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U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

OFFICE OF NUCLEAR REACTOR REGULATION

15.1.5 STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT (PWR)

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Evaluation Branch

I. AREAS OF REVIEW

The steam release resulting from a rupture of a main steam pipe will cause an increase in steam flow which decreases with time as the steam pressure decreases. The increased steam flow causes increased energy removal from the reactor coolant system and results in a reduction of coolant temperature and pressure. Due to the negative moderator temperature coefficient this cooldown causes an increase in core reactivity. The core reactivity increase may cause a power level increase and a decrease in shutdown margin. If the plant is at power, the reactor is automatically tripped and the main steam and feedwater line isolation valves are automatically closed. Decay heat is removed as necessary through the unaffected steam generators by venting steam from the secondary system safety and relief valves. The auxiliary feedwater system supplies makeup water to the unaffected steam generator(s).

Analysis of the transient following a steam line break is sensitive to the fluid discharge rate at the break so that a range of break sizes must be evaluated both inside and outside containment to determine the acceptability of the system response. Past experience generally shows that the worst break is that which results in the maximum cooldown rate. The course the transient takes and its ultimate effects also depend on the assumed initial power level and mode of operation (i.e., hot shutdown, full power, one-, two-, or three-loop operation). Analyses with various assumed initial conditions are required to verify that the condition leading to the severest consequences has been identified.

The topics reviewed include: postulated initial core and reactor conditions pertinent to the steam line break accident, methods of thermal and hydraulic analyses including the effects of hydraulic instabilities, postulated sequence of events including analyses to determine the time of reactor trip and time delays prior to and subsequent to initiation of the reactor protection system, assumed responses of the reactor coolant and auxiliary systems, functional and operational characteristics of the reactor protection system in terms of its effects on the sequence of events, operator actions required to secure and maintain the reactor in a safe shutdown condition, core power excursion due to power demand created by excessive steam flow out the break, and variables influencing neutronics.

The results of the analyses are reviewed to ensure that pertinent system parameters are within expected ranges. The parameters of importance for these transients include reactor coolant system (RCS) pressure, steam generator pressure, fluid temperatures, clad temperatures, discharge flow rates, steam line and feedwater flow rates, safety and relief valve flow rates, pressurizer and steam generator water levels, reactor power, total core reactivity, hot and average channel heat flux, and minimum departure from nucleate boiling ratio (DNBR).

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both RSB and ICSB. The RSB reviewer concentrates on the capability of the reactor protection system, the engineered safety systems, and operator action to secure and maintain the reactor in a safe condition.

In addition, the RSB will coordinate other branches' evaluations that interface with the overall review of the steam system piping failures as follows: the Auxiliary Systems Branch (ASB) reviews the auxiliary feedwater system to verify its ability to function following a steam line break given a single active component failure with either onsite or offsite power as part of its primary review responsibility for SRP Section 10.4.9. RSB reviews the auxiliary feedwater system to verify that the flow provided is acceptable for controlling the transient following a steam line break. The Mechanical Engineering Branch (MEB) reviews the effects of blowdown loads, including jet propulsion piping and component supports as part of its primary review responsibility for SRP Sections 3.6.2 and 3.9. In addition, MEB reviews the design bases for safety and relief valves in SRP Section 3.9.3. The Containment Systems Branch (CSB) evaluates the response of the containment to ruptures of steam lines with regard to the effects of pressure and temperature on the containment functional capabilities as part of its primary review responsibility for SRP Section 6.2.1. CSB also reviews the analytical methods for deriving NSSS mass energy releases exiting a postulated break. The Core Performance Branch (CPB) reviews all the nuclear design aspects, this includes power levels, power distributions, Doppler coefficients, moderator temperature coefficients, reactor kinetics parameters, DNB correlations and control rod worths as part of its primary review responsibility for SRP Sections 4.2, 4.3 and 4.4. The ICSB reviewer concentrates on the instrumentation and controls aspects of the sequence described in the SAR to evaluate whether the reactor and plant protection and safeguards controls and instrumentation systems will function as assumed in the safety analysis with regard to automatic actuation, remote sensing, indication, control, and interlocks with auxiliary or shared systems. ICSB also evaluates potential bypass modes and the possibility of manual control by the operator, as part of its primary review responsibility for SRP Sections 7.1 through 7.7.

For those areas of review identified above as being reviewed as part of the primary review responsibility of other branches, the acceptance criteria necessary for the review and their methods of application are contained in the referenced SRP section of the corresponding primary branch.

The analytical methods are reviewed by RSB to ascertain whether the mathematical modeling and computer codes have been previously reviewed and accepted by the staff. If a referenced analytical method has not been previously reviewed, RSB initiates a generic evaluation of the new analytical model.

A secondary review is performed by the Accident Evaluation Branch and the results are used by RSB to complete the overall evaluation of the break analysis. The Accident Evaluation Branch (AEB) evaluates the fission product release and verifies that the radiological consequences resulting from a steam line break are within acceptable limits. This evaluation is performed for the design basis case as described in the appendix to this SRP section. The results of AEB's analysis is transmitted to RSB for use in the SER writeup.

II. ACCEPTANCE CRITERIA

The general objective of the review of steam line rupture events is to verify that short-term and long-term coolability has been achieved by confirming that the primary reactor coolant system is maintained in a safe status for a break equivalent in area to the double-ended rupture of the largest steam line. RSB acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criteria 27 and 28, as they relate to the reactor coolant system being designed with appropriate margin to assure that acceptable fuel design limits are not exceeded, and that the capability to cool the core is maintained.

B. General Design Criterion 31, as it relates to the reactor coolant system being designed with sufficient margin to assure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

C. General Design Critierion 35, as it relates to the reactor cooling system and associated auxiliaries being designed to provide abundant emergency core cooling.

In addition, task action plan items necessary to meet the requirements to maintain adequate decay heat removal and reactor coolant pump integrity and operation are Items II.E.1.2, II.K.2.1, II.K.2.8, II.K.3.5, II.K.2.16, II.K.3.25, and II.K.3.40 of NUREGs 0694, 0718, and 0737. Specific criteria necessary to meet the relevant requirements of the above regulations are as follows:

1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

2. The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see SRP Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently

limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. The radiological criteria used in the evaluation of steam system pipe break accidents (PWRs only) appear in the appendix to this SRP section.

4. The integrity of the reactor coolant pumps should be maintained, such that loss of a-c power and containment isolation will not result in pump seal damage.

5. The auxiliary feedwater system must be safety grade and, when required, automatically initiated.

6. Tripping of the reactor coolant pumps should be consistent with the resolution to Task Action Plan item II.K.3.5.

There are certain assumptions regarding important parameters used to describe the initial plant conditions and postulated system failures which should be used. These are listed below:

a. The reactor power level and number of operating loops assumed at the initiation of the transient should correspond to the operating condition which maximizes the consequences of the accident. These assumed initial conditions will vary with the particular nuclear steam supply system (NSSS) design, and sensitivity studies will be required to determine the most conservative combination of power level and plant operating mode. These sensitivity studies may be presented in a generic report and referenced in the SAR.

b. Assumptions as to the loss of offsite power and the time of loss should be made to study their effects on the consequences of the accident. A loss of offsite power may occur simultaneously with the pipe break, or during the accident, or offsite power may not be lost. Analyses should be made to determine the most conservative assumption appropriate to the particular plant design. The analyses should take account of the effect that loss of offsite power has on reactor coolant pump and main feedwater pump trips and on the initiation of auxiliary feedwater flow, and the effects on the sequence of events for these accidents.

c. The effects (pipe whip, jet impingement, reaction forces, temperature, humidity, etc.) of postulated steam line breaks on other systems should be considered in a manner consistent with the intent of Branch Technical Positions ASB 3-1 and MEB 3-1 (Ref. 1).

d. The worst single active component failure should be assumed to occur. The assumed single failure may cause more than one steam generator to blow down, or may be in any of the systems required to control the transient.

e. The maximum-worth rod should be assumed to be held in the fully withdrawn position. An appropriate rod reactivity worth versus rod position curve should be used.

f. The core burnup (time in core life) should be selected to yield the most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

g. The initial core flow assumed for the analysis of the steam line rupture accident should be chosen conservatively. If the minimum core flow allowed by the technical specifications is assumed, the minimum DNBR margin results, but for the analysis of steam line break accidents this may not be the most conservative assumption. For example, maximum initial core flow results in increased reactor coolant system cooldown and depressurization, decreased shutdown margin, and an increased possibility that the core will become critical and return to power. Since it is not clear what initial core flow is most conservative, the assumed value should be justified.

h. For postulated pipe failure in nonseismically qualified portions of the main steam line (outside containment and downstream of the MSIVs) due to a seismically initiated event, only safety grade equipment should be assumed operative to mitigate the consequences of the break.

i. For postulated instantaneous pipe failures in seismically qualified portions of the main steam line (inside containment and upstream of the MSIVs), only safety grade equipment should be assumed operative. If, in addition, a single malfunction or failure of an active component is postulated, credit may be taken for the use of a backup nonsafety grade component to mitigate the consequences of the break.

j. During the initial 10 minutes of the transient, should credit for operator action be required (i.e., RCP trip), an assessment for the limiting consequence must be performed in order to account for operator delay and/or error.

III. REVIEW PROCEDURES

The procedures below are used during both the construction permit (CP) and operating license (OL) reviews. During the CP review, the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change. At the OL review stage, final values should be used in the analysis, and the reviewer should compare these to the limiting safety system settings included in the proposed technical specifications.

1. The reviewer determines the acceptability of the analytical models and assumptions as follows:

a. The values of system parameters and initial core and system conditions used as input to any analytical model are reviewed by RSB. Of particular importance are the reactivity coefficients and control rod worths used in the analysis, and the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The justification provided by the applicant to show that he has selected the core burnup that yields the minimum margins is evaluated. CPB is consulted regarding core-related parameters such as DNB correlations and the values of the reactivity parameters used in the analysis. The reviewer confirms that the amount of secondary coolant expelled from the system (for breaks outside containment) has been calculated conservatively by evaluating the applicant's methods and assumptions, by comparing with an acceptable analysis performed on another plant of similar design, or by comparing with staff calculations for typical plants done by RSB.

b. The acceptability of the methods equations, sensitivity studies, and models proposed by the applicant are evaluated.

c. Analytical models should be sufficiently detailed to simulate the reactor coolant (primary), steam generator (secondary), and auxiliary systems. The reviewer evaluates the following functional requirements:

(1) Reactor trip signal: credit taken for any reactor trip signal is reviewed by ICSB to confirm that, under accident conditions, the instrumentation and control systems are capable of the assumed response.

(2) Emergency core cooling system (ECCS): credit taken for actuation of the ECCS is reviewed by ICSB to verify the ability of the instrumentation and control systems to respond as assumed.

(3) Auxiliary feedwater system: the availability of the auxiliary feedwater system to supply adequate auxiliary feedwater flow to the intact steam generators during the accident and the subsequent shutdown condition is evaluated. This is done by ASB as to availability of the system and by RSB as to capability to effect an orderly shutdown. Since auxiliary feedwater system designs are diverse and may require both automatic and manual actuation, preoperational tests should be specified to identify any necessary operator actions and to establish times required for their completion.

d. The variations with time during the transient of the neutron power, heat fluxes (average and maximum), total core reactivity, reactor coolant system pressure, minimum DNBR; coolant conditions (inlet temperature, core average temperature and average exit and hot channel exit temperatures, fuel rod conditions (maximum fuel centerline temperature, maximum clad temperature, or maximum fuel enthalpy), steam generator pressure, containment pressure, relief and/or safety valve flow rates, discharge flow rate, steam line and feedwater flow rates and pressurizer and steam generator water levels are reviewed. The values of the more important of these parameters for the steam line break accident (as listed in subsection I) are compared with those predicted for other similar plants to see that they are within the range expected.

2. To the extent deemed necessary, the reviewer evaluates the effect of single active failures of systems and components that may affect the course of the accident. This phase of the review is done using the system review procedures described in the SRP sections for Chapters 5, 6, 7, 8, and 10 of the SAR. The reviewer also considers single failures that may cause more than one steam generator to blow down, thus increasing the reactivity addition to the core.

3. The reviewer confirms that a commitment has been made in the SAR to conduct preoperational tests for verifying that valve discharge rates and response times (including, for example, opening and closing times for main feedwater, auxiliary feedwater, turbine and main steam isolation valves, and steam generator and pressurizer relief and safety valves) have been conservatively modeled in the accident analyses. In addition, preoperational testing should include verification of reactor trip delay times, startup delay times for auxiliary feedwater system actuation, safety

injection signal delay time, and delay times for delivery of any high concentration boron solution required to bring the plant to a safe shutdown condition.

4. Based on the above information, AEB evaluates the radiological consequences of the design basis steam line break accident as described in the appendix to this SRP section.

5. Upon request from the primary reviewer, other secondary review branches will provide input for the areas of review stated in subsection I of this SRP section. The primary reviewer obtains and uses such input as required to assure that this review procedure is complete.

6. The reliability and operability of the auxiliary feedwater systems (AFWS) are reviewed to assure conformance to the following TMI Action Plan items (Ref. 6 through 8) as they relate to auxiliary feedwater system performance requirements for steam system piping failures:

   (a) Items II.E.1. and II.K.2.1,
   (b) Items II.E.1.2 and II.K.2.8.

7. The influence of reactor coolant pump trip during ECCS initiation is reviewed to assure conformance to the TMI Action Plan item II.K.3.5 (Ref. 6 through 8). Should tripping of the reactor coolant pumps require manual action, delays in operation actions must be assessed.

8. The reliability and integrity of the reactor coolant pump seals during loss of alternating-current power and loss of coolant to the seals (i.e., resulting from containment isolation) are reviewed to assure conformance to the TMI Action Plan items II.K.2.16, II.K.3.25, and II.K.3.40.

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and his review supports the following kinds of statements and conclusions, which should be included in the staff's safety evaluation report:

> The staff concludes that the consequences of postulated steam line breaks meet the relevant requirements set forth in the General Design Criteria 27, 28, 31, and 35 regarding control rod insertability and core coolability and TMI Action Plan Items. This conclusion is based upon the following:
>
> (a) The applicant has met the requirements of GDC 27 and 28 by demonstrating that the resultant fuel damage was limited such that control rod insertability would be maintained, and that no loss of core cooling capability resulted. The minimum departure from nucleate boiling ratio (DNBR) experienced by any fuel rod was ________, resulting in __% of the rods experiencing cladding performation.
>
> (b) The applicant has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.
>
> (c) The applicant has met the requirements of GDC 35 with respect to demonstrating the adequacy of the emergency cooling systems to

provide abundant core cooling and reactivity control (via boron injection).

(d) The analyses and effects of steam line break accidents inside and outside containment, during various modes of operation with and without offsite power, have been reviewed and were evaluated using a mathematical model that has been previously reviewed and found acceptable by the staff.

(e) The parameters used as input to this model were reviewed and found to be suitably conservative.

(f) The radioactivity release has been evaluated using the computer code SARA and a conservative description of the plant response to the accident. A decontamination factor of ______ between the water and steam phases and a X/Q value of ______ sec/m^3 has been used in our evaluation of radiological consequences. The calculated doses are presented in Table _____. Technical specification limits on primary and secondary coolant activities will limit potential doses to a small fraction of the 10 CFR Part 100 exposure guidelines. The potential doses are within 10 CFR Part 100 exposure guidelines even if the accident should occur coincident with an iodine spike.

(g) The applicant has met the requirements of Task Action Plan items II.E.1, II.K.2.1, II.E.1.2, and II.K.2.8 with respect to demonstrating the adequacy of the auxiliary feedwater design to remove decay heat following steam system piping failures.

(h) The applicant has met the requirements of Task Action Plan items II.K.2.16, II.K.3.25 and II.K.3.40 with respect to demonstrating the integrity and operation of the reactor coolant pumps to withstand the postulated accident.

(i) The applicant has met the requirements of Task Action Plan Item II.K.3.5 with respect to the operation and tripping of the reactor coolant pumps. The assumptions used are conservative and consistent with the generic resolution to item II.K.3.5.

V. IMPLEMENTATION

The following is intended to provide guidance to applicants and licensees regarding the NRC staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced NUREGs.

VI. REFERENCES

1. Branch Technical Positions ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," attached to SRP

Section 3.6.1, and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment," attached to SRP Section 3.6.2.

2. 10 CFR Part 50, General Design Criterion 27, "Combined Reactivity Control Systems Capability."

3. 10 CFR Part 50, General Design Criterion 28, "Reactivity Limits."

4. 10 CFR Part 50, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

5. 10 CFR Part 50, General Design Criterion 35, "Emergency Core Cooling."

6. NUREG-0694, "TMI-Related Requirements for New Operating Licenses."

7. NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses."

8. NUREG-0737, "Classification of TMI Action Plan Requirements."