UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
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NRC REGULATORY ISSUE SUMMARY 2005-29 ANTICIPATED TRANSIENTS THAT COULD DEVELOP INTO MORE SERIOUS EVENTS

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 notify licensees of a concern identified during recent reviews of power uprate license amendment requests. The licensing bases of certain licensees, as described in the plant updated final safety analysis reports (UFSARs), fail to demonstrate that anticipated transients (i.e., Condition II events) will not progress to more serious events (Condition III or IV events). Specifically, certain licensees have not shown that their plant designs and emergency operating procedures effectively preclude the initiation of any Condition III or IV events, resulting from relatively frequent Condition II events (e.g., the inadvertent emergency core cooling system (ECCS) actuation event). Proper consideration of this concern will facilitate NRC staff review of future amendment requests, such as power uprates and fuel transitions.

This RIS requires no action or written response on the part of an addressee.

BACKGROUND INFORMATION

Many licensees have incorporated ANS 51.1, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor [PWR] Plants 1983 (replaces ANSI N18.2-1973), or ANSI/ANS-52.1-1978, Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants, 1978 in their UFSARs, as appropriate. One of the ANS Design Requirements, contained in the section that defines Condition II, III and IV events, is that, "by itself, a Condition II incident cannot generate a more serious incident of the Condition III or IV type without other incidents occurring independently." This requirement, hereinafter referred to as the nonescalation criterion, is intended to limit the probability of initiating the more safety significant Condition III or IV events at the relatively high frequency of Condition II events. The nonescalation criterion is cited in the Standard Review Plan[1] sections on Condition II events.

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1 Standard Review Plan, NUREG-0800 (formerly NUREG-75/087), Draft Rev 3, April, 1996

While reviewing licensing basis analyses of the inadvertent ECCS actuation event, the staff identified a concern about whether PWRs comply with the nonescalation criterion when they are equipped with ECCSs capable of pressurizing the reactor coolant system (RCS) to levels greater than the opening setpoint pressures of any of their pressurizer relief or safety valves. Such ECCSs typically employ charging pumps in a safety injection mode. In these plants, the inadvertent ECCS actuation event, a Condition II event, can become a small-break loss-of-coolant accident (SBLOCA), a Condition III event, if the ECCS flow fills the pressurizer and a pressurizer relief or safety valve opens, discharges water, and then fails to close.

In 1993, Westinghouse noted that, for the inadvertent ECCS actuation event, many Westinghouse plants (more than two dozen in the US and about the same number in other countries), had licensing bases that did not address the possibility of this event developing into a SBLOCA. To remedy this situation, Westinghouse issued a nuclear safety advisory letter (NSAL),[2] advising its affected customers of the issue and offering them three alternative means of addressing it. For one of these alternatives, the NSAL stated that operation of the pressurizer power-operated relief valves (PORVs) may be credited to mitigate the inadvertent ECCS actuation event, but failed to note that such PORVs must be qualified as safety-related systems. In most plants, the PORVs are not qualified as safety-related systems. Although GI-70 discusses the specific circumstances under which non-safety-related PORVs may be credited in accident analyses, crediting the operation of non-safety-related systems to perform protection functions in response to design basis events is generally not accepted.

Since 1993, when this NSAL was issued, most of the affected Westinghouse plants have modified their licensing basis analyses, usually via the Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.59 process, to credit the operation of either the PORVs or the pressurizer safety valves without first qualifying them as safety-related systems capable of relieving water. The licensing bases of the remaining Westinghouse plants and certain, affected non-Westinghouse PWRs still do not address the nonescalation criterion.

There have been several inadvertent ECCS operation incidents that have resulted in filling the pressurizer and relieving water through the PORVs. The most recent event occurred on the morning of April 17, 2005, at Millstone, Unit 3. In this case, the PORVs relieved water and then failed to reseat completely. Millstone's licensing basis analyses assume that the PORVs are used as a protection system to prevent the relief of water through the pressurizer safety relief valves by limiting the RCS pressure to levels below the PORVs' opening setpoint. This is permitted since the Millstone, Unit 3 PORVs were qualified as safety-related components and accepted for this purpose in a 1998 license amendment. The technical specifications were revised to require the PORV block valves to be operable, and the PORVs to be available.

Justifying and crediting timely operator actions to mitigate the inadvertent ECCS actuation would be another way to address this issue. Some licensees have shown there is sufficient time for an operator to shut off the ECCS flow before the pressurizer is filled, demonstrating that any pressurizer PORVs or safety valves that open during the event would relieve only

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2 NSAL-93-013, G.G. Ament and K.J. Vavrek, Westinghouse ESBU, June 30, 1993, and NSAL-93-013, Supplement 1, J.S. Galembush, Westinghouse ESBU, October 28, 1994 (ML052930330)

steam. Licensees using this approach verify that assumed operator action times are reasonable and consistent with plant procedures and operator capabilities.

SUMMARY OF ISSUE

Consideration of a variety of scenarios is necessary in order to identify a Condition II event that could logically progress to a Condition III event without the influence of any independent equipment failures or operator errors. Typically, such scenarios involve filling the pressurizer (in PWRs) and subsequently discharging water through relief or safety valves that are not qualified for water relief. These valves are then assumed to fail in the open position and create a SBLOCA.

Condition II events could fill the pressurizer by heating and swelling the reactor coolant (heatup events) or by adding water to the reactor coolant inventory (water inventory addition events). In either case, reactor coolant could surge into the pressurizer, fill the pressurizer, and exit the RCS via the PORVs or pressurizer safety valves. Heatup events, such as loss of feedwater, loss of load, or rod withdrawal at power, usually cause the pressurizer level to rise as the reactor coolant expands until the reactor is automatically tripped on a pressurizer high-level signal. In water inventory addition events, such as inadvertent ECCS actuation or charging system malfunction at power, the reactor may also be tripped, but there is no automatic termination of ECCS and charging flows. The pressurizer level will continue to rise until the pressurizer becomes water-solid or the operator terminates the water inventory addition.

The purpose of this RIS is to ensure that licensees are aware of the issue and have the opportunity to resolve it before it arises during licensing actions such as power uprates.

The NRC staff is concerned that some licensees may be crediting PORVs without qualification for water relief and without establishing additional restrictions to ensure the availability of the PORVs and block valves. When the licensing bases for these plants include the nonescalation criterion defined in ANS 51.1, the NRC staff will apply the guidance from the applicable standard review plans during reviews in which the accident analysis is revised (e.g., power uprates) and may have questions about how this issue has been addressed.

Westinghouse's NSAL, however, does not advise addresses to qualify PORVs and their automatic control system circuitry. It states that PORVs may be allowed to relieve water, since the block valves may be used to isolate any PORVs that might fail to close. This reasoning is inconsistent with the nonescalation criterion, since such action must, necessarily, occur after a stuck-open PORV has transformed the inadvertent ECCS actuation event into a SBLOCA, a Condition III event (i.e., after the nonescalation criterion has been violated). In other words, isolating a PORV is an operation that is undertaken to mitigate a Condition III SBLOCA, not a Condition II inadvertent ECCS actuation.

The NRC staff's position is noted in the power uprate review standard, as follows: "For the inadvertent operation of emergency core cooling system and chemical and volume control system malfunctions that increase reactor coolant inventory events: (a) non-safety-grade pressure-operated relief valves should not be credited for event mitigation and (b) pressurizer level should not be allowed to reach a pressurizer water-solid condition."[3]

The NRC staff is also concerned that some licensees may have changed their licensing bases to credit the use of non-safety-related components and systems to mitigate design basis events without prior staff review. The NRC staff will review licensees' analyses and evaluations of Condition II events, such as the inadvertent ECCS actuation event, in order to ascertain whether the ANS nonescalation criterion is adequately addressed as part of technical reviews of applicable license amendment requests. The NRC staff reviews will also consider how licensees revised their licensing bases (e.g., via the 10 CFR 50.59 process), and, if applicable, how licensees have complied with 10 CFR Part 21. Licensees may wish to prepare for these reviews by examining their licensing basis records with respect to this issue.

BACKFIT DISCUSSION

This RIS requires no action or written response and, therefore, is not a backfit under 10 CFR 50.109. Consequently, the staff did not perform a backfit analysis.

FEDERAL REGISTER NOTIFICATION

A notice of opportunity for public comment on this RIS was not published in the *Federal Register* because this RIS is informational and pertains to a NRC staff position that does not depart from current regulatory requirements and practice.

SMALL BUSINESS REGULATORY ENFORCEMENT FAIRNESS ACT OF 1996

The NRC has determined that this action is not subject to the Small Business Regulatory Enforcement Fairness Act of 1996.

PAPERWORK REDUCTION ACT STATEMENT

This RIS does not contain information collections and, therefore, is not subject to the requirements of the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.).

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[3] RS-001, Revision 0, "Review Standard for Extended Power Uprates," December 2003, (Note 8 of Matrix 8 of Section 2.1)

CONTACT

Please direct any questions about this matter to the technical contact, or to the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

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Note: NRC generic communications may be found on the NRC public website, http://www.nrc.gov, under Electronic Reading Room/Document Collections.