U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

SECTION 15.6.5 LOSS-OF-COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

REVIEW RESPONSIBILITIES

Primary - Reactor Systems Branch (RSB)

Secondary - Accident Analysis Branch (AAB)
Auxiliary and Power Conversion Systems Branch (APCSB)
Containment Systems Branch (CSB)
Core Performance Branch (CPB)
Electrical, Instrumentation and Control Systems Branch (EICSB)
Mechanical Engineering Branch (MEB)

I. AREAS OF REVIEW

Loss-of-coolant accidents (LOCA) are postulated accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from piping breaks in the reactor coolant pressure boundary. The piping breaks are postulated to occur at various locations and to include a spectrum of break sizes, up to a maximum pipe break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant pressure boundary. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished.

Each pressurized water reactor (PWR) and boiling water reactor (BWR) must be equipped with an emergency core cooling system (ECCS) that refills the vessel in a timely manner to satisfy the requirements of the regulations for ECCS (Ref. 1) and the applicable general design requirements (see Standard Review Plan 6.3). The analysis of ECCS performance has an impact on the design of the piping and support structures for the reactor coolant system, the design of the steam generators, the containment design, and the possible need for pump overspeed protection.

The review of the applicant's analysis of the spectrum of postulated loss-of-coolant accidents is closely associated with the review of the ECCS, as described in Standard Review Plan (SRP) 6.3. As a portion of the review effort described in this plan and in SRP 6.3, RSB evaluates whether the entire break spectrum (break size and location) has been covered; whether the appropriate break locations, break sizes, and initial conditions were selected in a manner that conservatively predicts the consequences of the LOCA for evaluating

ECCS performance; and whether an adequate analysis of possible failure modes of ECCS equipment and the effects of the failure modes on ECCS performance have been provided. For postulated break sizes and locations, the RSB review includes the postulated initial reactor core and reactor system conditions, the postulated sequence of events including time delays prior to and after emergency power actuation, the calculation of the power, pressure, flow and temperature transients, the functional and operational characteristics of the reactor protective and ECCS systems in terms of how they affect the sequence of events, and operator actions required to mitigate the consequences of the accident.

The calculational framework used for the evaluation of the ECCS system in terms of core behavior is called an evaluation model. It includes one or more computer programs, the mathematical models used, the assumptions and correlations included in the program, the procedure for selecting and treating the program input and output information, the specification of those portions of the analysis not included in computer programs, the values of parameters, and all other information necessary to specify the calculational procedure. The evaluation model used by the applicant must comply with the acceptance criteria for ECCS (Ref. 1). The evaluation model must have been previously documented and reviewed and approved by the staff. Should the LOCA blowdown calculations be modified for the purpose of studying structural behavior (for example, core support structure design, control rod guide structure design, steam generator design, reactor coolant system piping and support structure design), all differences should be identified and described by the applicant. On request, RSB reviews these modifications, including analytical techniques, computer programs, values of input parameters, break size, type, and location, and all other pertinent information, and makes recommendations regarding their acceptability to other branches as required. RSB requests generic computer code reviews from CPB as required.

RSB is also responsible for the review of the failure mode analysis of the ECCS in conjunction with the effort described in SRP 6.3. APCSB and EICSB provide assistance in this review, on request.

AAB provides an evaluation of fission product releases and radiological consequences. This effort is described in the appendices to this review plan.

APCSB, as described in the plans for SAR Chapters 9 and 10, provides an evaluation of auxiliary systems (e.g., service water system, component cooling system, ultimate heat sink, condensate storage facility) to confirm that these systems can supply all the functions required to support the ECCS in performing its function during and following a loss-of-coolant event. APCSB also, on request from RSB, reviews the failure mode analysis of the ECCS.

CSB, as described in SRP 6.2.1, evaluates the functional capability of the containment for the spectrum of loss-of-coolant events. The assumptions used for the containment response analysis must be selected in a manner conservative for the purpose. CSB, on request from the RSB, also provides an evaluation of containment pressure calculations utilized in the reflood portion of the ECCS performance analyses.

CPB, upon RSB request, reviews the power transient calculations including moderator temperature, void and fuel temperature feedback effects, and decay heat; reviews the analytical techniques used for blowdown, reflood, and clad temperature calculations; and performs independent blowdown, reflood, and clad temperature calculations, as described in SRP 4.3 and the appendix to SRP 4.4.

EICSB, as described in SRP 7.2, 7.3, 8.3.1 and 8.3.2, reviews the protection system and ECCS-associated controls and instrumentation with regard to automatic actuation, remote sensing and indication, remote control, redundancy, and emergency onsite power functional capabilities.

EICSB also, on request from RSB, reviews the failure mode analysis related to the instrumentation and electrical power supply submitted by the applicant to show that the most damaging single active failure of the ECCS was selected for the LOCA analysis.

MEB, as described in SRP 3.62 and the 3.9 plans, is responsible for the review of the effects of blowdown loads on core support structures and on control rod guide structures. EB verifies that the core remains in place in case of a LOCA and that the control rods can inserted. MEB is also responsible for evaluating the effects of blowdown loads including jet forces on the piping of the reactor coolant system and on the support structures of the components of the reactor coolant system. MEB verifies that acceptable criteria have been employed in the design of the reactor coolant system and its supports to prevent failures in the reactor coolant pressure boundary or in engineered safety feature equipment in the event of a LOCA.

II. ACCEPTANCE CRITERIA

The objectives of the review of the applicant's analysis of loss-of-coolant accidents are to verify that:

1. An devaluation of ECCS performance has been performed in accordance with an approved

   breaks, the results of the evaluation must show that the requirements of the acceptance criteria for ECCS are satisfied, namely:

   a. The calculated maximum fuel element cladding temperature does not exceed 2200°F.

   b. The calculated total oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

   c. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

   d. Calculated changes in core geometry are such that the core remains amenable to cooling.

e. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

2. The radiological consequences of the most severe LOCA are within the guidelines of 10 CFR Part 100.

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, 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.

For the review of the ECCS performance analysis, as presented in the applicant's safety analysis report (SAR), the reviewer verifies the following:

1. The calculations were performed using an approved evaluation model. The application should clearly state this and properly reference the evaluation model. If the analysis is done with a new evaluation model, a generic review of the new model is required.

2. An adequate failure mode analysis has been performed to justify the selection of the most limiting single active failure. This analysis is reviewed in part under SRP 6.3. If the design has been changed from that presented in previous applications, changes in the reactor coolant system, reactor core, and ECCS are reviewed with respect to the most limiting single failure.

3. A variety of break locations and the complete spectrum of break sizes were analyzed. If part of the evaluation is done by referencing earlier work, design differences (ECCS, reactor coolant system, reactor core, etc.) between the facilities in question are reviewed. If there are significant differences, sensitivity studies on the important parameters should have been made by the applicant. If such sensitivity studies are not presented in the SAR, the reviewer requests that they be made.

4. The parameters and assumptions used for the calculations conform to those of the approved evaluation model and were conservatively chosen, including the following points:

   a. Initial power level should be 102% of the proposed licensed core thermal power, as given in SAR Section 4.1.

   b. The maximum linear heat generation rate used should be based on 102% of the proposed licensed core thermal power and the technical specification limit on peaking factors, or on the technical specification limits on maximum linear heat generation rate.

c. All permitted axial power shapes, as given in Section 4.3 of the SAR should be covered by the analyses. Normally, the evaluation model will identify the least favorable axial shape as a function of break size. If the evaluation model did not discuss axial shapes, or the discussion is not applicable to a given case, sensitivity studies are requested.

d. The initial stored energy was conservatively calculated by the applicant. The value used is checked against the applicant's steady-state temperatures, as given in SAR Section 4.4, similiar calculations performed by the staff, or calculations done for similar plants by previous applicants.

e. Appropriate analyses are presented to support any credit taken for control rod insertion.

5. Reactor protection system actions and safety injection actuation and delivery are consistent with the set points and the associated uncertainties and delay times listed in the SAR (OL review). The ECCS flow rates should be checked against the applicant's data on head-flow characteristics of the ECCS pumps given in Section 6.3 of the SAR and against typical safety injection tank discharge curves used for the analysis. The Regional Offices may be requested to provide data of this type from the startup tests for new designs and from periodic tests on duplicate designs.

6. The results of the applicant's calculations are consistent with those of staff calculations for typical plants and also with the results of calculations performed for similar systems by previous applicants. The following variables should be reviewed on a generic basis and spot-checked thereafter: power transients for various breaks; pressure transients at various system locations; flow transients near the break, in core, and in the downcomer; reactor coolant temperature and quality at core inlet, core outlet, and in-core; cladding temperature transients (core average, hot assembly, hot pin); heat transfer coefficients during blowdown, refill, and reflood; heat flux transients from piping and vessel walls; primary-secondary heat transfer (PWRs only); timing of clad rupture (if the peak clad temperature could be appreciably higher when perforation occurs at a different but equally probable time, calculations with modified assumptions are requested); peak clad temperature as a function of break size (if it is uncertain whether the peak value has been found, additional calculations are requested); predicted "end-of-bypass" time compared to calculated downcomer flow and to staff calculations for typical plants; pump speed transients; containment pressure transients (if staff calculations are not available, these are requested from CSB); and carryover fraction (if it is not an input to the calculations).

7. The calculated peak clad temperature, maximum local oxide thickness, and core average zirconium-water reaction meet the acceptance criteria for ECCS (Ref. 1).

8. The applicant's analysis covers the full LOCA sequence of events to the point where the plant is in the long-term cooling mode and removal of decay heat has been well

established. The reviewer checks the assumed sources of coolant water, the redundancy of delivery routes, the alignment of valves, and all required operator actions.

The review of fission product releases and radiological consequences of design basis (most severe) LOCA is performed by AAB as described in the appendix to this plan.

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 applicant has performed analyses of the performance of the emergency core cooling system (ECCS) in accordance with the Commission's regulations (10 CFR § 50.46). The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model which had been previously reviewed and approved by the staff. The results of the analyses show that the ECCS satisfy the following criteria:

1. The calculated maximum fuel rod cladding temperature does not exceed 2200°F.

2. The calculated maximum local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. Calculated changes in core geometry are such that the core remains amenable to cooling.

5. After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

"The radiological consequences of the postulated spectrum of loss-of-coolant accidents (LOCA) were evaluated from the viewpoint of site acceptability. For the purposes of this analysis, large fractions of the fission products were assumed to be released from the core even though these releases would be precluded by the performance of the ECCS."

The evaluation findings of the AAB resulting from the reviews detailed in Appendices A, B, C, and D, as applicable, should be inserted in the safety evaluation report draft at this point. See Appendices A - D for typical findings and conclusions.

"The staff concludes that the calculated performance of the emergency core cooling system following a postulated loss-of-coolant accident and the conservatively calculated radiological consequences of such an accident conform to the Commission's regulations and to applicable regulatory guides and staff technical positions and, accordingly, the ECCS is considered acceptable."

. REFERENCES

1. 10 CFR § 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," and Appendix K to Part 50, "ECCS Evaluation Models."

2. Standard Review Plan 6.3, "Emergency Core Cooling System."

3. Appendices A, B, C, and D, attached to this plan.

APPENDIX A

STANDARD REVIEW PLAN 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: CONTAINMENT LEAKAGE CONTRIBUTION

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Containment Systems Branch (CSB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

1. The review is concerned with the selection of the values of plant parameters used in calculating the radiological consequences of containment leakage following a loss-of-coolant accident. It is also concerned with selecting a dose computation model that incorporates conservative transport mechanisms and rates from various parts of the containment to the atmosphere, suitable breathing rates, dose conversion factors, and other physical and biological data that may affect the computed dose.

2. The calculated doses are compared with the appropriate exposure guidelines to confirm the acceptability of the nearest exclusion area boundary and low population zone (LPZ) outer boundary and to confirm the adequacy of the engineered safety features (ESF) provided for the purpose of mitigating potential accident doses.

The ETSB reviews ESF filter system design and filter efficiencies in SRP 6.5.1.

II. ACCEPTANCE CRITERIA

1. The fractions of the fission product inventory assumed to be available for release from the containment are acceptable if they agree with the values listed in Section C of Regulatory Guide 1.3 or Regulatory Guide 1.4. No specific list of isotopes or decay constants has been selected as standard.

Where the applicant claims a single containment system, this is accepted. To receive credit for a dual containment system, a determination must be made that the system meets the necessary requirements. These requirements are detailed in Standard Review Plan (SRP) 6.2.3 and SRP 6.5.3. Containments falling outside of these categories are evaluated on a case-by-case basis. For single containment systems, or leakage from the primary containment, the leakage rate stated in the technical specifications is accepted subject to verification by the CSB, provided a leakage rate of at least 0.1% per day is stated. For a boiling water reactor (BWR) the leakage rate is currently assumed constant over the course of the accident, while for a pressurized water reactor (PWR), the leakage rate is reduced after 24 hours to one-half its original value (see Refs. 2 and 3). Where a single containment is specified, no credit for exhaust filters is allowed, although internal recirculation filters can be credited, if present.

2. The methods used to calculate radiological consequences of a postulated LOCA are acceptable if they reflect the use of conservative design basis assumptions as outlined in Regulatory Guide 1.3 or 1.4 (Refs. 2 and 3). The requirements of 10 CFR Part 100 are that the total dose from a postulated loss-of-coolant accident (LOCA) to an individual (located at positions specified in 10 CFR Para. 100.11(a)) must be no greater than 300 rem to the thyroid and 25 rem to the whole body. At the construction permit (CP) stage, exposures of no more than 150 rem to the thyroid and 20 rem to the whole body are considered acceptable to allow for uncertainties in meteorology and other site-related data and to allow for system design changes that might influence the final design of engineered safety features or the dose reduction factors allowed for these features. This lower guideline is required at the CP stage to provide reasonable assurance that the 10 CFR Part 100 guideline values can be met at the operating license (OL) review stage.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as may be appropriate for a particular case. The judgment on the areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

1. The design (stretch) power level of the core is taken from the applicant's safety analysis report (SAR). The core is assumed to have operated at this power level for a sufficiently extended period (typically about 3 years) that a maximum equilibrium fission product inventory is present. At time of the accident, 25% of all the equilibrium iodine fission products and 100% of the noble gas fission products are assumed available for release from the containment within a very short time (effectively instantaneously) after the accident. The iodine is assumed to be composed of 91% elemental iodine, 4% organic iodides, and 5% particulate iodine.

2. From the applicant's SAR (parts of Section 1, Sections 6.2.1, and 6.2.3), the reviewer ascertains the type of containment system used. A check is made of the LOCA assumptions listed in Chapter 15 of the SAR to verif t the primary containment leakage rate has been assumed to remain constant over the course of the accident (for a BWR) or to be halved after 24 hours (for a PWR) and that the initial leak rate is at least 0.1% per day (a lower limit is set because of integrated containment leakage test sensitivity limitations). The leakage rate used should correspond with that given in the technical specifications in SAR Chapter 16.

3. Where credit for a dual containment system is claimed, the reviewer verifies (see SRP 6.2.3 and 6.5.3) that the system meets requirements such as existence of separate primary and secondary containments, adequate separation of the two, and ability to test the negative pressure capability of the secondary containment volume. Where credit

for a secondary containment with recirculation is claimed, adequate mixing in the secondary containment volume should be demonstrated in addition to meeting the above requirements for a dual containment system. For dual containment systems, the bypass leakage rate is noted. This leakage, usually expressed as a fraction or percentage of the primary containment leak rate, is assumed to go from the primary containment directly to the environment, bypassing the secondary containment. This bypass leakage rate, as well as any positive pressure conditions should be verified by the CSB. See SRP 6.2.3 for a detailed treatment of bypass leakage.

4. Credit, if any, to be given for any engineered safety features such as filters, sprays, or ice condenser that may be present, is determined in the review of Section 6.5 of the SAR. These features operate during the LOCA to mitigate the consequences by reducing the amount of iodine fission products released to the environment. Noble gas releases to the environment are unaffected by the presence of filters or sprays. Typically, single containments employ spray systems with a chemical additive (e.g., sodium hydroxide, sodium tetraborate) designed to scavenge iodine from the containment atmosphere. The iodine removal rates of an ice condenser or a chemical additive spray system are determined after consultation with specialists in this area. For filters, verification of acceptability of design and filter efficiencies is provided by the ETSB in SRP 6.5.1. In dual containment systems, a determination must be made by the AAB of the operational modes of the ESF with respect to the accident sequence in order for proper credit to be given.

5. Sections 2.1.2 and 2.1.3 of the applicant's SAR are examined to determine the minimum distances to the exclusion area boundary and to the LPZ outer boundary. Following the procedures given in SRP 2.1.2 and SRP 2.1.3, the reviewer confirms the validity of the applicant's values. From the SAR, the reviewer also obtains relevant information (e.g., locations and time durations) concerning activities unrelated to plant operation that may exist inside the exclusion area boundaries (see SRP 2.1.2). In some cases specific dose computations may have to be performed to assist in determining the adequacy of evacuation plans.

6. The SAB is requested to furnish suitable X/Q values to be used in analyzing the consequences of the accident. X/Q values are obtained not only at the nearest exclusion area boundary and the outer boundary of the LPZ, but also at those locations inside the exclusion boundary where significant activities may occur involving members of the public.

7. Based upon the review procedures already performed, a dose computation model is selected which conservatively represents the transfer of radioactivity from the containment to the environment. The reviewer may find it convenient to sketch a schematic arrangement to illustrate the compartments where radioactivity is located, with arrows drawn from one compartment to another indicating transport paths. The leak rates, spray removal rates, ice condenser efficiencies, filter efficiencies, and flow rates are all used to indicate the rates at which the activity moves from one compartment to another. Digital computer codes (Ref. 4) have been written to perform

the actual dose calculation. The analyst should select the code with capabilities that most closely fit the schematic model obtained above. The codes contain a basic library of physical and biological data which enter into the dose calculation, such as isotopic fission yields, half-lives, energies, and dose conversion factors.

8. The calculated doses, including the 2-hour thyroid inhalation and whole body doses at the nearest exclusion area boundary, the thyroid inhalation and whole body doses for the course of the accident at the outer boundary of the LPZ, and those doses calculated at other points within the exclusion area boundary where there may be activities unrelated to to plant operation at certain times, are compared with the dose guidelines as discussed in Section II.2 of this plan. Where the results of the dose calculations exceed the guidelines, the alternatives which would reduce the doses to an acceptable level are explored (e.g., increased distance, secondary containment, better filter or spray systems). The feasibility of the alternatives is also examined. The AAB Branch Chief is consulted as to appropriate action in this case.

IV. EVALUATION FINDINGS

If the AAB reviewer finds that the radiological consequences of the containment leakage contribution to a loss-of-coolant accident are acceptable, conclusions of the following type may be included with the RSB findings for this area in the staff's safety evaluation report:

"The radiological consequences of a loss-of-coolant accident as a result of leakage from the containment were evaluated. The analysis of the containment leakage doses following a postulated design basis loss-of-coolant accident included the influence of fission product removal and holdup systems and the containment leakage routes on the estimated radiological consequences.

"The review has included the applicant's proposed design criteria and design bases for the effect of containment leakage and his analysis of the manner in which the containment leakage consequences conform to the design criteria.

"The basis for acceptance in the staff review has been confirmation that the applicant's analysis conforms with the applicable regulations, regulatory guides, technical positions and industry standards as listed in Table 15.13.2-1. The staff concludes that the proposed design, including leakage rates and fission product removal and control systems conform to the Commission's regulations and to applicable regulatory guides and staff technical positions, and that the conservatively computed doses from containment leakage following a loss-of-coolant accident are within the exposure guidelines of 10 CFR Part 100."

Appropriate tables of assumptions used and the estimated consequences are to be included in the SER. The following should be added at the CP stage: "Because the proposed design meets the recommendations of Regulatory Guide 1.3(1.4), there is reasonable assurance that the exposure guidelines of 10 CFR Part 100 can be met at the OL stage."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2.

3. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Revision 2.

4. Computer codes are currently under development. Documentation will be published in a NUREG report.

APPENDIX B

STANDARD REVIEW PLAN 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: LEAKAGE FROM ENGINEERED SAFETY FEATURES COMPONENTS OUTSIDE CONTAINMENT

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Site Analysis Branch (SAB)
Effluent Treatment Systems Branch (ETSB)

I. AREAS OF REVIEW

A potential source of fission product leakage following a loss-of-coolant accident (LOCA) is the leakage from engineered safety features (ESF) equipment which is located outside the primary containment. Such leakage could occur during the recirculation phase for long-term core cooling and primary containment (spray) cooling. The total leakage from these sources is added to that resulting from the containment leakage following a LOCA. To calculate the maximum potential leakage from the recirculation loop, such sources as the following are considered: containment spray system, low pressure safety injection system, and high pressure safety injection system.

The ETSB reviews ESF ventilation system filters for conformance with Regulatory Guide 1.52 in SRP 6.5.1.

II. ACCEPTANCE CRITERIA

The source of leakage is related to the requirement to detect and isolate failures of passive components in the long-term (recirculation) mode for ESF systems. Therefore, leakage anywhere in the systems carrying recirculation water outside of containment is postulated. ESF-grade filtration systems to process potential leakage are required as the dose could exceed 10 CFR Part 100 guidelines without filters even at relatively low leakage rates resulting from passive failures. When ESF-grade filters are supplied, no doses resulting from passive failures need be considered.

The acceptance criterion for the dose resulting from leakage outside primary containment from the recirculation systems is that when it is added to the dose attributable to containment leakage, including any main steam isolation valve sealing system leakage (Appendices A and D of Standard Review Plan 15.6.5), the total dose is to be within the guideline values of 10 CFR Part 100. To provide assurance that this criterion is met at the operating license stage, the doses indicated in Regulatory Guides 1.3 and 1.4 are used as acceptance criteria at the construction permit stage.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review as may be appropriate for a particular case. The judgment on the areas to be given attention and emphasis in the review is based on an inspection of the material presented to see if it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The applicant's recirculation leakage calcualtion is checked against previously licensed plants for accuracy and completeness. It is assumed that 50% of the core iodine inventory, based upon the maximum reactor power level, is mixed in the sump water being circulated through the external piping systems. Credit may be allowed for radioactive decay of the iodine during the time period from the occurrence of the LOCA up to the beginning of recirculation when the sump water is circulated outside the containment.

The dose computed for presentation in the staff safety evaluation report (SER) should be based upon twice the maximum operational leakage and should be assumed constant for the course of the accident. The maximum operational leakage is defined as the sum of the leakage for all the recirculation systems (1) which is detectable during test and (2) above which the technical specifications whould require declaring a system out of service. The leakage is assumed to occur throughout the accident, starting at the earliest time that recirculation mode is initiated.

The applicant's data on sump water temperature versus time after the LOCA should be consulted and used. During the time that the circulating water temperature exceeds 212^{O}F, the fraction of water flashing to steam should be computed and taken as the fraction of iodine in the water which becomes volatile. In those cases where the circulating water temperature is less than 212^{O}F, 10% of the iodine in the water which leaks is assumed to become volatile unless a smaller amount is justified based on actual sump pH history and ventillation rates.

All the iodine becoming volatile is assumed to be released immediately to the environment, and atmospheric dispersion is based upon the ground level X/Q values determined by the SAB. Any ventilation system filters are evaluated by the ETSB for compliance with Regulatory Guide 1.52 (Ref. 4) and appropriate credit for iodine removal by the filters given. The nearest exclusion area boundary and LPZ outer boundary doses are calculated by standard methods as described in Appendix A to SRP 15.6.5.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included with the RSB findings for Section 15.6.5 in the SER:

"The staff concludes that doses resulting from the postulated leakage of post-LOCA recirculation water from pump seals, valve packings, etc., are low and, when added to the direct leakage LOCA doses, result in total doses that are within the guideline values of 10 CFR Part 100. Engineered safety feature-grade filtration systems are provided to process potential leakage from postulated failures of passive components in systems carrying post-LOCA recirculation water outside of containment."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2.

3. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Revision 2.

4. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

5. Appendices A and D, Standard Review Plan 15.6.5.

APPENDIX C

STANDARD REVIEW PLAN 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: HYDROGEN PURGE CONTRIBUTION

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Containment Systems Branch (CSB)
Effluent Treatment Systems Branch (ETSB)
Site Analysis Branch (SAB)

I. AREAS OF REVIEW

The radiological consequences of purging any hydrogen accumulation in the containment after a postulated loss-of-coolant accident (LOCA) are reviewed to establish that the LOCA-plus-purge doses are acceptable and, in some cases, to determine whether additional filtration systems are needed. The ETSB reviews hydrogen purge system filters in SRP 6.5.1.

II. ACCEPTANCE CRITERIA

The acceptance criteria for hydrogen purging doses are given in Section B of Branch Technical Position CSB 6-2 (Ref. 2).

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the area covered by this review plan as may be appropriate for each particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The reviewer determines which criteria in Reference 2 are to be met and then performs a purging dose calculation following the procedures outlined below.

1. Source Terms

a. Iodine

The initial airborne iodine-131 component is assumed to be 25% of the core inventory, as stated in Regulatory Guides 1.3 and 1.4. The iodine airborne activity at any subsequent time is subject to a removal factor due to the plant engineered safety features and to radioactive decay. (See Enclosure 1)

b. Noble Gases

The initial Xe-133 and Kr-85 activities are assumed to be 100% of the core inventory, as stated in Regulatory Guides 1.3 and 1.4. These nuclides are subject to removal through radioactive decay only. (See Enclosure 2, 3)

2. Purging

The model assumes a constant purge rate after initiation of purge. No credit is included for depletion of activity due to containment leakage.

3. Dose Model for I-131

a. Assumptions

	Code	Units
(1) 50% plateout	PLTOUT	-
(2) 50% released	REL	-
(3) Core inventory	TID	Ci/MWt
(4) Dose conversion factor	DCFLOD	Rad/Ci
(5) Breathing rate (4-30 day rate)	BR	m^3/sec
(6) Iodine-131 decay constant	LAMDA	day^{-1}

b. Variables

	Code	Units
(1) Purge rate	PURGRT	SCFM
(2) Iodine reduction factor	RF	-
(3) Purge time	PURGTM	days
(4) Power level	POWLEV	MWt
(5) X/Q	XQ	sec/m^3
(6) Hold up time	HOLDUP	days
(7) Containment building volume	VOLCON	ft^3

c. Model

	Code	Units
(1) Core inventory:	I-131	Ci
I-131 (PLTOUT)(REL)(POWLEV)(TID)/RF	-	-
(2) Activity in containment at time of purge:	CI-131	Ci
CI-131 = I-131 exp (-LAMDAxHOLDUP)	-	-
(3) Concentration in containment at time of purge:	CONCON	Ci/cm^3
CONCON = CI-131/VOLCON	-	-
(4) Differential change in containment atmosphere concentration due to replacing the portion of the atmosphere vented per unit time with clean air:		
BETA = PURGRT/VOLCON	BETA	$days^{-1}$
differential change = exp (-BETAxPURGTM)	-	-
(5) Total activity released during course of purge:	TAR	Ci
$TAR = \int_0^{PURGTM} (CONCON)(PURGRT) \times \exp(-LAMDA-BETA)\, t \; dt$	-	-
(6) Dose at boundary due to iodine-131:	DOSEIOD	Rem
DOSEIOD = (TAR)(BR)(DCFIOD)(XQ)	-	-

4. Dose Model for Xe-133

a. Assumptions

	Code	Units
(1) 0% plateout	PLTOUT	-
(2) 100% released	REL	-
(3) Core inventory	TIDXE	Ci/MWt
(4) Xe-133 decay constant	LAMDA	$days^{-1}$

b. Variables

	Code	Units
Use the I-131 variables and let iodine reduction factor = 1.	-	-

c. Model

	Code	Units
Use the I-131 model with Xe-133 assumptions.	-	-

5. Dose Model for Kr-85

a. Assumptions

	Code	Units
(1) 0% plateout	PLTOUT	-
(2) 100% released	REL	-
(3) Core inventory	TIDKR	Ci/MWt
(4) Kr-85 decay constant	LAMDA	$days^{-1}$
(5) Average gamma energy	GAMENG	Mev
(6) Average beta energy	BETENG	Mev

b. Variables

	Code	Units
Use Xe-133 variables.	-	-

c. Model

	Code	Units
(1) Core inventory:	KR-85	Ci
KR-85 = (PLTOUT)(REL)(POWLEV)(TIDKR)	-	-
(2) Activity in the containment at time of purge:	CIKR-85	Ci
CIKR-85 = KR-85 exp (-LAMDAxHOLDUP)		
(3) Concentration in containment at time of purge:	CONCON	Ci/cm^3
CONCON = CIKR-85/VOLCON		
(4) Differential change in containment atmosphere concentration due to replacing the portion of the atmosphere vented per unit time with clean air:		
BETA = PURGRT/VOLCON	BETA	$days^{-1}$
differential change = exp (-BETAxPURGTM)	-	-
(5) Total activity released during course of purge:	TAR	Ci
$TAR = \int_0^{PURGTM} (CONCON)(PURGRT) \times \exp(-LAMDA-BETA)\, t\, dt$	-	-
(6) Dose at boundary due to Kr-85 beta:	DOSWBB	Rem
DOSWBB = 0.246(TAR)(BETENG)(XQ)	-	-
(7) Dose at boundary due to Kr-85 gamma:	DOSWBG	Rem
DOSWBG = 0.246(TAR)(GAMENG)(XQ)	-	-
(8) Total dose at boundary due to Kr-85:	TKR-85	Rem
TKR-85 = DOSWBB + DOSWBG	-	-
(9) Total whole body dose at boundary:	TWBD	Rem
TWBD = TKR-85 + DOSXE	-	-

The data required for this calculation are obtained from the following sources. The CSB determines the purge rate, in SCFM, and the hold-up time (in days) prior to purging. The SAB determines the ground level release X/Q (30-day value) derived from onsite data. The ETSB in SRP 6.5.1 determines filter efficiencies in cases where filters are required to meet the dose criteria. LOCA analysis assumptions as to reactor power level, primary containment volume, and iodine reduction factor are obtained from the results of the AAB review under Appendix A to Standard Review Plan (SRP) 15.6.5.

For those plants not excepted from the requirements of Section B of Reference 2, the reviewer is responsible for transmitting the requirements for filters to the ETSB when such requirements are indicated by the results of the dose calculation.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included with the RSB findings for Section 15.6.5 in the staff's safety evaluation report:

> "The analysis of the radiological consequences of containment hydrogen purging following a LOCA yields acceptable thyroid and whole body dose values."

If the reviewer finds the consequences unacceptable, then the following may be stated for current reviews:

> "The analysis of the radiological consequences of containment hydrogen purging following a LOCA indicates that the total long-term doses from the LOCA and the purge exceed the guideline values of 10 CFR Part 100 at the LPZ outer boundary. Accordingly, remedial measures (inert gas injection or filters) are required to achieve acceptable dose levels."

Conclusions which match the acceptance criteria for older plants should be drafted for such plants.

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," attached to Standard Review Plan 6.2.5.

3. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

4. Appendix A, Standard Review Plan 15.6.5.

APPENDIX D

STANDARD REVIEW PLAN 15.6.5

RADIOLOGICAL CONSEQUENCES OF A DESIGN BASIS LOSS-OF-COOLANT ACCIDENT: LEAKAGE FROM MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEM (BWR)

REVIEW RESPONSIBILITIES

Primary - Accident Analysis Branch (AAB)

Secondary - Auxiliary and Power Conversion Systems Branch (APCSB)
Site Analysis Branch (SAB)
Containment Systems Branch (CSB)

I. AREAS OF REVIEW

A potential source of fission product leakage following a loss-of-coolant accident (LOCA) is the leakage past the main steam isolation valves in a BWR. This leakage is required to be controlled by a main steam isolation valve leakage control system (MSIVLCS). This system may act as a positive sealing system or a vacuum-type system which collects leakage between the closed isolation valves and releases it to the atmosphere through a filter system. The method of operation, time of operation, and release paths associated with the operation of the MSIVLCS are reviewed to calculate the fission product releases and their contributions to the doses at the nearest exclusion area boundary and LPZ outer boundary. Any leakage from the isolation valves (e.g., valve stem leakage) or any release from the MSIVLCS is added to the containment leakage and ESF leakage (Appendices A and B of Standard Review Plan 15.6.5) following a LOCA.

II. ACCEPTANCE CRITERIA

The calculated doses associated with operation of the MSIVLCS following a postulated LOCA should be limited so that when they are added to the dose contribution from containment leakage and leakage from ESF components outside containment, the total docs not exceed the guideline values of 10 CFR Part 100 (Ref. 1) at the operating license stage or Regulatory Guide 1.3 (Ref. 2) at the construction permit stage.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review as may be appropriate for a particular case. The judgment on the areas to be given attention and emphasis in the review is based on an inspection of the material presented to see if it is similar to that recently reviewed on other plants and whether items of special safety significance are involved.

The applicant's description of the MSIVLCS is reviewed to familiarize the reviewer with the system performance and to obtain the information needed to perform the dose calculation. For a positive sealing system, verification of the system operability assuming a single active failure, actuation time, and identification of any potential release paths is obtained from the APCSB. If the reviewer finds that no release paths exist and that the system can be actuated within an appropriate time after the accident, no further review is required.

For a vacuum-type system, which processes rather than seals the leakage, the following information, assuming the most adverse single failure of an active component, must be verified by the APCSB (and documented by buckslip to the AAB):

1. Release paths and fractions of the leakage through these paths, as a function of time, e.g., steam leakage, releases through a depressurization line, releases through drain lines, etc.

2. System actuation time.

3. Flow rates as a function of time.

4. Release points.

Interaction with systems used to mitigate the consequences of containment leakage should be noted. It may be necessary to establish with the CSB that the operation of the MSIVLCS does not adversely affect pressure transients in secondary containment regions.

The system is then modeled using a computer code (Ref. 3). The source assumed is the same as that used to estimate the containment leakage dose, but it is assumed to be instantaneously distributed in the drywell free volume at the time of the accident. Credit for decay in the drywell is given; no release is assumed up to the time of system actuation; but no credit is given for leakage from the drywell to the containment (Mark III) or the suppression pool region (Mark I and II). The main steam isolation valves are assumed to be leaking at their technical specification limit. Leakage through valve stems or drain lines to an untreated region is assumed to be released to atmosphere; releases through the MSIVLCS which are directed to treated regions are assumed to be directly to the filter intake unless the MSIVLCS flow is mechanically directed to a distributed header. If the latter is the case, then credit for mixing is given on the same basis as in other leakage to this system (see Standard Review Plan 6.5.3).

The resulting doses are calculated using the model described in Regulatory Guide 1.3 (Ref. 2). The X/Q values to be used are the accident X/Q's provided by the SAB. For systems which are designed for initial releases at later times into the accident, application of worst meteorology at the time of release may have to be considered; this will be handled on a case-by-case basis.

The doses are added to those estimated from containment leakage and the leakage from ESF components outside containment and the total is compared to the guidelines of 10 CFR Part 100 (Ref. 1) if the application is for an operating license and to the guidance of Regulatory Guide 1.3 (Ref. 2) if the application is for a construction permit.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and the review and calculations support conclusions of the following type, to be included with the RSB findings for Section 15.6.5 in the staff's safety evaluation report:

"The radiation doses resulting from main steam isolation valve leakage and operation of the main steam isolation valve leakage control system following a postulated LOCA were estimated assuming a single failure that is most adverse from the standpoint of radiological consequences. The analysis included the influence of fission product removal systems, delay times, and various release paths. The review has established that the applicant's design is sufficient to limit the radiological consequences due to the main steam isolation valve leakage or due to operation of the MSIVLCS such that when combined with the releases from other paths, the total potential consequences at the nearest exclusion area boundary and at the low population zone outer boundary are well within the guidelines of 10 CFR Part 100."

Appropriate tables of assumptions used and the estimated consequences are to be included in the SAR. The following should be added at the CP stage: "Because the proposed design meets the guideline values of Regulatory Guide 1.3, there is reasonable assurance that the exposure guideline of 10 CFR Part 100 can be met at the operating license stage."

V. REFERENCES

1. 10 CFR Part 100, "Reactor Site Criteria."

2. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," Revision 2.

3. Computer codes are currently under development. Documentation will be published in a NUREG report.

4. Appendices A and B, Standard Review Plan 15.6.5.

SRP 15.7.3