June 20, 2000

Mr. Valeri Tolstykh
Regulatory Activities Unit
Safety Assessment Section
Division of Nuclear Installation Safety
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100, A-1400
Vienna, Austria

Dear Mr. Tolstykh:

Enclosed are the following IRS reports:

- OFFSITE POWER VOLTAGE INADEQUACIES (NRC Information Notice 2000-06).

- INADEQUATE ASSESSMENT OF THE EFFECT OF DIFFERENTIAL TEMPERATURES ON SAFETY-RELATED PUMPS (NRC Information Notice 2000-08).

- USE OF RISK-INFORMED DECISIONMAKING IN LICENSE AMENDMENT REVIEWS (NRC Regulatory Issue Summary 2000-07).

Each report is being submitted in the following two media: (1) a hard copy of the input file for the AIRS database; and (2) a 3.5-inch HD diskette containing the input file for the AIRS database in Microsoft Word 6.0 format.

If you have any questions regarding these reports, please call Eric J. Benner of my staff. He can be reached at (301) 415-1171.

Sincerely,

/RA/
Ledyard B. Marsh, Chief
Events Assessment, Generic Communications and
Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Enclosures as stated

cc w/enclosures 1 and 2:
Mr. Lennart Carlsson
Nuclear Safety Division
Nuclear Energy Agency
Organization for Economic
Cooperation and Development
Le Seine Saint Germain
12, Boulevard des Iles
92130, Issy-les-Moulineaux, France

June 20, 2000

Mr. Valeri Tolstykh
Regulatory Activities Unit
Safety Assessment Section
Division of Nuclear Installation Safety
International Atomic Energy Agency
Wagramer Strasse 5
P.O. Box 100, A-1400
Vienna, Austria

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INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/03/27	

EVENT TITLE

OFFSITE POWER VOLTAGE INADEQUACIES (NRC Information Notice 2000-06)

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	(BWR or PWR)

INITIAL STATUS	RATED POWER (MWe NET)
N/A	N/A

DESIGNER	1st COMMERCIAL OPERATION
(WEST, GE, CE, B&W)	N/A

ABSTRACT

This IRS report discusses experience related to a possible concern regarding the voltage adequacy of offsite power sources, that is, power from the transmission system grid to nuclear power plants. NRC inspection findings and licensee event reports have indicated instances in which grid stability analyses had not been updated by the licensees to reflect changes in the grid power system. An Office of Nuclear Regulatory Research report, “The Effects of Deregulation of the Electric Power Industry on the Nuclear Plant Offsite Power System: An Evaluation,” dated June 30, 1999, recommended that the staff take certain followup actions to ensure that licensees will continue to maintain their licensing bases in this area. Industry deregulation can heighten the need to update the analyses on a more frequent basis. Some utilities have utilized on-line contingency analysis techniques in their grid control centers and implemented arrangements to be notified when the offsite system to their plant is in jeopardy of not providing its required capability. When the on-line capability is not available, other utilities have provided for updating of the analyses on a more frequent basis and have implemented procedures to identify when the plant and grid conditions are outside the bounds of the assumptions of the analyses, thereby providing the information to take compensatory actions as necessary.

OFFSITE POWER VOLTAGE INADEQUACIES (NRC Information Notice 2000-06)

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	**Reporting Categories:**	1.3	1.4	1.6
2.	**Plant Status Prior to the Event:**	2.1		
3.	**Failed/Affected Systems:**	3.EA		
4.	**Failed/Affected Components:**	4.0		
5.	**Cause of the Event:**	5.1.2.7		
6.	**Effects on Operation:**	6.4	6.10	
7.	**Characteristics of the Incident:**	7.0		
8.	**Nature of Failure or Error:**	8.0		
9.	**Nature of Recovery Actions:**	9.0		

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

March 27, 2000

NRC INFORMATION NOTICE 2000-06: OFFSITE POWER VOLTAGE INADEQUACIES

Addressees

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees of experience related to a possible concern regarding the voltage adequacy of offsite power sources, that is, power from the transmission system grid to nuclear power plants. It is expected that recipients will review the information for applicability to their facilities and consider actions as appropriate to avoid similar problems. No specific action or written response is required by this notice.

Description of Circumstances

On August 11, 1999, the Callaway plant experienced a rupture of a reheater drain tank line. As a result, the plant operators initiated a manual reactor trip. Since the plant was shutdown, offsite power was required to supply the plant equipment loads. During this period, the grid conditions were such that a substantial power flow was occurring from north to south through the local Callaway grid. The licensee stated that the deregulated wholesale market contributed to conditions in which higher grid power flows are likely to occur. The licensee stated that these large flows were observed at this time. This power flow, coupled with a high local demand and the loss of the Callaway generator, resulted in switchyard voltage at the site dropping below the minimum requirements for 12 hours. Although offsite power remained available during the reactor trip transient, the post-trip analysis indicated that in the event that additional onsite loads would have been in operation at the time of the event, 4-16 kV distribution voltage may have decreased below the setpoint of the second-level undervoltage relays separating the loads from offsite power. The NRC conducted a special inspection at Callaway from November 29 to December 3, 1999, on the circumstances surrounding the event. The inspectors found that similar conditions existed in 1995 that were undetected by the licensee (Licensee Event Report (LER) 50-483/99-005 (Accession No. 9909200074); NRC Inspection Report 50-483/99-15 (Accession No. ML003684343), dated February 15, 2000).

The following events identify additional combinations of main generator unavailability, line outages, transformer unavailability, high system demand, unavailability of other local voltage support, and high plant load that could result in inadequate voltages. Common among all the events is the inability to predict the inadequate voltages through direct readings of plant switchyard or safety bus voltages, without also considering grid and plant conditions and their associated analyses.

On July 11, 1989, safety systems at Virgil C. Summer Nuclear Station experienced a sustained degraded voltage condition and, as a result, the safety buses were automatically transferred from the offsite power system to onsite standby diesel generators. The degraded condition was caused by a turbine trip and deficiencies in the offsite power system's transmission network equipment. The transfer of power supplies was initiated by operation of degraded voltage protective relays, as designed. Nonsafety system loads remained operable while being powered for approximately 1 hour from the degraded offsite power source (LER 50-395/89-012 (Accession No. 8908140351)).

On November 5, 1991, the licensee for Arkansas Nuclear One, Units 1 and 2, reported that had its 500-kV auto-transformer been lost during summer peak conditions, the 161-kV system might not have been able to maintain adequate voltages to support the operation of the safety system loads of both units (LER 50-313/91-010 (Accession No. 9111150021)).

On December 30, 1993, Northeast Nuclear Energy Company concluded that with the switchyard at the worst case minimum voltage, Millstone Nuclear Power Station, Unit 1, loss-of-coolant accident (LOCA) mitigation loads could combine with normal loads that are not shed upon receipt of an accident signal to produce a voltage drop that would actuate degraded voltage relays resulting in separation from offsite power. The utility determined that this worst case minimum switchyard voltage could occur after the loss of Millstone Unit 1 generation when both Millstone Units 2 and 3 are off-line (LER 50-245/94-01 (Accession No. 950920001)).

On February 6, 1995, the licensee for Palo Verde Nuclear Generating Station, Units 1, 2, and 3, reported shortcomings in the plant site voltage regulation. Specifically, if a LOCA occurred with the switchyard voltage in the lower two-thirds of its operating range, the engineered safety feature (ESF) loads would begin sequencing onto the preferred offsite power source, and the house loads would fast transfer to the startup transformer following the main generator or turbine trip that would accompany the LOCA. The resulting voltage drops at the safety buses would cause the bus degraded voltage relays to drop out during the ESF load sequencing and subsequently resequence the loads onto the diesel generators. The licensee identified this scenario as “double sequencing” (LER 50-528/93-011-01 (Accession No. 9502160195)).

On August 8, 1995, Pacific Gas & Electric Company (PG&E) reported that during peak system loading, all transmission lines and a local fossil power plant (Morro Bay) needed to be in service to meet Diablo Canyon Nuclear Power Plant voltage requirements. A review of the available data by PG&E on the offsite power supplies identified 47 instances in which the system configuration could have resulted in a degraded voltage condition between 1990 and 1995. PG&E identified a potential “double sequencing” scenario at Diablo Canyon if a LOCA occurred during these degraded voltage conditions (LER 50-275/95-007-01 (Accession No. 9608140037)).

On July 22, 1997, the licensee for Clinton Power Station sought an exemption from offsite power regulatory requirements because of its analysis that offsite power would become inadequate under certain summer peak conditions following the loss of the nuclear unit. The exemption request was eventually withdrawn by the licensee.

NRC inspection findings and licensee event reports have indicated instances in which grid stability analyses had not been updated by the licensees to reflect changes in the grid power system. An Office of Nuclear Regulatory Research report, “The Effects of Deregulation of the Electric Power Industry on the Nuclear Plant Offsite Power System: An Evaluation,” dated June 30, 1999 (Accession No. 9907120008), recommended that the staff take certain followup actions to ensure that licensees will continue to maintain their licensing bases in this area.

Discussion

NRC Information Notice (IN) 98-07 discussed the possibility that the changes occurring as a result of deregulation of the electric utility industry could affect the reliability of the offsite power systems in nuclear power plants. Offsite power problems highlighted in licensee event reports were identified as potential sources of concern if not properly managed following the restructuring that occurs as a result of deregulation. NRC IN 95-37 alerted licensees to circumstances that could result in inadequate offsite power system voltages during design basis events.

The most recent problem, which was reported by the licensee for Callaway Unit 1, potentially tied the inadequate offsite system voltage problem to industry deregulation. The licensee stated in LER 50-483/99-005 (Accession No. 9909200074) that the magnitude of the power being transported across the grid during the period had not been previously observed and was far in excess of typical levels. LER 50-483/99-005 (Accession No. 9909200074) also stated that the deregulated wholesale power market contributes to conditions in which higher grid power flows are likely to occur, and these large flows were observed at this time.

Because the Callaway generator was supporting the grid voltage in the vicinity of the plant, the low grid voltage had not been observed until the Callaway generator voltage support was no longer available. However, if a design basis event had occurred during the period of high system demand, the consequential loss of the Callaway generator, combined with the plant electrical requirements associated with the event, could have actuated the plant’s degraded voltage protection and separated safety loads from offsite power, which is the preferred power supply under these circumstances.

The reports referenced in this notice also identify additional combinations of circumstances than those seen at Callaway that could result in inadequate offsite voltages. These circumstances include main generator unavailability, line outages, transformer unavailability, high system demand, unavailability of other local voltage support, and high plant load. The common characteristic of these problems is that the true capability of the offsite source cannot necessarily be verified through direct readings of plant switchyard or safety bus voltages.

Instead, analyses of grid and plant conditions must be relied upon to determine this capability, considering the postulated occurrence of an event. If these analyses are not accurate and up to date, licensees could inadvertently operate their plants in regions of inadequate voltages for some periods of time.

As demonstrated by the Callaway event, industry deregulation can heighten the need to update the analyses on a more frequent basis. Some utilities have utilized on-line contingency analysis techniques in their grid control centers and implemented arrangements to be notified when the offsite system to their plant is in jeopardy of not providing its required capability. When the on-line capability is not available, other utilities have provided for updating of the analyses on a more frequent basis and have implemented procedures to identify when the plant and grid conditions are outside the bounds of the assumptions of the analyses, thereby providing the information to take compensatory actions as necessary.

Maintaining plant operation in a region of adequate offsite voltage is especially important for licensees that may not have evaluated their plant safety systems for the double-sequencing scenario identified in the Palo Verde and Diablo Canyon LERs. The safety consequences that would result if an event occurred during a period of inadequate voltage can, therefore, be difficult to assess.

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

/RA/
Ledyard B. Marsh, Chief
Events Assessment, Generic Communications
and Non-Power Reactors Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

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INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/05/15	

EVENT TITLE

INADEQUATE ASSESSMENT OF THE EFFECT OF DIFFERENTIAL TEMPERATURES ON SAFETY-RELATED PUMPS (NRC Information Notice 2000-08)

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	(BWR or PWR)

INITIAL STATUS	RATED POWER (MWe NET)
N/A	N/A

DESIGNER	1st COMMERCIAL OPERATION
(WEST, GE, CE, B&W)	N/A

ABSTRACT

This IRS report discusses two events that appear to have been caused by inadequate engineering design assessment of the effect of differential temperatures on safety-related pumps. Safety-related pumps are expected to operate under a wide range of environmental conditions. These two events highlight the importance of assessing the effects of differential temperatures on safety-related pump operability. In addition, these events highlight the importance of having test programs that include suitable qualification testing under the most adverse design conditions (e.g., temperature and differential temperature), when the test program is used to verify the adequacy of a specific design feature (e.g. seal water supply).

INADEQUATE ASSESSMENT OF THE EFFECT OF DIFFERENTIAL TEMPERATURES ON SAFETY-RELATED PUMPS (NRC Information Notice 2000-08)

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	**Reporting Categories:**	**1.2.5**	**1.2.6**	
2.	**Plant Status Prior to the Event:**	**2.0**		
3.	**Failed/Affected Systems:**	**3.BG**	**3.CB**	
4.	**Failed/Affected Components:**	**4.2.1**		
5.	**Cause of the Event:**	**5.1.1.8**		
6.	**Effects on Operation:**	**6.0**		
7.	**Characteristics of the Incident:**	**7.5**		
8.	**Nature of Failure or Error:**	**8.3**		
9.	**Nature of Recovery Actions:**	**9.0**		

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

May 15, 2000

NRC INFORMATION NOTICE 2000-08: INADEQUATE ASSESSMENT OF THE EFFECT OF DIFFERENTIAL TEMPERATURES ON SAFETY-RELATED PUMPS

Addressees

All holders of operating licenses for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to inform addressees of the potential for differential temperature conditions to affect the operability of safety-related pumps. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

The following describes two events that appear to have been caused by inadequate engineering design assessment of the effect of differential temperatures on safety-related pumps.

Arkansas Nuclear One, Unit 1 (ANO-1)

In 1992, the licensee for ANO-1 implemented a design change to replace the cast iron inboard and outboard bearing housings on the low-pressure injection/decay heat removal (LPI/DHR) pumps with stainless steel for improved service water corrosion resistance. The LPI/DHR system is designed to remove decay heat from the core and sensible heat from the reactor coolant system (RCS) during the last stages of a plant cooldown. It also provides a means of automatically injecting borated water into the reactor vessel for cooling the core in the event of a loss-of-coolant accident during power operation. During the September 1999 refueling outage, the licensee implemented a design change to increase the viscosity of the lubricating oil for the LPI/DHR pump bearings in order to reduce wear.

On February 5, 2000, ANO-1 began cooling down the plant in preparation for entering a maintenance outage to install replacement parts on the “D” reactor coolant pump anti-rotation device. When the RCS temperature had been reduced to 280°F and the pressure had been reduced to 240 psig, the “A” LPI/DHR pump was placed in service for decay heat removal. After 52 minutes, the licensee was forced to secure the “A” LPI/DHR pump when the inboard bearing temperature reached approximately 160°F. The licensee then placed the “B” LPI/DHR pump in service for DHR but stopped it after 16 minutes due to a high inboard bearing temperature. The licensee tested both pumps by recirculating water from the borated water storage tank and noted that the bearing temperatures remained stable at approximately 80°F. During this test the pumped fluid, the borated water, was at ambient temperature. Upon switching the “B” pump suction back to the RCS, the bearing temperature again rose to approximately 160°F. In this instance the pumped fluid, the RCS water, was at a temperature of approximately 250°F.

On February 6, 2000, the licensee changed the “A” LPI/DHR pump bearing oil back to the original (lower viscosity) specification. When the “A” LPI/DHR pump was placed back in the DHR mode of operation, the bearing temperature stabilized at 119°F. The licensee then declared the LPI/DHR pump operable for the DHR mode only and proceeded to cool down the plant. The licensee then changed the “B” LPI/DHR pump bearing oil back to original specification. But, unlike the “A” pump, the “B” pump again had to be shut down due to high bearing temperature. Inspection of the “B” pump following shutdown indicated that the inboard bearing had to be replaced due to abnormal wear. Further details were provided by the licensee in its Licensee Event Report (LER) 50-313/2000-002-00 dated March 6, 2000 (Accession No. ML003691450).

Beaver Valley Power Station, Unit 1

On February 8, 2000, the licensee for the Beaver Valley Power Station, Unit 1 (the licensee), was performing a routine surveillance on the “B” river water pump. The pump tripped on over current protection after approximately 3 seconds. A few hours later, startup of the “C” river water pump was attempted and it also tripped after 3 seconds because of over current protection.

The licensee determined that the over current trips were a result of pump binding. The cause of the binding was thermal expansion of the pump shaft as a result of a temperature differential between the river water (35 degrees F) and an elevated seal injection water temperature (70 degrees F). The river water pump seal water was being supplied by the non-safety related filtered water system. At the time of the event, the filtered water system was in an abnormal configuration that created the elevated water temperature. Further details on this event are available in LER 50-334/2000-002-00 dated March 8, 2000 (Accession No. ML003692855), LER 50-334/2000-002-01 dated April 27, 2000 (Accession No. ML003712023), and in NRC Inspection Reports 05000334/2000-01 dated March 17, 2000 (Accession No. ML003693247), and 05000334/2000-02 dated April 28, 2000 (Accession No. ML003709259).

Discussion

At ANO, the NRC performed a special inspection (report number 50-313/00-04; 50-368/00-04, Accession No. ML003708466) to follow up on the events which led to declaring both Unit 1 LPI/DHR pumps inoperable. The inspectors concluded that the failure to complete adequate engineering evaluations for the replacement of the cast iron bearing housing with a stainless steel housing and the change in lubricating oil viscosity resulted in the inoperability of both

LPI/DHR pumps. The changes in the bearing housing material and use of a higher viscosity oil, in combination with low cooling water temperatures (<42°F), resulted in both low pressure injection/decay heat removal pumps operating with high bearing temperatures, which required the pumps to be secured. From January 28 to February 5, 2000, when the cooling water temperature was 42°F or less, both low pressure injection/decay heat removal pumps were not operable as they could not perform their intended safety function. These design deficiencies were not identified by post modification or surveillance testing. Testing performed by recirculating water from the borated water storage tank did not duplicate actual operational conditions because the pumped fluid (from the borated water storage tank) was at a much lower temperature than the RCS.

Subsequent investigation by the ANO licensee identified other potentially susceptible equipment in both units and took appropriate corrective actions.

At the Beaver Valley Power Station, the licensee determined that when warmer seal water is provided to an idle pump during cold river water conditions, the warmer seal water travels down the pump shaft and increases the shaft temperature. The pump casing is not in direct contact with the seal water and, therefore, is not affected by the increase in seal water temperature. This temperature differential resulted in elongation of the pump shaft, impeller contact with the pump casing, and eventual pump binding. The same warmer seal water supplied to the pumps when they are idle is also supplied to them when they are operating. However, the effect of having warmer seal water supplied to an operating pump was negligible because of the extremely large volume of pumped fluid acting as a heat sink on the small volume of seal water passing through the pump inner column. The licensee also determined that the filtered water system could introduce a common-mode failure to all three safety-related river water pumps. The filtered water system was subsequently isolated as a supply source to the river water pumps and the pumps were operated from their safety related supply.

During this operation, the licensee identified an inadequacy in the design of the safety-related seal water supply strainers. Since original plant operation in 1976, the non-safety-related filtered water system had been the primary supply to the river water pump seals. However, during operation of the pumps on their safety-related supply, the safety-related in line strainers fouled during high silt conditions.

Safety-related pumps are expected to operate under a wide range of environmental conditions. These two events highlight the importance of assessing the effects of differential temperatures on safety-related pump operability. In addition, these events highlight the importance of having test programs that include suitable qualification testing under the most adverse design conditions (e.g., temperature and differential temperature), when the test program is used to verify the adequacy of a specific design feature (e.g. seal water supply).

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

/RA/

Ledyard B. Marsh, Chief
Events Assessment, Generic Communications
and Non-Power Reactors Branch
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Office of Nuclear Reactor Regulation

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INCIDENT REPORTING SYSTEM

IRS NO.	EVENT DATE	DATE RECEIVED
	2000/03/28	

EVENT TITLE

USE OF RISK-INFORMED DECISIONMAKING IN LICENSE AMENDMENT REVIEWS
(NRC Regulatory Issue Summary 2000-07)

COUNTRY	PLANT AND UNIT	REACTOR TYPE
USA	Generic	(BWR or PWR)

INITIAL STATUS	RATED POWER (MWe NET)
N/A	N/A

DESIGNER	1st COMMERCIAL OPERATION
(WEST, GE, CE, B&W)	N/A

ABSTRACT

This IRS report discusses interim guidance on the use of risk information by the staff in its license amendment reviews, including reviews of license amendment requests that are not risk informed, and staff plans for finalizing this guidance. Upon receipt of a license, the staff will question risk further if there is a reason to believe that the proposed change would compromise the safety principles described in Regulatory Guide (RG) 1.174 and would substantially increase risk relative to the risk acceptance guidelines contained in the regulatory guide. In such instances, the staff will ask the licensee to address the safety principles and the numerical guidelines for acceptable risk increases contained in RG 1.174 in the submittal. The staff may ask the licensee to submit the information it needs to make an appropriate risk assessment. If an applicant does not choose to address risk, the NRC staff will not issue the requested amendment until it has assessed the risk implications sufficiently to determine that there is reasonable assurance that the public health and safety will be adequately protected if the amendment request is approved. A licensee's decision not to submit requested information could impede the staff's review and could also prevent the staff from reaching a finding that there is reasonable assurance of adequate protection.

USE OF RISK-INFORMED DECISIONMAKING IN LICENSE AMENDMENT REVIEWS
(NRC Regulatory Issue Summary 2000-07)

Please refer to the dictionary of codes corresponding to each of the sections below and to the coding guidelines manual.

1.	**Reporting Categories:**	**1.4**		
2.	**Plant Status Prior to the Event:**	**2.0**		
3.	**Failed/Affected Systems:**	**3.Z**		
4.	**Failed/Affected Components:**	**4.0**		
5.	**Cause of the Event:**	**5.1.0**		
6.	**Effects on Operation:**	**6.0**		
7.	**Characteristics of the Incident:**	**7.0**		
8.	**Nature of Failure or Error:**	**8.0**		
9.	**Nature of Recovery Actions:**	**9.0**		

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

March 28, 2000

NRC REGULATORY ISSUE SUMMARY 2000-07 USE OF RISK-INFORMED DECISIONMAKING IN LICENSE AMENDMENT REVIEWS

Addressees

All holders of operating licenses for nuclear power reactors, except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel.

Intent

The U.S. Nuclear Regulatory Commission (NRC) is issuing this regulatory issue summary (RIS) to advise addressees of interim guidance on the use of risk information by the staff in its license amendment reviews, including reviews of license amendment requests that are not risk informed, and staff plans for finalizing this guidance. This RIS requires no action or written response on the part of an addressee.

Background Information

Commission policy, as presented in the Probabilistic Risk Assessment Policy Statement and the “Discussion on Safety and Compliance” (COMSAJ-97-008), indicates that it is the staff’s responsibility to consider the change in risk, as well as compliance with the agency’s regulations and other requirements, when reviewing license amendment requests. The use of risk information is clear when the action is a risk-informed license amendment request. However, the staff’s responsibilities and authority for considering risk information and the Commission’s policy regarding the use of risk information in regulatory decisionmaking are not explicitly stated or defined for license amendment requests that are not risk informed (i.e., their acceptability is based solely on meeting the Commission’s deterministic rules and regulations).

The recent technical review of steam generator electrosleeves discussed in SECY-99-199, “Electrosleeve Amendment Issued to Union Electric Company for Callaway Plant, Unit 1," illustrates the difficulty of completing a review of a proposed license amendment request that is not risk informed and that satisfies existing design and licensing bases but introduces new potential risks. As a result of this experience, the staff proposed an approach for applying risk informed decisionmaking in similar technical reviews in SECY-99-246, “Proposed Guidelines for Applying Risk Informed Decisionmaking in License Amendment Reviews.” In the related staff

ML003680058

requirements memorandum, the Commission approved the approach and its implementation on an interim basis while the staff proceeds to engage stakeholders in the development of final guidance.

This RIS transmits the interim guidance on the use of risk information in regulatory decisionmaking regarding license amendment requests and describes the planned approach for finalizing this guidance.

Summary of Issue

When a license amendment request complies with the regulations and other license requirements, there is a presumption by the Commission of adequate protection of public health and safety (Maine Yankee, ALAB-161, 6 AEC 1003 (1973)). However, circumstances may arise in which new information reveals an unforeseen hazard or a substantially greater potential for a known hazard to occur, such as identification of a design vulnerability or an issue that substantially increases risk. In such situations, the NRC has the statutory authority to require licensee action above and beyond existing regulations to maintain the level of protection necessary to avoid undue risk to public health and safety. Section 182.a of the Atomic Energy Act of 1954, as amended, and as implemented by 10 CFR 2.102, gives the NRC the authority to require the submittal of information in connection with a license amendment request if NRC has reason to question adequate protection of public health and safety. The applicant may decline to submit such information, but it would risk having the amendment request denied if NRC cannot find that the requested amendment provides adequate protection of public health and safety.

Under unusual circumstances that could introduce significant and unanticipated risks, the NRC staff would assume the burden of demonstrating that protection is not adequate or that additional license conditions are justified despite the fact that current regulatory requirements appear to be met. Instances in which the staff would question licensees regarding risk are expected to be relatively rare.

The guidelines presented in SECY-99-246 for identifying those situations in which risk implications are appropriate to consider and for deciding if undue risk exists are described in Attachment 1 to this RIS. These guidelines will be used on an interim basis while the staff proceeds to engage stakeholders in the development of final guidance.

The staff will develop final guidelines that articulate what constitutes a special circumstance in a clear and objective manner and modifications to relevant guidance documents to incorporate this guidance. In particular, the staff will modify the regulatory guidance found in Regulatory Guide (RG) 1.174 to describe the concept of special circumstances and the staff's role in reviewing the risk implications of license amendment requests that are not risk informed. The staff will also evaluate whether any regulatory guides or standard review plans in deterministic review areas need to be modified to sensitize the technical staff to identifying potential risk implications of licensing changes within their deterministic review scope. The staff will ensure that both internal and external stakeholders are meaningfully engaged in the development of the final guidelines and related guidance documents.

The staff will subsequently reflect this information in internal, office-level documents that establish the process for reviewing license amendment requests, such as Office of Nuclear Reactor Regulation Office Letter 803, "License Amendment Review Procedures." In modifying the process documents, the staff will be careful to clearly differentiate the concept of adequate protection from the numerical risk acceptance guidelines of RG 1.174.

Backfit Discussion

This RIS requires no action or written response. Consequently, the staff did not perform a backfit analysis.

Federal Register Notification

The staff did not publish a notice of opportunity for public comment in the *Federal Register* because the RIS is informational and pertains to a staff position that does not represent a departure from current regulatory requirements and practice. NRC intends to work with the Nuclear Energy Institute, industry representatives, members of the public, and other stakeholders in developing final guidance and modifying related guidance documents.

If there are any questions about this matter, please contact the person listed below.

/RA by Ledyard Marsh Acting For/
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Attachments:
1. Interim Guidelines for Using Risk Information in Regulatory Decisionmaking

Interim Guidelines for Using Risk Information in Regulatory Decisionmaking

The process depicted in Figure 1 will be used in the staff review of both licensee-initiated risk-informed license amendment requests, as well as license amendment requests in which the licensee chooses to not submit risk information.

The staff will assess the requested changes and the need for and effectiveness of any compensatory measures that might be warranted because of risk considerations by evaluating the changes relative to the safety principles and integrated decisionmaking process defined in Regulatory Guide (RG) 1.174. The risk acceptance guidelines (Sections 2.2.4 and 2.2.5 of RG 1.174) describe acceptable levels of risk increase as a function of total core damage frequency (CDF) and large early release frequency and the manner in which the acceptance guidelines should be applied in the review and decisionmaking process. The guidelines serve as a point of reference for gauging risk impact but are not legally binding requirements.

For non-risk-informed license amendment requests, the preliminary assessment would be qualitative with a decision based on engineering judgment since quantitative risk information would not generally be presented in submittals that are not risk informed. If “special circumstances” are believed to exist, the staff will explore in more detail the underlying engineering issues contributing to the risk concern, and the potential risk significance of the license amendment request. These “special circumstances” represent conditions or situations that would raise questions about whether there is adequate protection and that could rebut the normal presumption of adequate protection from compliance with existing requirements. The application and related issues would be given increased attention from the U.S. Nuclear Regulatory Commission management at this point.

With management concurrence, the staff will question risk further if there is a reason to believe that the proposed change would compromise the safety principles described in RG 1.174 and would substantially increase risk relative to the risk acceptance guidelines contained in the regulatory guide. In such instances, the staff will ask the licensee to address the safety principles and the numerical guidelines for acceptable risk increases contained in RG 1.174 in the submittal. The staff may ask the licensee to submit the information it needs to make an appropriate risk assessment. If an applicant does not choose to address risk, the NRC staff will not issue the requested amendment until it has assessed the risk implications sufficiently to determine that there is reasonable assurance that the public health and safety will be adequately protected if the amendment request is approved. A licensee’s decision not to submit requested information could impede the staff’s review and could also prevent the staff from reaching a finding that there is reasonable assurance of adequate protection. A licensee’s failure to submit requested information could also be a basis for rejection pursuant to 10 CFR 2.108.

The staff will inform the Commission if it determines that a license amendment application meets the “special circumstances” standard, the basis for that determination, the licensee’s response to the staff’s determination, any delay in the license amendment review process, and any generic implications.

Situations that exceed RG 1.174 guidance could constitute a trigger point at which questions are raised as to whether the proposed change provides reasonable assurance of adequate protection. A more in-depth assessment of the special circumstances, the safety principles, and the issues identified for management attention in Section 2.2.6 of RG 1.174 would then be made in order to reach a conclusion regarding the level of safety associated with the requested change. The final acceptability of the proposed change would be based on a consideration of current regulatory requirements, as well as on adherence to the safety principles, and not solely on the basis of a comparison of quantitative probabilistic risk assessment results with numerical acceptance guidelines. The authority provided by the Atomic Energy Act and current regulations requires rejection of a license amendment request if the NRC finds that adequate protection is not provided.

Figure 1 - Process and Logic for Considering Risk in License Amendment Reviews