage was observed extending approximately 20% of the 113 inch circumference of the pipe. The wastage is suspected to have been the result of boric acid and galvanic corrosion. Leakage in the area of the RCP appears to be the initiating cause of the corrosion.

Corrosive attack of the RCS pipe, if allowed to proceed undetected, could result in reduction in the primary system boundary integrity. The RCP bi-metallic suction pipe weld, because of its location below the pump, is susceptible to corrosion as a result of leakage from various sources around the pump.

Comment:

The event is reportable because it is indicative of a material (e.g., metallurgical, chemical) problem that has caused abnormal degradation of the Reactor Coolant System pressure boundary [50.73(a)(2)(ii)].

TITLE: REACTOR SHUTDOWN DUE TO HIGH STEAM GENERATOR CONDUCTIVITY

The licensee began a precautionary shutdown from full power because of high conductivity in the steam generators. Leakage of water from service water supply valves to the Auxiliary Feedwater Pumps (AFP) suction during routine AFP testing resulted in contamination of steam generators with lake water.

The in-test conductivity sample in the steam generators indicates 68 micro-mho/$cm^2$. (Licensee's administrative limit is 10 micro-mho/$cm^2$.) After reaching cold shutdown, several days of flushing and draining were required to clean out the secondary systems.

Comment:

1. The event is not reportable unless the shutdown was required by the plant's Technical Specifications [50.73(a)(2)(i)(A)].

2. The event would be reportable if the Technical Specification conductivity limit was exceeded and the Technical Specification did not permit continued operation with the existing conductivity level [50.73(a)(2)(i)(B)].

TITLE: STUCK CONTROL ROD

The plant operator initiated a manual trip from 100% power due to low Steam Generator level after loss of Main Feed Pump 22. This was a repeat of an earlier manual trip except that one control rod (CEA-19) stuck about 8 inches above the core bottom. Shutdown margin met requirements. With NSSS vendor assistance, CEA-19 was freed by varying mag-jack sequence and voltage. Cold checks showed operability.

Comments:

1. The event is not reportable because the system is typically designed to function with the rod with the most worth stuck in the fully withdrawn position. Therefore, the stuck rod it involves an equipment failure (i.e., a stuck control rod) alone would not have prevented the fulfillment of a safety function (e.g., shutdown of the reactor) [50.73(a)(2)(v)].

2. The event is reportable because of the manual actuation of the Reactor Protection System [50.73(a)(2)(iv)].

3. The event is not reportable due to the failure of the Main Feed Pump or the stuck rod if only a random failure of a single component was involved.

TITLE: COMPONENT COOLING WATER HEAT EXCHANGER INOPERABLE

During a routine inspection, an operator noticed a service water leak emanating from a crack in a dissimilar metal weld on the drain valve between the valve and the No. 12 Component Cooling Water Heat Exchanger. Service water was isolated and the No. 12 Component Cooling Water Heat Exchanger was declared inoperable. The dissimilar metal weld between the valve body and carbon steel pipe corroded and cracked.

The Technical Specifications requires:

> "With only one Component Cooling Water Loop operable, restore at least two loops to operable status within 72 hours or be in at least hot standby within the next 6 hours and in cold shutdown within the following 30 hours."

The valve was removed by cutting the pipe above the cracked weld. The pipe was plugged and welded. Service water was restored and No. 12 Component Cooling Water Heat Exchanger was declared operable. The Component Cooling Water Loop was inoperable for 46 hours.

Comment:

The event is not reportable because the condition was corrected before the time limit for achieving hot standby was reached [50.73(a)(2)(i)(A)].

TITLE: INOPERABLE CHECK VALVE

With the unit operating at steady state power level of 100%, a reduction in service water pressure to the charging pumps was experienced. This condition existed only when Pump B was operating. Subsequent troubleshooting indicated that the discharge check valve on the non-operating redundant pump was open. This is contrary to Technical Specifications. While investigating the event, it became apparent that Pump B may have been operated in degraded mode without proper administrative measures in place. Specifically, the internals for the check valve were removed and documentation of this temporary modification was not performed. This is contrary to Technical Specifications. When this modification was performed, cannot be determined. However, both pumps were proven operable twenty days before this event.

The charging pump service water pumps supply cooling water to the charging pump intermediate seal and lube oil coolers. With the "A" pump's discharge isolated, the "B" pump was operable and was performing its intended function.

<u>Comment</u>:

1. The event is reportable because the plant operated with a condition (i.e., the open check valve) prohibited by the Technical Specifications [50.73(a)(2)(i)(B)].

2. The event is reportable because administrative controls required by the Technical Specifications were violated [50.73(a)(2)(i)(B)].

TITLE: DECREASED RCS WATER LEVEL

Decay Heat (DH) Pump 1-2 was stopped when it was discovered that Reactor Coolant System (RCS) water level had decreased from 70" to 37" above the hot leg piping centerline. This placed the unit in violation of an Action Statement of the Technical Specifications which requires that while in Mode 5, at least one reactor coolant loop must be in operation with an associated reactor coolant pump or DH pump operating. The pump was off for 29 minutes, and the RCS temperature remained significantly below the saturation temperature.

Comment:

The event is reportable because the unit violated an Action Statement of the Technical Specifications [50.73(a)(2)(i)(B)].

TITLE: DROPPED FUEL ASSEMBLY

During refueling, a new fuel assembly had just been placed in the core. In the process of removing the fuel grapple and telescoping mast from this fuel assembly the latching mechanism did not retract fully from the fuel upper end fitting. As a result upon retraction of the mast, the fuel assembly was removed from the core in an "unsecured" condition to a point just above the upper core grid. The assembly then fell onto the upper core grid and came to rest in a diagonal position about 35 degrees above the horizontal. The handle end of the assembly was resting on the reactor vessel wall at the height of the feedwater spargers while the lower end was in contact with the core grid.

The apparent cause of this event was the failure of the operator to detect that he was raising the fuel assembly after the fuel grapple failed to release when the operating switch was placed in the "grapple open" position. The failure of the grapple to open was probably a result of the refueling mast not being in the fully lowered position causing the tip of the hook to hang up on the fuel assembly handle. When the mast was raised, the fuel assembly came up while grapple opening air pressure continued to force the grapple hook to the unlatch position. At a point shortly after the fuel assembly cleared the upper grid, the grapple hook opened, releasing the fuel bundle.

The operator failed to detect that he had a fuel assembly on the grapple because he did not perform an adequate rotational check of the telescoping mast before raising the mast. Operators are trained to attempt to rotate the mast to verify that a bundle is or is not fully engaged, however, this requirement was not included in the fuel handling procedures.

Comments:

This event is reportable because the requirement to rotate the mast to verify that a bundle is not fully engaged was not included in the fuel handling procedure and, as a result, the safety function of the fuel handling equipment was lost [50.73(a)(2)(ii)(C)].

TITLE: LOW CONTAINMENT PRESSURE

During a plant cooldown, containment pressure decreased below negative 12 inches of water. Containment cooling was being supplied by three CAR fans in fast with normal and emergency RBCCW cooling. Containment cooling was not changed to match decreasing heat loads during the cooldown. Containment temperature dropped from 103 to 90 degrees F.

CAR fans were shifted to slow with normal RBCCW flow only. Containment pressure increased above negative 12 inches of water within 10 minutes. Operators have been cautioned to balance containment cooling with heat loads during heatups and cooldowns.

Comment:

This event is not reportable if no Technical Specification limits were violated [50.73(a)(2)(i)(B)] and the condition was not outside the design basis of the plant [50.73(a)(2)(ii)(B)].

TITLE: CONTROL ROD FAILURE

While at 97 percent power, turbine load reduction was initiated in response to a steam generator feed pump low suction pressure alarm. After initial rod motion, "control rod urgent failure" was annunciated and the rods could not be moved. Boration was initiated to reduce $T_{ave}$. High pressure in the steam generator caused a safety valve to open. It failed to reseat due to fouling of the manual operating arm. Primary pressure reached 2320 psig and was reduced by spray. A spray valve failed to reseat, reducing pressure to 2140 psig before manual pressure control was effected. At 2:10 a.m., the plant was stable at 46 percent power. The safety valve was reseated. Rod control was restored by replacing the firing circuit and a failed fuse. During the transient, $T_{ave}$ exceeded the LCO limit of 582 degrees F for five minutes, peaking at 592 degrees F. The plant is limited to less than 50 percent power for 24 hours due to accumulated axial flux difference (AFD) penalty minutes. Secondary parameters were recorded and will be evaluated to determine the cause of the initiating loss of feedwater suction pressure.

Comments:

1. The event is reportable because the combination of active failures during the event resulted in the plant being in an unanalyzed condition that significantly compromised plant safety [50.73(a)(2)(ii)A)].

2. The event would not be reportable if the failure of the rods to move would not have prevented the fulfillment of a safety function [50.73(a)(2)(v)].

3. The event is reportable because the operation was prohibited by the Technical Specifications (i.e., an LCO limit ($T_{ave}$ greater than 582°F) was exceeded) [50.73(a)(2)(i)(B)].

TITLE: SPENT FUEL POOL VENTILATION SYSTEM

While conducting a surveillance test procedure, an out-of-specification negative pressure condition was found to exist for the spent fuel pool (SFP) area. The cause of insufficient negative pressure was determined to be operation of one of two area supply fans. With either of the two exhaust fans in operation and the supply fans secured, acceptable negative pressure could not be attained (greater than .125 inches of water). With one supply fan in operation, only .06 inches of water was attained and with both supply fans in operation, essentially no negative pressure existed. Insufficient negative pressure had not been noted during earlier surveillance testing since supply fans were not normally running during the testing period. Supply fan operation was not addressed as a Limiting Condition for Operation in the Technical Specification and was, therefore, not included in the surveillance test procedure.

Maintaining a negative pressure is a typical design condition for certain plant areas to ensure a controlled discharge path to the environment. In this instance, there is no assurance that the controlled release path can be maintained with supply fan(s) in operation.

Comment:

The event is reportable because it demonstrates a condition that was not covered by the plant's operating procedures [50.73(a)(2)(ii)].

TITLE: BOMB THREAT

The FBI notified the licensee's security offices regarding generic bomb threat information that they had received. A female reported to the FBI that while riding on the subway, she had overheard another passenger state that he was "going to blow up a post office or a nuclear power plant." The subject was described as a male Caucasian, approximately 70 years old, wearing several layers of clothing. Corporate security offices each notified their respective sites and gave them the subject's description. Searches were performed to determine that no one matching the above description was onsite.

Comment:

The event is not reportable under 50.73. It may be reportable as an LER under 73.71

TITLE: SUSPECTED SABOTAGE

While the plant was operating at 100% power one of the two operating steam generator feedwater pumps tripped. Rapid operator action to reduce power to 60% prevented a reactor trip. Subsequent investigation by the licensee determined that the isolation valve to the high back pressure switch on the pump's turbine exhaust to the main condensor had been shut and a vent valve in that line had been opened. As a result, feedwater pump turbine protective control circuitry saw a loss of condensor vacuum and initiated an automatic pump trip. Valve alignment was restored to normal and full power operation was resumed. No manipulation of the valves had been authorized. The licensee has concluded that this was a deliberate act to trip the plant. Normally, under these circumstances, a plant trip would have occurred.

Comment:

The event is not reportable under 50.73. It may be reportable as an LER under 73.71.

TITLE: FIRE IN AUXILIARY BUILDING

A minor fire occurred in the auxiliary building. The plant fire brigade responded and extinguished the fire in approximately five minutes. The fire occurred when a contractor workman was using a torch to cut steel rebar from a wall being torn down. A piece of hot rebar fell on plastic sheeting being used to catch debris falling to the floor. The wall was part of a room in the southeast corner of the auxiliary building used to compact low-level waste for shipment offsite. The licensee reported that initial analysis of smoke samples taken during the fire indicated no detectable amounts of radioactivity. Fire brigade members wore self-contained breathing apparatus. Two contractor employees reported inhalation of a small amount of smoke. No burns or injuries occurred. No plant safety systems were affected by the fire. Smoke from the fire was partially exhausted from the building through a door opened for that purpose and was visible from the adjacent Unit 2 construction site.

Comment:

The event would not be reportable if the fire did not pose an actual threat to the safety of the nuclear power plant or site personnel, nor significantly hamper site personnel [50.73(a)(2)(x)].

TITLE: DIESEL GENERATOR LUBE OIL FIRE HAZARD

While performing a routine surveillance test of the emergency diesel generator, a small fire started due to lubricating oil leakage from the exhaust manifold. The manufacturer reviewed the incident and determined that the oil was accumulating in the exhaust manifold due to leakage originating from above the upper pistons of this vertically opposed piston engine. The oil remaining above the upper pistons after shutdown leaked slowly down past the piston rings, into the combustion space, past the lower piston rings, through the exhaust ports, and into the exhaust manifolds. The exhaust manifolds became pressurized during the subsequent startup which forced the oil out through leaks in the exhaust manifold gaskets where it was ignited.

Similar events occurred previously at this plant. In these previous cases, fuel oil accumulated in the exhaust manifold due to extended operation under "no load" conditions. Operation under loaded conditions was therefore required before shutdown in order to burn off any accumulated oil.

Comments:

1. The event is not reportable if the fire did not pose a threat to the plant (i.e., it only affected a single component) [50.73(a)(2)(x)].

2. The event would be reportable if it demonstrates a design, procedural, or equipment deficiency that could have prevented the fulfillment of a safety function (i.e., if the redundant diesels are of similar design and, therefore, susceptible to the same problem) [50.73(a)(2)(v)].

TITLE: UNPLANNED GASEOUS RELEASE

While transferring gas from the waste gas overhead surge tank to the waste gas decay tank, a pressure relief valve lifted. An estimated 120 cubic feet of fission product gas, primarily Xenon, was released to the atmosphere through the process vent stack. The duration of the release was approximately five minutes. The licensee calculated that a total of 46.4 curies of noble gas were released. The licensee reported that the release was 2.58 times the plant Technical Specification limit for an instantaneous release, but less than 1 percent of the quarterly Technical Specification limit.

Comments:

The event would be reportable if the release exceeded 2 times the applicable concentrations specified in Appendix B, Table 2 of 10 CFR Part 20 averaged over a time period of one hour [50.73(a)(2)(viii)(A)].

TITLE: RADIATION OVEREXPOSURE OF MAINTENANCE PERSONNEL

During removal of shield plugs from the reactor pressure vessel (36 inch thick tiered plugs which rest on the upper grid following feedwater sparger replacement) a workman was exposed to approximately 21.2 REM; this exposure was confirmed by film badge data. This exposure occurred while the workman was directing the overhead crane operator during the lifting operation. Work was being performed during a refueling outage with all fuel removed from the vessel and the vessel water level below the upper core gridplate.

Comment:

The event is not reportable under 50.73. However, it is reportable as an LER under Part 20.403(b)(1) and 20.405(a)(4).

TITLE: UNPLANNED RELEASE OF RADIOACTIVE GASES IN THE AUXILIARY BUILDING

While draining the reactor coolant system to the reactor coolant drain tank, to perform maintenance on the HPI nozzles, a release of noble gases, principally Xe-133, occurred. The gas and a small quantity of liquid (approximately one gallon) escaped from a vacuum breaker in the nitrogen cover gas line located in the Auxiliary Building east decay heat cooler room. The release was initially identified by the room air monitor. The auxiliary building stack gas monitor increased to 1,000 counts per minute. Using the public address system, the licensee evacuated the minus 20-foot and minus 47-foot levels of the auxiliary building. The licensee estimates that the release rate was approximately 0.53 percent of the Technical Specification instantaneous release rate limit. The licensee stopped draining the Reactor Coolant System and plans to correct the fault with the vacuum breaker before continuing draining the system.

Comment:

1. The event is not reportable if the release did not exceed the limits in Appendix B, Table II of 10 CFR Part 20 [50.73(a)(2)(viii)].

2. The event would be reportable if the release significantly hampered site personnel in the performance of duties necessary for the safe operation of the plant [50.73(a)(2)(x)].

TITLE: CONTAMINATED MATERIAL FOUND IN ONSITE SCRAP MATERIAL AREA

During routine surveillance of an onsite scrap material storage area, which is not designated as a restricted area, licensee personnel identified several items with surface contamination. The levels ranged from 800 counts per minute to 15,000 counts per minute. The licensee performs periodic surveys of the material in the scrap area, but this survey used more sensitive instrumentation than previous surveys. There was no significant personnel radiation exposure associated with the contaminated material.

Comment:

This event is not reportable under 50.73.

SUBJECT: WORK HALTED BY LICENSEE FOLLOWING DOUSING OF QUALITY CONTROL INSPECTORS

The licensee suspended work at the construction site pending an investigation of an incident in which three quality control inspectors were doused with water. According to preliminary information, a bucket of water had been positioned over a desk in the reactor building. The bucket was then tipped over onto the QC inspectors by someone pulling a long rope attached to the bucket. The dousing incident followed instances of verbal abuse earlier in the week.

Comment:

The event is not reportable under the provisions of 50.73 because it occurred at a plant that did not yet have an operating license (i.e., a plant under construction) [50.73(a)]. The event may be reportable under 10 CFR 50.55(e), and would have been reportable if the event had occurred at a plant with an operating license [50.73(a)(2)(x)].

TITLE: LOSS OF SALT WATER COOLING SYSTEM AND FLOODING IN SALTWATER PUMP BAY

It was found that both trains of Saltwater Cooling had been lost for 24 minutes. The Saltwater Cooling System is the ultimate heat sink for the facility. The reactor was in Mode 5, depressurized with 115°F average reactor coolant system temperature at the start of the event.

During maintenance activities on the South Saltwater Pump, the licensee was removing the pump internals from the casing. Since (1) the floor of the pump bay was below sea level, (2) the seawater inlet gates were open, and (3) main circulating water pumps were secured, flooding began from the sea into the pump bay. The water level in the pump bay reached about four feet (sea level at low tide). The North Saltwater pump was secured to prevent pump damage. In addition, the licensee cross-connected the traveling screen wash water pumps to the saltwater system to reestablish saltwater cooling and terminate system temperature rise and replaced the South Saltwater pump internals in the casing to terminate flooding. The final reactor coolant system temperature was about 117 degrees F while the component cooling water temperature increased by 15 degrees F during the event to 77 degrees F.

Comment:

The event is reportable because of the failure of the Saltwater Cooling System, which is the ultimate heat sink for this facility, to perform its safety function [50.73(a)(2)(v)].

TITLE: UNPLANNED SHUTDOWN DUE TO CONDUCTIVITY

The licensee commenced an unscheduled reduction from 100 percent power following an increase in electrical conductivity of the reactor coolant water. Conductivity reached a level (10.5 umho/cm) at which plant Technical Specifications require cold shutdown within 24 hours.

The licensee reports that the cause of the higher than normal conductivity is believed to be due to the intrusion of resins from either a condensate or reactor water filter demineralizer into the reactor cooling water system.

In addition, a total of 26 local power range monitors (LPRMs) have failed due to degraded water chemistry. The licensee plans to replace all 31 LPRMs. This may require a partial unloading of the fuel. Water chemistry has improved considerably following recirculation through the reactor water cleanup demineralizers.

Comments:

1. The event is reportable because the licensee completed a shutdown required by the Technical Specifications. [50.73(a)(2)(i)(A)].

2. The event is reportable because it includes a material (e.g., metallurgical, chemical) problem that caused abnormal degradation of multiple components (i.e., 26 LPRMs) [50.73(a)(2)(vii)].

TITLE: REACTOR TRIP AND SAFETY INJECTION

While the unit was shutting down in preparation for a 10-day maintenance outage, a reactor trip and safety injection occurred. The trip and safety injection occurred after erratic turbine control resulted in a high differential pressure between the main steam line and header.

Comment:

The event is reportable because the reactor tripped and because safety injection occurred. The event is reportable even though the unit was in the process of shutting down, because conditions developed during the shutdown that required an automatic reactor trip [50.73(a)(2)(iv)].

TITLE: STUCK HIGH PRESSURE INJECTION SYSTEM CHECK VALVES

The licensee reported that check valves in three of four high pressure injection lines were stuck closed. The unit had been shut down for refueling and maintenance.

A special test of the check valves revealed that three 2½ inch stop check valves remained closed when 130 pounds per square inch (psi) of differential pressure was applied to the valve. An additional test was performed which revealed the failure of the valve to open when 400 psi of differential pressure (the capacity of the pump) was applied to the valve.

The licensee is currently reviewing additional testing and inspection procedures to open the valves and determine why they have failed to open.

Comment:

The event is reportable because a single cause or condition caused at least two independent trains of the HPI system to become inoperable [50.73(a)(2)(vii)].

TITLE: FIRE IN MAIN GENERATOR EXCITOR

The licensee reported a fire in the Main Generator Excitor housing. The reactor was manually tripped. All systems responded as designed and the reactor remained in a stable condition. The reactor will be taken to the cold shutdown mode. The station fire brigade responded to the fire and successfully extinguished the fire. No offsite firefighter assistance was required. Smoke from the fire was released to the environment via the Turbine Building. There were no radioactive releases or injuries to plant personnel.

Comment:

1. The event is reportable if it threatened the safety of the nuclear power plant (e.g., the fire was sufficiently severe to require a manual trip) [50.73(a)(2)(x)].

2. The event is reportable because the reactor wa. manually tripped [50.73(a)(2)(iv)].

TITLE: WATER SPILL

The licensee reported that a seal on a condensate system water box manway located on the secondary side of the main condenser, ruptured, allowing approximately 150,000 gallons of river water to flood the turbine and radioactive waste buildings. The basement floor of the radioactive waste building was covered with about six inches of water. Approximately one and one-half inches of water covered the floor of the turbine building. The water, drawn from the river and used to condense steam after it has passed through the turbines, picked up small amounts of radioactive contamination from the flooded basement area, but none was released to the environment. The water is being processed through the radioactive waste process system. The plant, operating at 20 percent power at the time of the seal failure, was manually scrammed.

Comment:

1. The event is reportable because the magnitude of the flooding posed a threat to the safety of the nuclear power plant [50.73(a)(2)(x)].

2. The event is reportable because the reactor was manually scrammed [50.73(a)(2)(iv)].

TITLE: OVERPRESSURIZATION OF THE REACTOR COOLANT SYSTEM

The reactor coolant system (RCS) was overpressurized on two occasions during startup following a refueling outage. The reactor was shut down and the RCS was in a water solid condition with a pressure and temperature of approximately 310 psig and 110°F, respectively. Two overpressure conditions of 1100 and 750 psig at 110°F developed for which the overpressure mitigating system (OMS) failed to operate. These events exceeded the pressure limit of 400 psig at 110°F specified in technical specifications which prescribe the allowable pressure and temperature limits to prevent reactor vessel brittle fracture.

The OMS is specifically designed to prevent this type of overpressurization. The reason the OMS did not operate as designed was as follows: (1) A pressure transmitter was unintentionally isolated. This transmitter provides input into the OMS circuit to automatically open a power operated relief valve (PORV) on high pressure conditions; (2) a summator failed on the electrical circuitry which prescribed the pressure at which the CMS is to initiate PORV actuation; and (3) the redundant OMS circuit was out of service for calibration. The transmitter isolation valve was found closed and was opened after the first event. The failed summator was identified and corrected after the second event.

During both occurences, the operator took action to stop the charging pump which was providing the source of rapid pressurization. RCS charging and letdown flow was in progress prior to each event. However, once the letdown was significantly reduced or terminated by closure of the residual heat removal system isolation valve, timely operator action to prevent the overpressurization was precluded by the rapidity of the transient. The operator decreased the pressure to the desired level within two minutes by manually operating the PORV.

Comment

1. The event is reportable because the overpressurization violated the Technical Specification limits [50.73(a)(2)(i)(B)].

2. The event is reportable because the Overpressure Mitigation System (OMS) failed to perform its intended function [50.73(a)(2)(v)].

TITLE: FAILED COMPONENTS ON DIESEL GENERATORS

The licensee reported a failure of the shaft-driven cooling pump on emergency diesel generator No. 2, caused by sheared cap screws and broken dowel pins in the pump coupling:

Metallurgical examination of cap screws and dowell pins in diesel generator No. 2 indicated that the failure was probably due to metal fatigue brought about by the number of starts (1400) of the generator since installation. As a result of these findings, the diesel generator manufacturer, recommended that coupling cap screws and dowel pins be replaced after 1200 starts.

The licensee replaced cap screws and dowel pins on diesel generator numbers 2 and 4, where broken screws and cracked dowel pins were found after disassembly. The disassembly of diesel generator No. 1 also disclosed some broken cap screws. Diesel generator No. 3 was disassembled and checked in the same fashion. The licensee stated that such components on all four diesel generators would be replaced before the units returned to power.

Comment:

1. The event is reportable because a single cause (fatigue failure of broken cap screws) caused at least two trains of a system (emergency electrical power) to be inoperable [50.73(a)(2)(vii)].

2. The event is reportable if this condition could have prevented fulfillment of a safety function [50.73(a)(2)(v)].

TITLE: RADIOACTIVE RELEASE EXCEEDING TECHNICAL SPECIFICATIONS

The controlled radioactive gaseous release rate limit was exceeded for about 30 seconds during sampling of the waste gas stripper surge drum at the primary sample sink. An estimated 3.2 curies of mostly Xenon-133 was released, exceeding the controlled release rate listed in plant Technical Specifications by about 60 percent. No particulate or Iodine radioactivity was released. The licensee estimated that the dose rate at the site boundary for a short period of time was 0.07 millirem per hour. An individual standing at the site boundary would have received an estimated whole body dose of 0.001 millirem.

Comment:

1. The event is reportable because a Technical Specification limit was violated [50.73(a)(2)(i)(B)].

2. The event is not reportable under 50.73(a)(2)(viii) if the release did not exceed two times the applicable limit of Appendix B, Table II, of Part 20 when averaged over one hour.

TITLE: REACTOR COOLANT SYSTEM LEAKAGE

During power operation, the results of a surveillance test indicated reactor coolant leakage of 2.09 gpm. A subsequent test indicated leakage of 1.62 gpm. The source of the leak was identified as valve packing on the Loop A Hot Leg RTD Manifold Outlet Isolation Valve. The cause of the leak was a failure of the gland flange on the valve. A split gland flange was used as a modification to repair the valve and reduce the possibility of another failure. The plant was placed in Hot Standby within 6 hours in accordance with the plant's Technical Specific• ions.

Comment:

The event is reportable because the plant completed a shutdown that was required by it's Technical Specifications [50.73(a)(2)(i)(A)].

TITLE: PRESSURIZER LEVEL DEVIATION FROM PROGRAM LEVEL

During a normal start up, pressurizer level deviated from program level by more than +/-5 percent during changes in RCS temperature. The CVC system operated normally during the event to assist in controlling pressurizer level. Pressurizer level was returned to program and verified steady.

The cause for level fluctuation was normal start up operations. The pressurizer is designed to accommodate in and out surges in response to changes in Reactor Coolant System (RCS) temperature. The pressurizer level control system normally operates to control level within -15 to +39 inches of program level during transient conditions. The Technical Specifications require pressurizer level to be within 5% of program. This Technical Specification requirement only allows a 7-inch deviation at low load conditions and an 11-inch deviation at full load conditions from program level. During low power operation, Reactor Coolant System temperature swings of ten degrees are not uncommon while paralleling the main turbine to the off-site power system or controlling steam generator levels in manual. These temperature swings can result in deviations of greater than 5% from program pressurizer level.

Comment:

This event is reportable because it is a violation of the plant's Technical Specifications [50.73(a)(2)(i)(B)].

TITLE: OPERATOR ACTION TO INHIBIT THE REACTOR PROJECTION SYSTEM

With the unit in mode 5 (95°F and 0 psig prior to initial criticality) and a post-modification test in progress on the train A Reactor Protection System, the operator observed that both train A and train B source range detectors were disabled. During post-modification testing on train A Reactor Protection System, instrumentation personnel placed the train B input error inhibit switch in "inhibit." With both trains' input error inhibit switches in "inhibit," source range detector voltage is disabled. The input error inhibit switch was immediately returned to normal and a caution was added to appropriate plant instructions.

Comment:

The event is reportable because the actions would have prevented fulfillment of a safety function [50.73(a)(2)(v)].

TITLE: CONTAMINATED HYDRAULIC FLUID DEGRADES MSIV OPERATION

During a routine shutdown, the operator noted that the #11 MSIV closing time appeared to be excessive. A subsequent test revealed the #11 MSIV shut within the required time, however, the #12 MSIV closing time exceeded the maximum at 7.4 sec. Contamination of the hydraulic fluid in the valve actuation system had caused the system's check valves to stick and delay the transmission of hydraulic pressure to the actuator. Three more filters will be purchased providing supplemental filtering for each MSIV. Finer filters will be used in pump suction filters to remove the fine contaminants. The #12 MSIV was repaired and returned to service. Since the valves were not required for operation at the time of discovery the safety of the public was not affected.

Comments:

1. The event is reportable because a single condition could have prevented fulfillment of a safety function [50.73(a)(2)(v)].

2. The fact that the condition was discovered when the valves where not required for operation does not affect the reportability of the condition.

TITLE: MANUAL SCRAM DURING NORMAL SHUTDOWN

During a normal reactor shutdown for an extended refueling outage, the reactor was scrammed from 30% power in order to speed the shutdown by passing the requirements imposed by the Rod Sequence Control System (PSCS). The scram is specifically required in the shutdown procedure and was not necessitated by failure of equipment or because of any Technical Specification requirement.

Comment:

Although the scram was an actuation of the Reactor Protection System (RPS), the scram was "part of a preplanned sequence during testing or operation" and therefore need not be reported [50.73(a)(2)(iv)].

TITLE: HIGH ENERGY LINE BREAK RESTRAINTS NOT INSTALLED

While at power, the plant was notified by the Architect/Engineer that the control rod drive system, raeactor water cleanup system, reactor core isolation cooing system, and auxiliary steam system did not hove the high energy line break restraints required in the Final Safety Analysis Report (FSAR) supplement 15A. The high energy line break restraints were omitted from four plant systems during construction due to Architect/Engineer design oversight.

Comment:

The event is reportable because it is an event or condition that was outside the design basis of the plant and is an unanalyzed condition that significantly compromised plant safety [50.73(a)(2)(ii)(A) and (B)].

TITLE: HVAC WELDS FOUND DEFECTIVE

During a refueling outage, it is discovered that all of the HVAC welds are undersized or are otherwise inadequate because of a major breakdown in quality assurance (QA) during construction that occurred years earlier. As far as the licensee knows, the NRC has not identified the problem at this particular plant. The HAVAC system is used to provide control room habitability during accident conditions and therefore has a safety function.

Comment:

The event is reportable because it is an event or condition that was outside the design basis of the plant and is an unanalyzed condition that significantly compromises plant safety [50.73(a)(2)(ii)(A) and (B)].

NRC FORM 335

U.S. NUCLEAR REGULATORY COMMISSION

**BIBLIOGRAPHIC DATA SHEET**

1 REPORT NUMBER (Assigned by DDC)

NUREG-1022

4 TITLE AND SUBTITLE (Add Volume No., if appropriate)

LICENSEE EVENT REPORT SYSTEM
Description of System and Guidelines for Reporting

2 (Leave blank)

3 RECIPIENT'S ACCESSION NO.

7 AUTHOR(S)

F. J. Hebdon

5 DATE REPORT COMPLETED

| MONTH | YEAR |
|---|---|
| September | 1983 |

9 PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Office for Analysis and Evaluation of Operational Data
U. S. Nuclear Regulatory Commission
Washington, DC 20555

DATE REPORT ISSUED

| MONTH | YEAR |
|---|---|
| September | 1983 |

6 (Leave blank)

8 (Leave blank)

12 SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)

Same as 9. above

10 PROJECT/TASK/WORK UNIT NO.

11 FIN NO.

13 TYPE OF REPORT

Guide

PERIOD COVERED (Inclusive dates)

15 SUPPLEMENTARY NOTES

14 (Leave blank)

16 ABSTRACT (200 words or less)


On July 26, 1983, the Commission published in the Federal Register a final rule that modifies and codifies the Licensee Event Report (LER) system. The rule becomes effective on January 1, 1984. This NUREG provides supporting information and guidance that will be of interest to persons responsible for the preparation and review of LERs. The information contained in this NUREG includes: (1) a brief description of how LERs are used by the NRC, (2) a restatement of the guidance contained in the Statement of Consideration that accompanied the publication of the LER rule, (3) a set of examples of potentially reportable events with staff comments on the actual reportability of each event, (4) guidance on how to prepare an LER, including the LER form, and (5) guidance on the submittal of LERs.


17 KEY WORDS AND DOCUMENT ANALYSIS

17a DESCRIPTORS

17b IDENTIFIERS/OPEN-ENDED TERMS

18 AVAILABILITY STATEMENT

Unlimited

19 SECURITY CLASS (This report)

Unclassified

20 SECURITY CLASS (This page)

Unclassified

21 NO. OF PAGES

22 PRICE