NUREG-0800
(Formerly NUREG-75/087)

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~~SRXB)[1]

Secondary - ~~Accident Evaluation Branch~~Emergency Preparedness and Radiation Protection Branch (PERB)[2]

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 (AFWS)[3] 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.~~e.g.,[4] 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.[5]

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).[6]

The sequence of events described in the applicant's safety analysis report (SAR) is reviewed by both ~~RSB~~SRXB[7] and ~~ICSB~~ the Instrumentation and Control Branch (HICB).[8] The ~~RSB~~SRXB 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.

The analytical methods are reviewed by ~~RSB~~SRXB 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~~SRXB initiates a generic evaluation of the new analytical model.[9]

~~The Core Performance Branch (CPB)~~ SRXB reviews all the nuclear design aspects; this includes power levels, power distributions, Doppler coefficients, moderator temperature coefficients, reactor kinetics parameters, departure from nucleate boiling (DNB)[10] correlations, and control rod worths as part of its primary review responsibility for Standard Review Plan (SRP)[11] Sections 4.2, 4.3, and 4.4.[12]

Review Interfaces[13]

In addition, the ~~RSB~~SRXB will coordinate other branches' evaluations that interface with the overall review of the steam system piping failures, as follows:

1. ~~the Auxiliary Systems Branch (ASB)~~ The Plant Systems Branch (SPLB)[14] 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. The SPLB review includes an evaluation of the AFWS to include (a) a simplified reliability analysis, (b) a design basis review, and (c) flow design basis and criteria in accordance with 10 CFR 50.34(f)(1)(ii) (applicable to pressurized water reactors, PWRs, only).[15] The SPLB also reviews the integrity of the reactor coolant pump seals as part of its primary review responsibility for SRP Section 9.2.2.[16]

2. ~~RSB~~SRXB reviews the auxiliary feedwater system to verify that the flow provided is acceptable for controlling the transient following a steam line break.

3. The Mechanical Engineering Branch (~~MEB~~EMEB)[17] 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 3.9.1 through 3.9.3.[18] In addition, ~~MEB~~ EMEB reviews the design bases for safety and relief valves in SRP Section 3.9.3.

4. The Materials and Chemical Engineering Branch (EMCB) reviews the fracture toughness properties of the reactor coolant pressure boundary and reactor vessel as part of its primary review responsibility for SRP Sections 5.2.3 and 5.3.1.[19]

5. The Containment Systems and Severe Accident Branch (~~CSB~~SCSB)[20] 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~~ SCSB also reviews the analytical methods for deriving nuclear steam supply system (NSSS)[21] 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.~~[22]

6. The ~~ICSB~~ HICB 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. The HICB review includes an evaluation of the instruments and controls required to ensure automatic and manual axillary feedwater system initiation and flow indication in the control room in accordance with 10 CFR 50.34(f)(2)(xii) as part of its primary review responsibility for SRP Section 7.1.[23] ~~ICSB~~ HICB 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 sections ~~of the corresponding primary branch~~.[24]

~~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.~~[25]

A secondary review is performed by the ~~Accident Evaluation Branch~~ PERB,[26] and the results are used by ~~RSB~~SRXB to complete the overall evaluation of the break analysis. The ~~Accident Evaluation Branch (AEB)~~PERB 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~~PERB's analysis ~~is~~ are transmitted to ~~RSB~~SRXB for use in the safety evaluation report (SER) writeup.[27]

II. ACCEPTANCE CRITERIA

The general objective of the review of steam line rupture events is to verify that short-term and long-term ~~cool~~ability to cool the core[28] 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~~SRXB acceptance criteria are based on meeting the relevant requirements of the following regulations:

A. General Design Criterion 17 (GDC 17), as it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during an anticipated operational occurrence and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.[29]

~~A~~B. General Design Criteria 27 (GDC 27) and 28 (GDC 28),[30] as they relate to the reactor coolant system being designed with appropriate margin to ~~assure~~ensure[31] that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.

~~B~~C. General Design Criterion 31 (GDC 31),[32] as it relates to the reactor coolant system being designed with sufficient margin to ~~assure~~ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

~~C~~D. General Design Criterion 35 (GDC 35),[33] 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.~~ Requirements for ensuring adequate decay heat removal and reactor coolant pump integrity and operation are specified in 10 CFR 50.34(f)(2)(xii) and 10 CFR 50.34(f)(1)(iii), respectively. In addition, Task Action Plan item II.K.3.5 of NUREG-0737 addresses automatic trip of reactor coolant pumps during a loss-of-coolant accident (LOCA). This issue was resolved in Generic Letters 83-10A through 83-10F, 85-12, 86-05, and 86-06.[34] 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 ac 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~~)~~[35] 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. For new applications, loss of offsite power should be considered in addition to any limiting single active failure. (This position is based upon interpretation of GDC 17, as documented in the Final Safety Analysis Report for the ABB-CE System 80+ design certification.)[36]

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. For new applications, loss of offsite power should not be considered as a single failure, (see assumption b above).[37] 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; however, 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 main steam isolation valves,[38] 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.~~e.g., reactor coolant pump ~~RCP~~[39] trip), an assessment for the limiting consequence must be performed in order to account for operator delay and/or error.

Technical Rationale[40]

The technical rationale for application of these acceptance criteria to reviewing analyses of transients initiated by steam system piping failures is discussed in the following paragraphs:[41]

1. Compliance with GDC 17 requires (in part) that onsite and offsite electrical power systems be provided to ensure the functioning of structures, systems, and components important to safety. The safety function for each power system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that (a) the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (b) the core is cooled and containment integrity and other vital functions are maintained.

   GDC 17 is applicable to this section because it requires that the loss of offsite power be considered — not as a single failure event, but assumed in the analyses for each event without changing the event category. Thus, the applicant is required to consider a loss of offsite power concurrent with a single failure in the analysis of steam system piping failures.

   Meeting the requirements of GDC 17 provides assurance that the specified acceptable fuel design limits and design conditions of the design reactor coolant pressure boundary are not exceeded as a result of steam system piping failures concurrent with loss of offsite power, that the core is cooled, and that containment and other vital functions are maintained.[42]

2. Compliance with GDC 27 requires that reactivity control systems be designed to have a combined capability (in conjunction with poison added by the emergency core cooling system) of reliably controlling reactivity changes, thereby ensuring that the capability for core cooling is maintained under postulated accident conditions and with appropriate margin for stuck rods.

   Compliance with GDC 28 requires that reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (a) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (b) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor temperature and pressure, and addition of cold water.

GDC 27 and GDC 28 are applicable to this section because the reviewer evaluates steam system piping failures, both inside and outside containment, that could cause transient conditions affecting reactor coolant temperature and pressure, including complex changes in core reactivity. The applicant's analyses of these transients in the SAR must demonstrate that reactivity, pressure, and temperature changes will not be severe enough to cause an unacceptable impact on the reactor coolant pressure boundary or on the capability for cooling the core. These analyses must be independently reviewed by the staff in accordance with this SRP section.

Meeting the requirements of GDC 27 and 28 provides assurance that a transient initiated by a steam system pipe break will not result in (a) an unacceptable stress on the reactor coolant pressure boundary, (b) acceptable fuel design limits being exceeded, or (c) loss of capability of the core cooling systems or the reactivity control systems to perform their design safety functions.[43]

3. Compliance with GDC 31 requires that, under the stress of operation, maintenance, testing, and postulated accidents, the reactor pressure boundary shall be designed with sufficient margin to ensure that (a) the boundary behaves in a nonbrittle manner and (b) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other boundary material variables under a full range of conditions. The design will also address such issues as the uncertainties of determining material properties; the effects of irradiation on material properties; residual, steady state, and transient stresses; and the sizes of flaws.

   GDC 31 is applicable to this section because the reviewer evaluates steam system piping failures, both inside and outside containment, that could cause transient conditions with a potentially harmful effect on the reactor coolant pressure boundary. A steam system piping break can result in a rapid decrease in reactor coolant temperature and steam generator pressure, placing undue stress on the reactor coolant pressure boundary. This potential problem could be aggravated by a pressurization of the primary system when the emergency core cooling system is activated. The amount of stress to the reactor coolant pressure boundary depends on the severity of the transient. The severity of the transient is assessed by the applicant in the SAR and is reviewed by the staff in accordance with this SRP section.

   Meeting the requirements of GDC 31 provides assurance that a transient initiated by a steam system pipe break will not result in an unacceptable stress on the reactor coolant pressure boundary.[44]

4. Compliance with GDC 35 requires a system that will provide abundant emergency core cooling. The system safety function shall be to transfer heat from the reactor core after any loss of reactor coolant at a rate ensuring that (a) fuel and clad damage interfering

with continued effective core cooling is prevented and (b) clad metal-water reaction is limited to negligible amounts.

GDC 35 is applicable to this section because the reviewer evaluates steam system piping failures, both inside and outside containment, that could cause transient conditions with the potential to challenge the emergency core cooling system. During a steam system piping break, excessive steam loss will result in a rapid reduction of reactor coolant temperature and steam generator pressure. A subsequent reactor trip can further reduce the primary system pressure, producing a void within the pressure vessel and creating the need for emergency core cooling. As noted above, the severity of this transient is assessed by the applicant in the SAR and is reviewed by the staff in accordance with this SRP section.

Meeting the requirements of GDC 35 provides assurance that a system will be available for providing abundant core cooling, thereby ensuring that the core will be effectively cooled and that any clad metal-water reaction will be limited to negligible amounts in the event of a transient initiated by a steam system pipe break.[45]

III. REVIEW PROCEDURES

The procedures below are used during ~~both~~ the construction permit (CP), ~~and~~ operating license (OL), and combined license (COL)[46] 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 or COL[47] 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~~SRXB. Of particular importance are (1) the reactivity coefficients and control rod worths used in the analysis and (2) the variation of moderator temperature, void, and Doppler coefficients of reactivity with core life. The reviewer will evaluate the justification provided by the applicant to show that ~~he has selected~~ the core burnup ~~that~~ yieldings the minimum margins has been selected. ~~is evaluated.~~[48] ~~CPB is consulted regarding~~SRXB also reviews[49] 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 methods and assumptions, by comparing these results with those of an acceptable analysis performed on another plant of similar design, or by comparing them with staff calculations for typical plants done by ~~RSB~~SRXB.[50]

   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~~ HICB 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~~ HICB 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~~ SPLB[51] as to availability of the system and by ~~RSB~~SRXB 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 center-line 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.~~ Time-related variations of the following parameters are reviewed:

- reactor 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 center–line 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.[52]

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. For new applications, loss of offsite power should not be treated as a single active failure, as discussed under subsection II, assumptions b and d.[53] 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~~PERB 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~~ensure that this review procedure is complete.

6. The reliability and operability of the auxiliary feedwater systems ~~(AFWS)~~[54] are reviewed to ~~assure~~ensure ~~conformance to the following TMI Action Plan items (Ref. 6 through 8)~~ compliance with the requirements of 10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii) 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.~~[55]

7. The influence of reactor coolant pump trip during ECCS initiation is reviewed to ~~assure conformance to the~~ ensure compliance with the requirements of TMI Action Plan item II.K.3.5 and the resolution thereto contained in References 9 through 17. ~~(Ref. 6 through 8)~~.[56] 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.~~e.g.,[57] resulting from containment isolation) are reviewed to ~~assure~~ensure ~~conformance to the TMI Action Plan items II.K.2.16, II.K.3.25, and II.K.3.40.~~compliance with the requirements of 10 CFR 50.34(f)(1)(iii).[58]

For standard design certification reviews under 10 CFR Part 52, the procedures above should be followed, as modified by the procedures in SRP Section 14.3 (proposed), to verify that the design set forth in the standard safety analysis report, including inspections, tests, analysis, and acceptance criteria (ITAAC), site interface requirements and combined license action items, meet the acceptance criteria given in subsection II. SRP Section 14.3 (proposed) contains procedures for the review of certified design material (CDM) for the standard design, including the site parameters, interface criteria, and ITAAC.[59]

IV. EVALUATION FINDINGS

The reviewer verifies that the SAR contains sufficient information and ~~his~~ that the[60] 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 17,[61] 27, 28, 31, and 35 regarding ~~control rod insertability and core coolability~~ (1) the ability to insert the control rods and to cool the core and (2) TMI Action Plan items.[62] This conclusion is based upon the following:
>
> (a) The applicant has met the requirements of GDC 27 and GDC 28 by demonstrating that the resultant fuel damage was limited such that the ability to insert control rods ~~insertability~~[63] 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 perforation.
>
> (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 (as required by GDC 17),[64] 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/m3$ 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~~ 10 CFR 50.34(f)(1)(ii) and 10 CFR 50.34(f)(2)(xii) with respect to demonstrating the adequacy of the auxiliary feedwater design to remove decay heat following steam system piping failures.[65]

(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~~ 10 CFR 50.34(f)(1)(iii) with respect to demonstrating the integrity and operation of the reactor coolant pumps to withstand the postulated accident.[66]

(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.

For design certification reviews, the findings will also summarize, to the extent that the review is not discussed in other safety evaluation report sections, the staff’s evaluation of inspections, tests, analyses, and acceptance criteria (ITAAC), including design acceptance criteria (DAC), site interface requirements, and combined license action items that are relevant to this SRP section.[67]

V. IMPLEMENTATION

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

This SRP section will be used by the staff when performing safety evaluations of license applications submitted by applicants pursuant to 10 CFR 50 or 10 CFR 52.[68] 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.

The provisions of this SRP section apply to reviews of applications docketed six months or more after the date of issuance of this SRP section.[69]

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced NUREGs~~.~~ except for the position stated in subsection II, assumptions b and d, and in III.2 regarding loss of offsite power and assumed single failures. This new position will be applied to new applications (for a Construction Permit, a manufacturing license, or design certification).[70]

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 17, "Electric Power Systems."[71]

~~2~~3. 10 CFR Part 50, General Design Criterion 27, "Combined Reactivity Control Systems Capability."

~~3~~4. 10 CFR Part 50, General Design Criterion 28, "Reactivity Limits."

~~4~~5. 10 CFR Part 50, General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary."

~~5~~6. 10 CFR Part 50, General Design Criterion 35, "Emergency Core Cooling."

~~6. NUREG-0694, "TMI-Related Requirements for New Operating Licenses."~~[72]

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

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

9. Generic Letter 83-010A - Resolution of TMI Action item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Combustion Engineering (CE) Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

10. Generic Letter 83-010B - Resolution of TMI Action item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Combustion Engineering (CE) Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

11. Generic Letter 83-010C - Resolution of TMI Action item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Westinghouse Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

12. Generic Letter 83-010D - Resolution of TMI Action item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Westinghouse Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

13. Generic Letter 83-010E - Resolution of TMI Action item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Babcock & Wilcox (B&W) Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

14. Generic Letter 83-010F - Resolution of TMI Action item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Licensees with Babcock & Wilcox (B&W) Designed Nuclear Steam Supply Systems (NSSSs), February 8, 1983.

15. Generic Letter 85-012 - Implementation of TMI Action item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Applicants and Licensees with Westinghouse (W) Designed Nuclear Steam Supply Systems (NSSSs), July 28, 1985.

16. Generic Letter 86-005 - Implementation of TMI Action item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Applicants and Licensees with Babcock and Wilcox (B&W) Designed Nuclear Steam Supply Systems (NSSSs), May 29, 1986.

17. Generic Letter 86-005 - Implementation of TMI Action item II.K.3.5, "Automatic Trip of Reactor Coolant Pumps," sent to all Applicants and Licensees with Combustion Engineering (CE) Designed Nuclear Steam Supply Systems (NSSSs), May 29, 1986.[73]

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Attachment A - Proposed Changes in Order of Occurrence

Item numbers in the following table correspond to superscript numbers in the redline/strikeout copy of the draft SRP section.

Item	Source	Description
1.	Current PRB abbreviation	Changed RSB to SRXB (global change for this section).
2.	SRP-UDP format item	Changed Accident Evaluation Branch to Emergency Preparedness and Radiation Protection Branch (PERB).
3.	Editorial	Provided "AFWS" as initialism after first use of "auxiliary feedwater system." AFWS is used elsewhere in the section.
4.	Editorial	Corrected misusage of "i.e."
5.	Editorial	Reorganized paragraph to aid the reader.
6.	Editorial	Reorganized paragraph to aid the reader.
7.	SRP-UDP format item	Changed RSB to SRXB (global change for this section).
8.	SRP-UDP format item	Changed ICSB to Instrumentation and Control Branch (HICB) (global change for this section).
9.	SRP-UDP format item	Moved sentence forward from "Review Interfaces."
10.	SRP-UDP format item	Defined DNB.
11.	SRP-UDP format item	Defined SRP.
12.	SRP-UDP format item	Replaced CPB with SRXB and moved sentence forward from "Review Interfaces" because CPB has been combined with SRXB, eliminating an interface.
13.	SRP-UDP format item	"Review Interfaces" added to AREAS OF REVIEW and organized in numbered paragraph form to describe how SRXB coordinates the review of reactor temperature/pressure transients with under other NRR branches.
14.	SRP-UDP format item	Changed "the Auxiliary Systems Branch (ASB)" to "The Plant Systems Branch (SPLB)."
15.	Integrated Impact No. 1056	Added statement on the primary review responsibility of the PLB for 10 CFR 50.34(f)(1)(ii) under SRP Section 10.4.9.
16.	Editorial	Added a review interface with SRP Section 9.2.2. SRP Section 15.1.5 contains Acceptance Criteria, Review Procedures, and Evaluation Findings with regard to TMI Action Items II.K.2.16 and II.K.3.25. Conformance with these TMI Action Items is reviewed in SRP Section 9.2.2.

Item	Source	Description
17.	SRP-UDP format item	Changed MEB to EMEB (global change for this section).
18.	SRP-UDP format item	Specified SRP Sections 3.9.1 through 3.9.3 because there is no SRP Section 3.9.
19.	Editorial	SRP Section 15.1.5 contains Acceptance Criteria, (GDC 31) and Evaluation Findings regarding the integrity of the reactor coolant pressure boundary (RCPB). GDC 31 establishes the general requirements for fracture toughness of the RCPB, which are implemented by 10 CFR 50.60 and associated Appendices G and H. Conformance with the requirements of 10 CFR 50.60, Appendix G and Appendix H is reviewed in SRP Section 5.2.3 for the RCPB (other than the reactor vessel) and SRP Section 5.3.1 for the reactor vessel.
20.	SRP-UDP format item	Changed CSB to SCSB (global change for this section).
21.	SRP-UDP format item	Defined NSSS.
22.	SRP-UDP format item	Moved sentence forward because CPB has been combined with SRXB, eliminating the interface.
23.	Integrated Impact No. 1049	Added statement on the primary review responsibility of the HICB for 10 CFR 50.34(f)(2)(xii) under SRP Section 7.1.
24.	Editorial	Simplified for clarity and readability.
25.	SRP-UDP format item	Moved sentence forward from "Review Interfaces" because it was inappropriately positioned.
26.	SRP-UDP format item	Changed Accident Evaluation Branch to PERB (global change for this section).
27.	Editorial/SRP-UDP formate item	Changed "is" to "are" and defined SER.
28.	Editorial	Replaced "coolability" with "ability to cool the core."
29.	**Integrated Impact 1368**	Added GDC 17 to ACCEPTANCE CRITERIA and relettered subsequent criteria.
30.	Editorial	Added abbreviations for General Design Criteria 27 and 28.
31.	Editorial	Changed "assure" to "ensure" (global change for this section).
32.	Editorial	Added abbreviation for General Design Criterion 31.
33.	Editorial	Added abbreviation for General Design Criterion 35.

Item	Source	Description
34.	Integrated Impact Nos. 1045, 1049, 1053, 1080, 1101, 1118, and 1128	1045Deleted TMI Task Action Plan item II.K.2.8 citation; 1049Replaced II.E.1.2 citation with 10 CFR 50.34(f)(2)(xii); 1053Deleted 1080Deleted citation of NUREG 0694 replaced items II.K.3.25 and II.K.2.16 citations with 10 CFR 50.34(f)(1)(iii); 1101Deleted item II.E.2.16 citation (partial duplication of 1080); 1118Revised include issue resolution described in Generic Letters 83-10A through 83-10F, 85-12, 86-05, and 86-06; and II.K.3.40 citation.
35.	SRP-UDP format item	Deleted definition of NSSS, which is defined in item I.4.
36.	**Integrated Impact 1368**	Added the new staff position that GDC 17 requires that LOOP not be considered as a single failure.
37.	**Integrated Impact 1368**	Added the new staff position that GDC 17 requires that LOOP not be considered as a single failure.
38.	SRP-UDP format item	Defined MSIVs.
39.	Editorial	Corrected misusage of "i.e." and defined RCP.
40.	SRP-UDP format item, develop technical rationale	"Technical Rationale" added to "ACCEPTANCE CRITERIA" and organized in paragraph form.
41.	SRP-UDP format item - Develop technical rationale	Added lead-in sentence for "Technical Rationale."
42.	**Integrated Impact 1368**	Added technical rationale for GDC 17.
43.	SRP-UDP format item	Added technical rationale for GDC 27 and GDC 28.
44.	SRP-UDP format item	Added technical rationale for GDC 31.
45.	SRP-UDP format item	Added technical rationale for GDC 35.
46.	SRP-UDP format item	Added a reference to combined license (COL) reviews.
47.	SRP-UDP format item	Added a reference to COL review stage.
48.	Editorial	Modified to improve clarity and eliminate gender-specific reference.
49.	SRP-UDP format item	Revised sentence because CPB has been combined with SRXB, eliminating a need for consultation.
50.	Editorial	Revised sentence to improve clarity.
51.	SRP-UDP format item	Changed ASB to SPLB.
52.	Editorial	Reorganized a complex sentence for clarification.
53.	**Integrated Impact 1368**	Added discussion that loss of offsite power may not be considered as a single failure.
54.	Editorial	Deleted "(AFWS)" because the abbreviation had been defined previously.

SRP Draft Section 15.1.5

Attachment A - Proposed Changes in Order of Occurrence

Item	Source	Description
55.	Integrated Impact Nos. 1045, 1049, 1053, and 1056	1045Deleted TMI Task Action Plan item II.K.2.8 citation; 1049Replaced II.E.1.2 citation with 10 CFR 50.34(f)(2)(xii); 1053Deleted 1056Replaced item II.E.1 citation with 10 CFR 50.34(f)(1)(ii).
56.	Integrated Impact No. 1118	Revised sentence to include compliance with the requirements of Task Action Plan item II.K.3.5 and the staff resolution of this issue.
57.	Editorial	Corrected misusage of "i.e."
58.	Integrated Impact Nos. 1080 and 1128	1080Replaced items II.K.3.25 and II.K.2.16 citations with 10 CFR 50.34(f)(1)(iii); and 1128Delete II.K.3.40 citation.
59.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard paragraph to address application of Review Procedures in design certification reviews.
60.	Editorial	Eliminated gender-specific reference.
61.	**Integrated Impact 1368**	Identified GDC 17 as a criterion to be considered in EVALUATION FINDINGS.
62.	Editorial	Replaced "insertability" and "coolability."
63.	Editorial	Replaced "insertability."
64.	**Integrated Impact 1368**	Inserted reference to GDC 17 that should be considered in EVALUATION FINDINGS of SRP Section 15.1.5.
65.	Integrated Impact Nos. 1045, 1049, 1053, and 1056	1045Deleted TMI Task Action Plan item II.K.2.8 citation; 1049Replaced II.E.1.2 citation with 10 CFR 50.34(f)(2)(xii); 1053Deleted 1056Replaced item II.E.1 citation with 10 CFR 50.34(f)(1)(ii).
66.	Integrated Impact Nos. 1080 and 1128	1080Replaced items II.K.3.25 and II.K.2.16 citations with 10 CFR 50.34(f)(1)(iii); and 1128Delete II.K.3.40 citation.
67.	SRP-UDP Format Item, Implement 10 CFR 52 Related Changes	To address design certification reviews a new paragraph was added to the end of the Evaluation Findings. This paragraph addresses design certification specific items including ITAAC, DAC, site interface requirements, and combined license action items.
68.	SRP-UDP Guidance, Implementation of 10 CFR 52	Added standard sentence to address application of the SRP section to reviews of applications filed under 10 CFR Part 52, as well as Part 50.
69.	SRP-UDP Guidance	Added standard paragraph to indicate applicability of this section to reviews of future applications.
70.	**Integrated Impact 1368**	Discussed implementation of the new staff position regarding loss of offsite power and single failures.

Item	Source	Description
71.	**Integrated Impact 1368**	Added GDC 17 as a listed reference and renumbered subsequent references.
72.	Integrated Impact No. 1080	Deleted reference to NUREG-0694, which was superseded by NUREG-0737. Renumbered references that followed.
73.	Integrated Impact No. 1118	Added a reference for each of the generic letters that addressed the resolution of Task Action Plan item II.K.3.5.

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Attachment B - Cross Reference of Integrated Impacts

Integrated Impact No.	Issue	SRP Subsections Affected
1045	Consider deleting current citations of TMI Action Plan item II.K.2.8. Item II.K.2.8 was one of the short-term action items applicable only to Babcock and Wilcox (B&W) plants. The requirements of item II.K.2.8 were completed by the licensees and evaluated by the NRC staff before allowing restart of those plants. Long-term actions related to the issue were addressed under TMI Action Plan items II.E.1.2 and II.E.1.2 of NUREG 0737. These items were applicable to all operating licensees and applicants and were also cited in SRP Section 15.1.5. The II.K.2.8 citation is outdated and duplicative and need not be retained.	Subsection II, ACCEPTANCE CRITERIA, second paragraph Subsection III, REVIEW PROCEDURES, paragraph 6 Subsection IV, EVALUATION FINDINGS, paragraph (g)
1049	Consider deleting citations to TMI Action Plan item II.E.1.2; referencing 10 CFR Part 50.34(f)(2)(xii), which established equivalent requirements; and addressing the review of instrumentation and controls associated with auxiliary feedwater systems conducted by HICB under SRP Sections 7.1, 7.3, and 7.5.	Subsection I, AREAS OF REVIEW, Review Interfaces, paragraph 5 Subsection II, ACCEPTANCE CRITERIA, second paragraph Subsection III, REVIEW PROCEDURES, paragraph 6 Subsection IV, EVALUATION FINDINGS, paragraph (g)
1053	Consider deleting current citations of TMI Action Plan item II.K.2.1. Item II.K.2.1 was one of the short-term action items applicable only to B&W plants. The requirements of item II.K.2.1 were completed by the licensees and evaluated by the NRC staff before allowing restart of those plants. Long-term actions related to the issue were addressed under TMI Action Plan items II.E.1.1 and II.E.1.2 of NUREG 0737. These items were applicable to all operating licensees and applicants and were also cited in Section 15.1.5. The II.K.2.1 citation is outdated and duplicative and need not be retained.	Subsection II, ACCEPTANCE CRITERIA, second paragraph Subsection III, REVIEW PROCEDURES, paragraph 6 Subsection IV, EVALUATION FINDINGS, paragraph (g)

Integrated Impact No.	Issue	SRP Subsections Affected
1056	Consider deleting citations to TMI Action Plan Item II.E.1.1; referencing 10 CFR Part 50.34(f)(1)(ii) that established equivalent requirements; and addressing the review of auxiliary feedwater systems conducted by SPLB under SRP Section 10.4.9.	Subsection I, AREAS OF REVIEW, Review Interfaces, paragraph 1 Subsection III, REVIEW PROCEDURES, paragraph 6 Subsection IV, EVALUATION FINDINGS, paragraph (g)
1080	Consider revising ACCEPTANCE CRITERIA associated with TMI Action Plan items II.K.3.25 and II.K.2.16. These items of NUREG-0694 are currently cited in SRP Section 15.1.5. They were two short-term action items applicable only to B&W plants, which were completed by the licensees and evaluated by the NRC staff before allowing restart of those plants. Long-term actions related to these issues were addressed by NUREG 0737, also cited in Section 15.1.5. Subsequently, both issues were addressed by 10 CFR 50.34(f)(1)(iii). Thus, the II.K.3.25 and II.K.2.16 citations are outdated and duplicative.	Subsection II, ACCEPTANCE CRITERIA, second paragraph Subsection III, REVIEW PROCEDURES, paragraph 8 Subsection IV, EVALUATION FINDINGS, paragraph (h)
1101	Consider revising ACCEPTANCE CRITERIA associated with TMI Action Plan item II.K.2.16 of NUREG-0694, currently cited in SRP Section 15.1.5. This is a short-term action item applicable only to B&W plants that was completed by the licensees and evaluated by the NRC staff before allowing restart of those plants. Long-term actions related to this issue were addressed by NUREG 0737, which was also cited in Section 15.1.5. Subsequently, the issue was addressed by 10 CFR 50.34(f)(1)(iii). Thus, the II.K.2.16 citation is outdated and duplicative. (This issue was also addressed as part of Integrated Impact No. 1080.)	Subsection II, ACCEPTANCE CRITERIA, second paragraph Subsection III, REVIEW PROCEDURES, paragraph 8 Subsection IV, EVALUATION FINDINGS, paragraph (h)
1118	Consider revising existing REVIEW PROCEDURES to incorporate the resolution of TMI Action Plan item II.K.3.5 regarding automatic reactor coolant pump trips.	Subsection II, ACCEPTANCE CRITERIA, second paragraph Subsection III, REVIEW PROCEDURES, paragraph 7 Subsection VI, REFERENCES, new References 8 through 16

Integrated Impact No.	Issue	SRP Subsections Affected
1128	Consider deleting citation of TMI Action Plan item II.K.3.40. TMI Action Plan item II.K.3.40 was superseded by item II.K.2.16 before the publication of NUREG-0660.	Subsection II, ACCEPTANCE CRITERIA, second paragraph Subsection III, REVIEW PROCEDURES, paragraph 8 Subsection IV, EVALUATION FINDINGS, paragraph (h)
1368	Modify ACCEPTANCE CRITERIA, REVIEW PROCEDURES, and EVALUATION FINDINGS to include General Design Criterion (GDC) 17, "Electric Power Systems." GDC 17 provides the basis for requiring the applicant to demonstrate the following by analysis for steam system piping failures concurrent with loss of offsite power and in combination with a single failure: (1) that acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded, and (2) that the core is cooled and containment integrity and other vital functions are maintained.	Subsection II, ACCEPTANCE CRITERIA, Criterion A Subsection III, REVIEW PROCEDURES, paragraph 2 Subsection IV, EVALUATION FINDINGS, Introduction to example statement, and paragraph d Subsection V, IMPLEMENTATION Subsection VI, REFERENCES, Reference 2.