POLICY ISSUE

(Information)

December 19, 1994

SECY-94-302

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: SOURCE TERM-RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT-WATER-REACTOR DESIGNS

PURPOSE:

To inform the Commission of staff positions pertaining to source term-related technical and licensing issues and the staff's methodologies for using these staff positions in the evolutionary and passive light-water-reactor (LWR) design certification reviews.

BACKGROUND:

In SECY-90-016, "Evolutionary Light-Water-Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," the staff made a recommendation to the Commission on the use of the physically-based source term. For evolutionary LWRs, the staff proposed to consider any deviations from current methodology for calculating 10 CFR Part 100 doses on a case-by-case basis using engineering judgment and updated information on source term and equipment reliability. The Commission approved this recommendation in a staff requirements memorandum dated June 26, 1990.

Contacts: Jay Y. Lee, NRR
504-1080

John A. Kudrick, NRR
504-2871

SECY NOTE: TO BE MADE PUBLICLY AVAILABLE IN 10 WORKING DAYS FROM THE DATE OF THIS PAPER.

In SECY-92-127, "Revised Accident Source Terms for Light-Water Nuclear Power Plants," the staff provided the Commission a draft report (later published as draft NUREG-1465) describing revised accident source terms for light-water nuclear power plants to replace those of Technical Information Document (TID) -14844 which was published in 1962. The staff also described the process it would use to finalize the report, including both a public comment and peer review period. Any specific applications of the new source term prior to the completion of this process were to be submitted to the Commission for preliminary review. The staff expects to issue the final version of NUREG-1465 in late 1994.

In this paper, the staff notes that the review process for NUREG-1465 is essentially complete and presents its positions on (1) the closure of source term-related issues in its SERs for the EPRI requirements documents for both evolutionary and passive plant designs and (2) the generic implementation of source term-related issues in evolutionary and passive LWR design certification reviews. The staff is not requesting a review by the Commission of these positions because they are primarily technical applications of previous Commission policy decisions. The application of the revised source terms is specific to the design of the reactor. Therefore, these final staff positions are those taken in the staff's review of the Asea Brown Boveri-Combustion Engineering (ABB-CE) System 80+ and are those that will be taken for the passive LWR design reviews.

The final NUREG-1465 will not alter the staff's conclusion in the System 80+ FSER in regard to its acceptance of the System 80+ design for meeting the dose reference values given in 10 CFR Part 100 and the control room operator dose limit specified in GDC 19.

DISCUSSION:

The current reactor accident source terms for fission-product release from the reactor core into the containment are set forth in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." These source terms were derived from TID-14844. The regulatory guide source terms are used in conjunction with postulated design-basis accidents (DBAs). Revised reactor accident source terms have been proposed in draft NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants." These source terms were derived from an examination of a set of accident sequences for current LWR designs and reflect the current understanding of severe accidents and fission-product behavior.

The staff intends to use the reactor accident source terms given in NUREG-1465 in radiological consequence assessments in the following areas of evolutionary and passive LWR design certification reviews:

(1) equipment qualification
(2) control room habitability
(3) engineered safety features atmosphere cleanup systems
(4) primary containment leak rate
(5) containment isolation timing
(6) post-accident sampling
(7) shielding and vital area access

In this paper the staff identifies its significant technical positions relative to the implementation of the revised source terms for evolutionary and passive LWR designs. These positions are more fully discussed in Attachment 1. On the basis of these positions, the staff closed out all source term-related open issues in its SER for the EPRI Requirements Document for passive plant designs and the ABB-CE System 80+ design. These positions will provide the bases for the staff's review of passive LWR certification applications.

The revised source terms are not proposed as a requirement for existing plants. Preliminary indications are that the revised release estimates, in conjunction with improved insights on timing and best estimate credit for fission product removal mechanisms, would not result in more stringent requirements if applied to existing plants, and may support relaxation of certain present regulatory requirements, particularly those associated with time of fission-product appearance in the reactor containment.

Because it reflects our current understanding of source term behavior, the staff plans to make the revised source term information available to existing licensees for voluntary proposals to modify current requirements that may be overly conservative. However, due to the large number of design and operational areas impacted by the source term and the plant specific nature of these impacts, the staff has not determined at this time the extent to which these insights could be implemented for existing plants. A memorandum to the Commission entitled "Use of NUREG-1465 Source Term at Operating Reactors," dated September 6, 1994, discussed the staff's plans for evaluating the application of the revised source term to operating reactors, including interaction with the public and industry.

In Attachment 1, the staff discusses 12 source term-related issues of both a technical and licensing nature that pertain to either evolutionary LWRs, passive LWRs, or both. The staff previously identified all of these issues in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water (ALWR) Designs," dated April 2, 1993. Of these 12, the staff highlights *(in italics)* the 5 most significant issues (Issues 1 through 5) in the following summary discussion.

Draft NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants"

In SECY-92-127, the staff provided, for the Commission's information, a draft staff report containing proposed revised accident source terms for light-water nuclear power plants to replace those of Regulatory Guides 1.3 and 1.4. The draft report was issued for public comment in July 1992 as NUREG-1465. Draft NUREG-1465 addressed the current understanding of LWR severe accident and fission-product behavior, including the quantity and chemical forms of fission products released into the reactor containment following a severe reactor accident as well as fission-product release timing. Since issuing draft NUREG-1465, the staff has received more than 200 comments from 20 U.S. and foreign organizations.

The most significant public comments concerned the release fractions of the less volatile and non-volatile elements such as barium, strontium, and cerium during the early in-vessel release phase. The release fractions in draft NUREG-1465 of the less volatile and non-volatile elements are about 10 to 100 times higher than those proposed by EPRI in its Utility Requirements Document for Advanced Light-Water Reactors. In response to comments, the staff re-examined the release of non-volatile elements during the early in-vessel phase, and determined that the "mean" values of draft NUREG-1465 were not appropriate since they were well in excess of other measures of the uncertainty distribution, such as the 75th percentile.

Final NUREG-1465 is expected to utilize the 75th percentile values to account for the considerable uncertainty regarding the releases of the non-volatile elements in the early in-vessel phase. This will significantly reduce the release fractions from those specified in draft NUREG-1465 for the non-volatile elements in the early in-vessel phase.

The staff found in its radiological consequence assessments that the less volatile and non-volatile elements play a less significant role, contributing less than a few percent of the overall radiological consequences resulting from a DBA, compared to the other major nuclides such as noble gases, iodine, and cesium. Therefore, the staff believes that revisions to the release fractions of the less volatile and non-volatile elements in final NUREG-1465 will not materially change the ongoing staff reviews of the passive LWR designs.

Draft NUREG-1465 indicated that iodine is released initially into containment primarily (95%) in particulate form as CsI, with about 5% in volatile form, as elemental iodine (I_2). If the pH of the water in the containment is maintained at a value of 7 or above, elemental iodine would remain at a low level. Organic forms of iodine, primarily methyl iodide, are formed largely from reactions of organic compounds with elemental iodine, and may contribute significantly to offsite doses because they are not easily removed by engineered safety features such as sprays or filters.

Based on past research, about 3 percent of the elemental iodine present is expected to be converted to organic iodine. As discussed in the DBA assessment section below, the staff intends to assume, where the pH of the water in the containment is controlled, that 0.15 percent (3 percent of 5 percent) of the core iodine inventory is in organic form and available for leakage from the containment to the environment following a DBA. Although draft NUREG-1465 did not address the formation of organic iodine, final NUREG-1465 will address this issue.

The fission-product release timing differs with reactor type and the bounding design basis accident sequence chosen for the source term application. The staff will review each ALWR design on a design specific basis to determine appropriate fission-product release timing.

Most of the remaining public comments are largely favorable regarding draft NUREG-1465. The staff briefed the Commission on the status of the proposed source term update, including the discussion of major comments received on the draft NUREG-1465, on August 3, 1993. The staff also discussed it with ACRS on September 9, 1994. The staff expects to issue the final version of NUREG-1465 in late 1994.

Draft NUREG-1465 also contained a section on mechanisms for removing fission products from the reactor containment atmosphere following a severe reactor accident that was based on the realistic estimates of the natural processes as used in the NUREG-1150 ("Severe Accident Risks: An Assessment for 5 U.S. Nuclear Power Plants") analyses. To obtain more comprehensive information on removal mechanisms to be used in the licensing reviews, the staff is continuing to explore this area with NRC contractor assistance.

The staff's positions on the source term issues discussed in this paper are based on NRC contractor findings in NUREG/CR reports, preliminary findings in NRC contractor reports, extensive consultation with these contractors, the staff's engineering judgments, and the need to maintain an appropriate margin of safety.

Design-Basis-Accident Assessment

To evaluate the passive LWR submittals, the staff will utilize the current insights from source term research as described in draft NUREG-1465 regarding fission-product releases into the containment. In determining the effects of such removal mechanisms as sprays, filters, plateout, and aerosol deposition, the staff will use the best available information, including engineering judgment, for the applicable parameters.

The use of physically-based source terms as described in draft NUREG-1465 constitutes a major departure from the use of the current regulatory guide source terms. The proposed source terms lead to a qualitatively different fission-product distribution, which is a result of (1) physically-based composition, release timing, and rate of releases and (2) inclusion of removal mechanisms based on natural processes.

In Attachment 2, the staff presents the revised source terms to be used in conjunction with DBAs for reviewing passive LWR designs, including their release timing as well as duration for boiling-water reactors (BWRs) (Table 1) and for pressurized-water reactors (PWRs) (Table 2). The information in these tables, taken from the proposed final NUREG-1465, may be revised when NUREG-1465 is issued in final form. The staff does not expect that any revision will materially change its evaluation or conclusions pertaining to these designs.

The staff-proposed source term release fractions (magnitudes) in proposed final NUREG-1465 are compared to those proposed in the evolutionary and passive plant designs in Table 3 of Attachment 2. The staff has used only the gap release and the early in-vessel release for design-basis calculations in ABB-CE System 80+ design review and will be using them for the passive LWR designs reviews. These source terms encompass a broad range of accident scenarios, including significant levels of core damage with the core remaining in the vessel. These would be the most severe scenarios from which the plant could be expected to return to a safe shutdown condition. Table 3 in Attachment 2 shows that the proposed final NUREG-1465 release fractions associated with these releases are generally comparable to those of TID-14844.

The staff considers the inclusion of the ex-vessel and late in-vessel source terms to be unduly conservative for DBA purposes. Such releases would only result from core damage accidents with vessel failure and core-concrete interactions. For evolutionary and passive LWRs, the estimated frequencies of such scenarios are low enough that they need not be considered credible for the purpose of meeting 10 CFR Part 100.

On the basis of a comparison of the gap and in-vessel releases (excluding ex-vessel releases), the staff finds that the quantities of noble gases and radioiodines are in reasonable agreement with those in TID-14844, while the proposed new source terms include substantially more of the low volatility solids. For the regulatory guide source terms, the staff assumed that the fraction of these solids released represents less than 1 percent of the activity available for release, while the revised source terms show significantly larger release fractions for these materials. The important difference is that the proposed new source terms are released over a period of 1.8 (PWR) to 2.0 hours (BWR), rather than instantaneously.

In radiological assessments of DBAs for passive light-water reactor designs, the staff will:

Selectively use the source terms given in draft NUREG-1465 using only "Gap Release" and "Early In-Vessel Release" (excluding "Ex-Vessel Release" and "Late In-Vessel Release" associated with vessel failure and core-concrete interaction) in evaluating radiological consequences for DBAs, the DBA radiation environmental qualification of electrical and mechanical equipment important to safety, post-accident shielding, and Three Mile Island-related requirements (Issue 1).

Use the chemical forms of iodine of at least 95 percent cesium iodine as stated in draft NUREG-1465 with 4.85 percent of elemental iodine and hydrogen iodide and 0.15 percent organic iodide (Issue 2).

Severe Accident Assessment

The staff recognizes that the new source terms have implications for the staff consideration of severe accidents as well. The staff's resolution of severe accident issues is based on the inclusion of specific design features to provide assurance of containment integrity for approximately 24 hours following the onset of core damage. Part of the basis for the 24-hour criterion is the fact that natural fission-product removal mechanisms can significantly reduce the suspended aerosol source terms present in the containment within this time period.

Another aspect of the resolution of severe accident issues is the requirement that reasonable assurance exist that equipment needed for severe accident mitigation and post-accident sampling will survive in the severe accident environment. This was an issue raised to the Commission in SECY-90-016 and SECY-93-087 and approved in staff requirements memoranda dated June 26, 1990, and July 21, 1993. It is the responsibility of the design certification applicant to demonstrate that all such equipment can survive the radiation environment following a severe accident. As discussed in Enclosure 1, the staff believes that the design-basis source terms described above may not represent reasonable surrogates for the environmental conditions (such as pressure, temperature, radiation, and humidity) in a severe accident. For a severe accident, the design certification applicant should include the contribution of fission products that may be released ex-vessel and late in-vessel. For those reactors that have a reliable ex-vessel flooding mechanism for the core debris, this contribution may be significantly reduced by aerosol scrubbing.

In radiological assessments of equipment survivability as a result of a severe reactor accident, the staff will require that the equipment and features needed for severe accident prevention, mitigation, and post-accident sampling be designed to provide a reasonable level of confidence that they will operate in a severe accident environment. This environment would include the ex-vessel release, with proper credit for design features to mitigate that release, and the late in-vessel release in addition to the releases for a DBA (Issue 3).

Finally, the probabilistic safety assessment (PSA) for each advanced reactor includes an evaluation of the dose consequences associated with a variety of core melt scenarios and containment failure modes. These calculations were performed with the current understanding of source term behavior. While these assessments generally contain some discussion of uncertainties in fission-product behavior, they do not include any explicit provision for conservatism. As is generally the case in PRAs, these calculations are done on a realistic basis.

Design-Specific Reviews and Staff Positions

The staff has completed its review of the General Electric (GE) advanced boiling water reactor (ABWR) and the ABB-CE System 80+ designs and is currently reviewing the Westinghouse AP600 reactor design.

ABB-CE initially proposed to use the TID-14844 source terms. However, it subsequently submitted to the staff an amendment to its standard safety analysis report (SSAR) incorporating the revised accident source terms as described in draft NUREG-1465.

The System 80+ design provided the same mitigating features for the radiological consequences resulting from a DBA as those provided for the currently operating PWRs (containment spray, engineered safety features (ESF) filtration, etc.), except that safety-grade charcoal adsorbers are not provided. ABB-CE demonstrated in its SSAR that the System 80+ design will meet the dose reference values given in 10 CFR Part 100 and GDC 19 using the source terms provided in draft NUREG-1465 in conjunction with the use of atmospheric relative concentrations (χ/Qs) provided in its Tier 1 Design Document. The staff reviewed the System 80+ design using the staff's positions discussed in this Commission paper and concluded that the design will meet the above relevant dose reference values.

The final NUREG-1465 will not alter the staff's conclusion in the System 80+ FSER in regard to its acceptance of the System 80+ design for meeting the dose reference values given in 10 CFR Part 100 and the control room operator dose limit specified in GDC 19. As shown in Table 3 of Attachment 2, the fission-product release fractions given in final NUREG-1465 are the same (for noble gases, iodine, and cesium) or lower than those given in draft NUREG-1465 (for non-volatile elements). For fission-product chemical forms, the percent of core iodine inventory in organic form is reduced to 0.15 in final NUREG-1465 from 0.25 percent used in the CE System 80+ design review. There were no changes made in fission-product release timing for PWR in final NUREG-1465 from that in draft NUREG-1465.

For the AP600 design, Westinghouse proposed the same accident source terms as those proposed by EPRI for passive plants. However, the staff will use the revised source terms (fission-product release fractions and chemical forms) as described in final NUREG-1465 for the AP600 design review. The staff is currently evaluating the fission-product release timing proposed by Westinghouse. The staff realizes that fission-product release timing is dependent upon reactor design and reactor design basis accident sequences chosen for the source term application. The fission-product release timing given in final NUREG-1465 is based on the current LWR designs. This issue is discussed in more detail in Item 6 of Attachment 1.

Unlike the current generation of operating PWRs, the AP600 design does not include either an ESF filtration (e.g., charcoal adsorbers) or a containment spray system. In SECY-93-087 (Item III.F, Radionuclide Attenuation), the staff informed the Commission that it is still evaluating the need for a containment spray system for the passive plant designs. The staff will review the AP600 radiological consequence assessment to determine if the dose reference values in 10 CFR Part 100 and the control room operator dose criteria of General Design Criterion 19 of Appendix A to 10 CFR Part 50 can be met without a containment spray system. That determination will involve evaluation of aerosol removal rates by natural deposition mechanisms, containment leak rates, and post-accident pH control of the water in the containment. This is addressed in more detail as Issue 9 in Attachment 1.

NUREG-1242, "NRC Review of Electric Power Research Institute's Advanced Light Water Reactor Utility Requirements Document," resolves the major source term related issues in principle except iodine chemical species for BWR, amount of gap activity, magnitude of low- and non-volatile fission products released, and fission-product release timing. As discussed in NUREG-1242, it was expected that these differences would be resolved by the issuance of final NUREG-1465. The numbers presented in Table 3 of Attachment 2 show that the positions being proposed for final NUREG-1465 is much closer to the EPRI positions. Therefore, the staff believes that issuance of final NUREG-1465 will resolve these differences.

GE demonstrated in its SSAR that the ABWR design will meet the offsite dose reference values given in 10 CFR Part 100 using the current TID-14844 source terms in conjunction with the use of atmospheric relative concentrations (χ/Qs) provided in its Tier 1 Design Document. The staff reviewed the ABWR design, performed an independent analysis of the radiological consequences resulting from a postulated DBA, and concluded in the FSER that the ABWR design will meet the dose reference values given in 10 CFR Part 100.

In its review, the staff accepted the ABWR design without a main steamline leakage control system (LCS) that is designed to process main steamline leakage through main steamline isolation valves following a DBA. The staff also allowed credit for removal of radioactive iodine in main steamline leakage by holdup and plateout in the main steamline and condenser following a DBA.

As discussed in Attachment 1, these departures from the guidance of Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," and the principle of not allowing credit for non-seismically qualified systems are based on the conclusion that these systems are expected to retain sufficient structural integrity to remain effective for fission-product holdup and plateout throughout a DBA.

The staff will accept the passive BWR plant design without a LCS and allow an appropriate credit for iodine removal in the main steamline and condenser following a DBA (Issue 4).

In its SBWR design, GE initially proposed the same accident source terms as those published in draft NUREG-1465, but amended its SSAR to reflect the EPRI accident source terms. The SBWR design provides non-safety-grade containment spray systems without an ESF filtration system. This reduction of safety margin for mitigating radiological consequences may be compensated for by the secondary containment ("safety envelope") design. The safety envelope is a reinforced-concrete structure within the reactor building that forms an envelope completely surrounding the primary containment. It is designed to be capable of periodic testing to ensure its intended performance. The design also relies on lower core power density (lower reactor power) and slower, delayed releases of fission products into the containment.

The staff is not reviewing the source term related technical and licensing issues for the SBWR design at the present time. However, when the staff resumes its review of the SBWR design, the staff will allow appropriate credit for the SBWR safety envelope based on fission-product holdup and decay within this envelope if (1) the vendor specifies that the secondary containment leakage and mixing performance be consistent with the values used by the staff in its radiological assessment and (2) the COL combined license applicant incorporates the secondary containment leakage value specified by the vendor into the plant-specific technical specifications (Issue 5).

The staff has completed its review of source term-related issues in the EPRI requirements documents for evolutionary and passive plant designs and in the System 80+ design. It is currently reviewing the AP600 design. The staff positions presented in this paper will form a basis for closing all source term-related open issues associated with the passive ALWR designs. The staff does not expect that the publication of NUREG-1465 in final form will materially change its evaluation or conclusions pertaining to these designs.

<u>Margins of Conservatism</u>

The use of the revised source term information is an important departure from previous practice. The new approach will employ a physically-based source term based on substantial research and experience gained over two decades. The TID-14844 non-mechanistic methodology with the staff's application of conservative assumptions was intended to ensure that future plants would provide sufficient safety margins even with the recognized uncertainties associated with accident sequences and equipment reliability. While the TID-14844 source term has served its intended purpose, research started prior to the Three Mile Island Accident provides substantial information on plant behavior under accident conditions. The results of this research are incorporated in draft NUREG-1465.

The staff believes that accident research insights and information provided in draft NUREG-1465 and this paper provide a sound, contemporary basis for reactor accident mitigation system designs. While the new information may lead to relaxation in some aspects of the design, it also provides safety benefits by removing unrealistically stringent testing requirements. For example, the TID-14844 source term specifies the instantaneous release of fission-products into the primary containment while reactor accident research insights indicate that releases occur over a period of hours. This change, when used as a partial basis for less stringent closure requirements for fast-acting isolation valves, could result in enhanced reliability and integrity of these valves. Another example includes giving credit for fission-product plateout in the ABWR main steamlines and condenser when no credit has previously been given. This is based on our improved understanding of iodine transport mechanisms and iodine behavior from the revised accident source term research insights. The credit for fission-product plateout in the main steamlines and condenser could also result in enhanced reliability and integrity of the main steam isolation valves, by allowing the staff to use the higher leak rates and less stringent closure requirements.

The NUREG-1465 accident research insights strongly link the revised iodine chemical forms discussed in this paper with the ability to control the pH of the water in the containment above 7.0 during the course of an accident. If the proposed plant design features provide for adequate pH control based on a consideration of the various sources of acid additions that could occur during the course of an accident, the staff will not need to consider re-evolution of elemental iodine. Accordingly, there may not be a need to consider accident mitigation features such as containment sprays and charcoal adsorbers. If, however, a particular design does not provide adequate pH control, the staff will require the addition of non-safety grade charcoal adsorbers or a containment spray system.

With considerations such as those described above, the staff believes that the use of the new source terms provides a reasonable basis for establishing acceptable designs.

SUMMARY:

The methodology for implementing the revised source terms gives credit for improved understanding of fission product release timing resulting from accident research as well as more realistic estimates of reductions in the magnitude of potential fission-product releases from containment. These changes will result in reduced requirements for mitigation systems. The staff recognizes this effect and believes that it is justified on the basis of the improved understanding of fission-product behavior. The revised source terms should provide the staff with adequate technical bases for ensuring that safety margin is maintained.

The staff presents its positions on 12 source term-related issues in Attachment 1. These positions provided the basis for the staff to close out all source term related open issues in the staff's SERs for the EPRI Requirements Document for passive plant designs and for the ABB-CE System 80+ design and to proceed with its review of passive LWR plant designs. The final NUREG-1465 will not alter the staff's conclusion in the ABB-CE System 80+ SER.

COORDINATION:

The Office of the General Counsel has reviewed this paper and has no legal objection. The ACRS Full Committee was briefed on February 10, and March 10, 1994. The ACRS provided its comments in a letter to the Chairman dated March 15, 1994. The staff has considered the ACRS comments in developing the staff positions contained in this paper.

CONCLUSIONS:

The staff believes that the revised source term as given in NUREG-1465 is appropriate for use in the licensing review of evolutionary and passive LWR designs. The staff intends to continue application of its methodology and the staff positions described in this position paper to the on-going ALWR certification reviews. The staff intends to place this information paper into the PDR.

James M. Taylor
Executive Director
for Operations

Attachments:
1. Source Term-Related Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water-Reactor Designs
2. Proposed Reactor Accident Source Terms

DISTRIBUTION:
Commissioners
OGC
OCAA
OIG
OPA
OCA
REGIONS
EDO
ACRS
SECY

SOURCE TERM-RELATED POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT-WATER-REACTOR DESIGNS

ISSUE 1: Selective Use of Accident Source Terms Given in Draft NUREG-1465 (for System 80+, AP600, and SBWR).

In SECY-90-016, the staff recommended to the Commission that it approve staff consideration of deviations from current methodology used to calculate 10 CFR Part 100 doses on a case-by-case basis using engineering judgment and updated information on source terms and equipment reliability for the evolutionary plant designs. The Commission approved this recommendation in a staff requirements memorandum of June 26, 1990. This section of the Commission paper describes such deviations to selectively use the fission-product release fractions described in NUREG-1465 for radiological consequence assessments of design-basis accidents for the evolutionary and passive light-water-reactor designs.

Present regulations (10 CFR Part 100) require that a fission-product release from the reactor core into the containment as a result of a reactor accident be postulated and that its radiological consequences at the exclusion area boundary and low population zone outer radius be evaluated assuming the "expected demonstrable leak rate" from the containment and the meteorological conditions pertinent to the reactor site, with the implicit assumption that the containment remains intact against the maximum credible accident. Footnote 1 to 10 CFR Part 100 states that the fission-product release to be assumed for Part 100 dose calculations should be "based on a major accident...that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products."

The current source terms are specified in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant for Pressurized Water Reactors." The source terms were derived from Technical Information Document (TID)-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." Although the staff considered the consequences of fission-product release into the containment to represent the bounding reactor accident in terms of credibility, it also evaluated other accident types to verify that these did not result in greater consequences. This practice led the staff eventually to develop and consider a group of accidents, referred to as "design-basis accidents (DBAs)." The DBAs routinely evaluated by the staff as part of the license review include (1) loss-of-coolant accident (LOCA), (2) fuel handling accident, (3) steam generator tube rupture accident (PWR), (4) main steamline break outside the containment, (5) control rod drop accident (BWR), (6) control rod ejection accident (PWR), (7) failure of small lines carrying primary coolant outside the containment, and (8) spent fuel cask drop accident. The staff used the calculated radiological consequences resulting from these DBAs to ensure that the distances to the exclusion area boundary and the outer radius of the low population zone for a nuclear power plant, in conjunction with the operation of the dose-mitigating engineered safety features (ESF) systems, are sufficient to provide reasonable assurance that the dose reference values in 10 CFR Part 100 will not be exceeded.

For environmental qualification of electrical equipment, 10 CFR 50.49 states that safety-related electrical equipment, certain non-safety-related electrical equipment, and certain post-accident monitoring equipment should remain functional during and following design-basis events. Design-basis events are further defined as conditions of normal operation, including anticipated operational occurrences, DBAs, external events, and natural phenomena that the plant must be designed and built to withstand.

In Regulatory Guide 1.89 (Revision 1), "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants" (June 1984), the staff assumes the TID-14844 source term for determining the radiation environment for qualification of electrical equipment important to safety and for calculating the time integrated radiation doses in the containment and at a point just above the containment sump.

For additional Three Mile Island (TMI)-related requirements, 10 CFR 50.34 refers to the TID-14844 source term (1) to perform radiation and shielding design review, (2) to assess post-accident sampling capability, (3) to assess the system design for leakage control and detection outside the containment, and (4) to evaluate control room habitability under accident conditions.

Draft NUREG-1465 lists five fission-product release phases of a severe LWR accident: (1) coolant activity release, (2) gap activity release, (3) early in-vessel release, (4) ex-vessel release, and (5) late in-vessel release. The coolant activity release phase begins with a postulated pipe rupture and ends when the first fuel rod has been estimated to fail. During this phase, the fission products released to the containment atmosphere are those associated with radioactive material in the reactor primary coolant during normal plant operation. Draft NUREG-1465 does not specify fission-product releases to the containment during this phase, since the amounts of the fission products released are small compared with later release phases. The gap activity release phase begins when fuel cladding failure commences and ends when the fuel starts to melt. This phase involves the release of that radioactive material that has collected in the gap between the fuel pellet and cladding.

The early in-vessel release phase begins when fuel and other materials melt and fall to the bottom of the reactor pressure vessel. During this phase, significant quantities of the volatile radionuclides in the core inventory, as well as some of the less volatile radionuclides, are estimated to be released into the containment. This release phase ends when the bottom head of the reactor pressure vessel fails, allowing molten core debris to fall onto the concrete below the reactor pressure vessel.

The ex-vessel release phase begins when the molten core debris exits the reactor pressure vessel and the core-concrete interaction takes place. This phase ends when the debris has cooled sufficiently so that significant quantities of fission products are no longer being released. During this phase, significant quantities of the volatile radionuclides not already released during the early in-vessel phase as well as lesser quantities of non-volatile radionuclides are released into the containment. Finally, the late in-vessel release phase commences at vessel breach and proceeds simultaneously with the ex-vessel release phase.

Tables 1 and 2 of Attachment 2 list the proposed fission-product release fractions from proposed final NUREG-1465 of core inventory into the containment, including their release timing and duration.

The staff proposes to use only the gap release and the early in-vessel release for design-basis calculations for the evolutionary and passive LWR designs. These source terms encompass a broad range of accident scenarios, including significant levels of core damage with the core remaining in the vessel. These would be the most severe scenarios from which the plant could be expected to return to a safe shutdown condition. Table 3 in Attachment 2 shows that the NUREG-1465 release fractions associated with these releases are generally comparable to those of TID-14844 for the noble gases and iodine, but include additional nuclides, such as cesium.

The staff considers the inclusion of the ex-vessel and late in-vessel source terms to be unduly conservative for DBA purposes. Such releases would only result from core damage accidents with vessel failure and core-concrete interactions. For evolutionary and passive LWRs, the estimated frequencies of such scenarios are low enough that they need not be considered credible for the purpose of meeting 10 CFR Part 100.

The staff has addressed Issue 1 by selectively using the source terms given in proposed final NUREG-1465 using only "Gap Release" and "Early In-Vessel Release" (excluding "Ex-Vessel Release" and "Late In-Vessel Release" associated with vessel failure, and core-concrete interaction) in evaluating radiological consequences for DBAs, in evaluating the DBA radiation environmental qualification of electrical and mechanical equipment important to safety, and in evaluating TMI-related requirements (for System 80+, AP600, and SBWR).

<u>ISSUE 2</u>: Iodine Chemical Form (for System 80+, AP600, and SBWR)

Regulatory Guides 1.3 and 1.4 specify that fission-product releases into the containment consist of 100 percent of the core inventory of noble gases and 50 percent of iodines (half of which are assumed to deposit on interior surfaces of the containment very rapidly). The iodine chemical form is specified to be predominantly elemental iodine (91 percent), with 5 percent assumed to be particulate iodine and the remaining 4 percent assumed to be in the organic form. The 1 percent of "solid" fission products included in TID-14844 was dropped from consideration in Regulatory Guides 1.3 and 1.4.

In draft NUREG-1465, the staff concluded that iodine entering the containment from the reactor core is composed of at least 95 percent cesium iodide (CsI) with no more than 5 percent of iodine (I) and hydrogen iodide (HI). Once within the containment, highly soluble cesium iodide will readily dissolve in water pools forming iodide (I^-) in solution and deposit onto the interior surfaces. The staff also stated in draft NUREG-1465 that the radiation-induced conversion of iodide (I^-) in water into elemental iodine (I_2) is strongly dependent on the pH. The staff indicated that without pH control, large fractions of iodine dissolved in water pools in ionic form will be converted to elemental iodine and will be released into the containment atmosphere if the pH is less than 7. On the other hand, if the pH is maintained above 7, very little (less than 1 percent) of the dissolved iodine will be converted to elemental iodine. The EPRI requirements documents for evolutionary and passive plant designs and all evolutionary and passive LWR vendors require that the pH of the water in the containment be maintained above 7 (alkaline state) for the entire duration of the accident to minimize the formation of elemental iodine in the containment water in order to reduce the subsequent release of iodine into the containment atmosphere. The staff agrees with this requirement.

In draft NUREG-1465, the staff did not address the formation of organic iodide in the containment following an accident. However, organic iodide can be produced primarily by the reaction of elemental iodine with organic materials present in the containment. EPRI proposed the use of 0.15 percent organic iodide based on 3 percent conversion of elemental iodine to organic iodine (i.e., 3 percent of 5 percent is 0.15 percent). Based on studies by NRC contractors documented in NUREG/CR-4461, -5232, and -5950, the staff estimates that no more than 3 percent of the airborne elemental iodine will be converted into organic species. This amount of organic iodide would thus correspond to about 0.15 percent of the core iodine inventory (i.e., 3 percent conversion of 5 percent elemental iodine is 0.15 percent). Final NUREG-1465 will address this issue.

The final NUREG-1465 will not alter the staff's conclusion in the System 80+ FSER in regard to its acceptance of the System 80+ design for meeting the dose reference values given in 10 CFR Part 100 and the control room operator dose limit specified in GDC 19. As shown in Table 3 of Attachment 2, the fission-product release fractions given in final NUREG-1465 are the same (for noble gases, iodine, and cesium) or lower than those given in draft NUREG-1465 (for non-volatile elements). For fission-product chemical forms, the percent of core iodine inventory in organic form is reduced to 0.15 in final NUREG-1465 from 0.25 percent used in CE System 80+ design review. There were no changes made in fission-product release timing for PWR in final NUREG-1465 from that in draft NUREG-1465.

The staff has addressed Issue 2 by partitioning radioiodine chemical forms as follows: 95 percent cesium iodide (CsI, or I^- in the aqueous phase); 4.85 *percent elemental iodine (I_2), and 0.15 percent as organic iodide (e.g., CH_3I). The latter two values will be used in preference to a single value of 5 percent I_2 recommended in draft NUREG-1465. These values for the iodine chemical form will be used in the AP600 and SBWR reviews.*

The evolutionary and passive LWR vendors have stated that they will use the staff's proposed values until final NUREG-1465 is published. The staff will use these chemical forms in evaluating the containment spray removal efficiencies, BWR suppression pool decontamination factors, passive deposition of fission products in the containment, control room habitability, and engineered safety features (ESF) filtration systems in its evolutionary and passive LWR design reviews. The application of these chemical forms in each evolutionary and passive LWR design review will be discussed in the staff's forthcoming safety evaluation reports.

ISSUE 3: Equipment Survivability for Design Features Needed for Severe Accident Mitigation and Containment Integrity (for System 80+, AP600, and SBWR).

As the staff concluded in SECY-90-016, the equipment and features provided for preventing and mitigating severe reactor accidents need not be subject to the rigorous 10 CFR 50.49 environmental qualification requirements. However, mitigation features and post-accident sampling systems for severe reactor accidents must be designed to provide a reasonable level of confidence that they will survive the severe accident environment for which they are intended and be able to operate over the timespan for which they are needed.

For the purpose of radiological assessments, the staff proposes to define the DBA radiation environment as that resulting from fission-product releases from coolant activity release, gap release, and in-vessel release. The staff is also defining the severe reactor accident radiation environment as that resulting from the above fission-product releases plus ex-vessel release and late-in-vessel release that are associated with vessel failure and core-concrete interaction.

If safety-related equipment is relied on to cope with severe accident situations, there should also be a reasonable level of confidence that this equipment will survive the severe accident environment, including radiation, for the period that it is needed to perform its intended function. The staff will evaluate the evolutionary and passive LWR vendors' identification of such equipment and features, their intended functions, location, arrangement, shielding, and the period during which they are needed to perform on a case-by-case basis.

For issues related to severe accidents, such as the survivability of equipment needed for severe accident mitigation and post-accident sampling, the staff believes that the ex-vessel and late-in-vessel releases can represent an important part of the source terms. However, for those evolutionary and passive designs that include reliable ex-vessel flooding mechanisms for the core debris, the ex-vessel contribution may be significantly reduced by aerosol scrubbing by a water pool overlying core debris.

The staff has addressed this part of Issue 3 by requiring that equipment and features needed for severe accident prevention, mitigation, and post-accident sampling be designed to provide a reasonable level of confidence that they will operate in a severe accident environment. This environment would include the ex-vessel release, with proper credit for features to mitigate that release, and the late in-vessel release (for CE System 80+, AP600, and SBWR).

ISSUE 4: Radioactive Iodine Deposition on BWR Main Steamlines and Condensers (for ABWR and SBWR).

The main steamlines in BWR plants, including the ABWR and SBWR, contain dual quick-closing main steam isolation valves (MSIVs). These valves function to isolate the reactor system in the event of a break in a steamline outside the primary containment, a design-basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leaktight barrier, it is recognized that some of the valves will leak. Operating experience has indicated that the leaktightness of MSIVs has occasionally degraded and the leakage limits in the technical specifications have not always been maintained.

When calculating offsite consequences of potential accidents, the staff conservatively assumed that radioactive material passing through the MSIV at its technical specifications leakage limit is released directly into the environment. No credit was given for the integrity and leaktightness of the main steam piping and condenser to provide holdup and plateout of fission products.

Because of recurring problems with excessive leakage of MSIVs, the staff recommended installation of a supplemental leakage control system (LCS) in Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," to ensure that the isolation function of the MSIVs is in accord with the specified limits. Most of the operating BWRs have an LCS.

In response to the MSIV leakage concerns, the BWR Owners Group commissioned a program of studies to determine the causes of high leak rates and the means to eliminate them. The results of these studies were submitted to the staff in General Electric (GE) proprietary reports, NEDO-31643P (November 1988) and NEDO-31858P (February 1991), both entitled "Increasing Main Steam Isolation Valve Leakage Rate Limits and Elimination of Leakage Control Systems."

EPRI and GE referenced these reports in the requirements documents and the ABWR and SBWR standard safety analysis reports (SSARs), respectively, as the basis for not providing an LCS and for requesting a substantially higher MSIV total leak rate [1.1×10^{-3} m^3/sec (140 ft^3/hr) rather than 3.5×10^{-4} m^3/sec (45 ft^3/hr)] than the typically allowed limit for operating BWRs. In addition, GE claimed in the ABWR and SBWR SSARs, that elemental and particulate iodine will deposit on the main steamlines (MSLs) and condenser, thus mitigating the radiological consequences of an accident. The basis for GE's claim was contained in a model developed for this purpose and documented in the referenced NEDO reports.

Following a LOCA, three potential release pathways exist for main steam leakage through the MSIVs:

(1) main steam drainlines to the condenser with delayed release to the environment through the low-pressure turbine seals

(2) turbine bypass lines to the condenser with delayed release to the environment through the low-pressure turbine seals

(3) main steamline turbine stop and control valves through the high-pressure turbine seals to the environment bypassing the condenser

The consequences of leakage from pathways 1 and 2 will be essentially the same because the condenser can be used to process MSIV leakage. The iodine removal efficiency of the condenser will vary depending on the inlet location of the bypass or drainline piping, but in either case, iodine will be removed. For pathway 3, MSIV leakage through the closed turbine stop and control valves will not be processed via the condenser. For this case, the high-pressure turbine (having a large internal surface area associated with the turbine blades) will remove iodine.

The staff believes that as long as either the turbine bypass or drainline leakage pathway is available, MSIV leakage through the closed turbine stop and control valves (pathway 3) will be negligible. Essentially all of the releases will be through the main condenser because there will be no differential pressure in the MSL downstream of the MSIVs following the closure of the valves.

Furthermore, MSIV leakage through pathway 3, if any, will have been subjected to the same iodine-removal processes in the MSLs (up to turbine stop valves) as those in the other pathways. The leakage will be further subjected to iodine removal by deposition in the high-pressure turbine internal surfaces. Removal by the main condenser is not applicable in pathway 3.

Basic principles of chemistry and physics predict that gaseous iodine and airborne iodine particulate material will deposit on surfaces. Several laboratory and in-plant studies have demonstrated that gaseous iodine will be deposited by chemical adsorption, and particulate iodine will be deposited through a combination of sedimentation, molecular diffusion, turbulent diffusion, and impaction. The transport of gaseous iodine in elemental and particulate forms also has been studied for many years, and several groups have proposed different models to describe the observed phenomena.

The staff has developed a model that treats the MSIV leakage pathway as a sequence of small segments for which instantaneous and homogeneous mixing is conservatively assumed; the mixing computed for each segment is passed along as input to the next segment. The number of segments depends on the parameters of the line and flow rate and can be as many as 100,000 for a long, large-diameter pipe and a low flow rate. Each line segment is divided into five compartments that represent the concentrations of the three airborne iodine species, the surface that contains iodine available for resuspension, and surface iodine that has reacted and is fixed on the surface.

The staff's model considers three iodine species: elemental, particulate, and organic. A fourth species, hypoiodous acid, was considered for the purpose of the staff's model to be a form of elemental iodine. All iodine in the segment undergoes radioactive decay. The resulting concentration from each segment of the deposition compartment serves as the input to the next segment. The technical references indicate that particulate and elemental iodine would be expected to deposit on surfaces with rates of deposition varying with temperature, pressure, gas composition, surface material, and particulate size. Therefore, the staff believes that an appropriate credit for the removal of iodine in the MSLs and main condensers should be provided in the radiological consequence assessment following a design-basis accident.

In the review of the ABWR design, the staff accepted GE's proposed elimination of the LCS and allowed a higher MSIV leakage rate providing an appropriate credit for the removal of iodine in the MSLs and condenser. The staff gave a detailed technical evaluation in the ABWR draft final safety evaluation report (DFSER). Even though the staff used the TID-14844 source terms (mostly elemental iodine) for the ABWR design, as proposed by GE, the staff's model is also applicable for the NUREG-1465 source terms (mostly iodine in particulate forms as aerosol).

The GE model, as well as the one used by the staff, is based on time-dependent temperature adsorption phenomena with instantaneous and perfect mixing in a given volume. Both models use the same MSIV leakage pathways. They differ, however, in the treatment of buildup of iodine in the MSLs and condenser. GE assumed steady-state iodine in equilibrium in a large volume, while the staff assumed transient buildup of iodine in a finite number of small volumes. The staff does not consider these differences to be significant and finds that the resulting radiological consequences (offsite doses) are in good agreement. It therefore finds the GE model to be acceptable.

Sections III(c) and VI of Appendix A to 10 CFR Part 100 require that structures, systems, and components necessary to ensure the capability to mitigate the radiological consequences of accidents that could result in exposures comparable to the dose guidelines of Part 100 be designed to remain functional during and after a safe shutdown earthquake (SSE). Thus, the MSL, portions of its associated piping, and the main condenser are required to remain functional if credit is taken for deposition of iodine and if the SSE occurs. Consequently, the staff's past practice has been to classify these components as safety related and seismic Category I. In addition, Appendix A to 10 CFR Part 100 requires that the engineering method used to ensure that the safety functions are maintained during and after an SSE involves the use of either a suitable dynamic analysis or a suitable qualification test.

In Section II.E, "Classification of Main Steamlines in Boiling Water Reactors," of SECY-93-087, the staff stated that the classification of the MSL in the turbine building as non-seismic Category I is needed for consistency with the classification of the turbine building. On this basis, the quality and safety requirements imposed on the MSL from the outmost isolation valve up to, but not including, the turbine stop valve are equivalent to the staff guidelines in Appendix A to SRP Section 3.2.2, "System Quality Group Classification," and Regulatory Guide 1.29, "Seismic Design Classification."

In SECY-93-087 (Item II.E), the staff proposed for Commission approval, among other things, that:

(1) neither the main steam drain and bypass lines from the first valve up to the condenser inlet nor the piping between the turbine stop valve and the turbine inlet be classified as safety related or as seismic Category 1; rather, these lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions;

(2) the turbine stop, control, and bypass valves and the MSLs from the turbine control valves to the turbine meet all of the quality group and quality assurance guidelines specified in Appendix A to SRP Section 3.2.2; and

(3) seismic analyses be performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after the SSE.

The staff concluded in SECY-93-087 that the above-described approach provides reasonable assurance that the main steam piping from the outmost isolation valve up to the turbine stop valve, the main steam drain and bypass lines up to the condenser, and the main condenser will retain their pressure and structural integrity during and following SSE. The Commission approved the above staff proposals in its staff requirements memorandum of July 21, 1993.

In its review of the ABWR design, the staff concluded in the FSER that the ABWR design provides reasonable assurance that the main steam piping from the outmost isolation valve up to the turbine stop valve, the MSIV leakage pathway (i.e., the drainline or bypass line) up to the condenser, and the main condenser will remain structurally intact and leaktight, so that they can act as a holdup volume for fission products during and following an SSE.

The staff also determined that the ABWR design meets the requirements of 10 CFR Part 100 because the structures, systems, and components described above are designed to remain functional during and following an SSE.

For the purpose of providing a credit for iodine holdup and plateout, the staff's model requires that the main steam piping (including its associated piping to the condenser) and the condenser remain structurally intact following an SSE, so they can act as a holdup volume for fission products. By the term "structurally intact," the staff assumes the steamline will retain sufficient structural integrity to transport the MSIV bypass leakage at its relatively low flow rate throughout the steamlines and condenser. The staff considers, in its radiological consequence assessment, that the condenser is open to the atmosphere via leakage through the low-pressure turbine seals. Thus, it is only necessary to ensure that gross structural failure of the condenser will not occur.

The staff has addressed Issue 4 by accepting the passive BWR plant designs without an LCS and allowing an appropriate credit for iodine removal in the MSL and condenser following a design-basis accident (for ABWR and SBWR).

The robustness of the leak pathways and the amounts of iodine removed are plant design specific and, therefore, will be determined on a case-by-case basis.

<u>ISSUE 5</u>: Fission-Product Holdup in the Secondary Containment (for SBWR).

The structure or structures that completely surround the reactor primary containment and that are held at a negative pressure can be classified as a secondary building for the purpose of fission-product control and, therefore, for the mitigation of radiological consequences. The EPRI requirements documents for evolutionary and passive plans designs require no holdup and retention of fission products in the PWR secondary building. Westinghouse has not claimed any credit for fission-product holdup in the AP600 auxiliary (secondary) building.

GE proposed a safety envelope design for the SBWR plant. This envelope is a reinforced concrete structure (secondary containment) within the reactor building that forms an envelope completely surrounding the primary containment (except the basement drywell head and drywell top slab). During normal operation, the safety envelope is maintained at a slightly negative pressure relative to the atmosphere by the reactor building heating, ventilation, and air conditioning system. The safety envelope is designed to be automatically isolated on detection of, among other things, a high radiation environment. Its design leakage rate is not to exceed 25 percent of the safety envelope free volume per day at a differential pressure of 6 mm (0.25 inch) of water.

GE stated in the SBWR standard safety analysis report that the safety envelope is designed to be capable of periodic testing to ensure its intended performance and that testing and inspection of the integrity of the safety envelope, including its leakage rate, will be performed in accordance with the specific plant technical specifications. To provide a credit for fission-product holdup, the staff will require periodic leak rate testing similar to that performed for the primary containment. For safety envelope air mixing, GE proposed 50 percent on the basis of one-dimensional modeling of the KEMA computer code using simplified compartment volumes. GE informed the staff that it will follow this simplified modeling with a three-dimensional modeling effort using a computational fluid dynamic computer code to better define the capability of the SBWR safety envelope to hold up radioactive material and mitigate any potential leakage. GE further informed the staff that this effort will be completed in 1994.

The staff has addressed Issue 5 by allowing appropriate fission-product holdup (for decay) credit without requiring that a negative pressure be maintained in the SBWR secondary containment (building) if (1) the vendor specifies that the secondary containment leakage and mixing performance be consistent with the values used by the staff in its radiological assessment and (2) the COL combined license applicant incorporates the secondary containment leakage value specified by the vendor into the plant-specific technical specifications (for SBWR).

ISSUE 6: Fission-Product Release Timing (for System 80+, AP600, and SBWR)

In SECY-90-309, "Impacts of Source Term Timing on NRC Regulatory Positions," the staff informed the Commission of its plans to change regulatory positions for both existing and future advanced reactor designs that result from more realistic treatment of fission-product release timing. In Regulatory Guides 1.3 and 1.4, the instantaneous release of fission products into the containment on receipt of a DBA signal from maximum full-power operation of the core is postulated and fission products released are assumed to be immediately available for leakage from the containment atmosphere to the environment.

The staff evaluated a number of items in SECY-90-309 to determine if more realistic treatment of fission-product releases would lead to less demanding regulatory requirements. The staff concluded that relaxation of requirements was justified for (1) the containment purge/vent isolation valve closure time, (2) the main steam isolation valve closure time, and (3) the emergency diesel generator start time. It also concluded that relaxing the timing requirements for each of these items would affect more than radiological consequences and, therefore, a plant-specific safety analysis would be required for each proposed relaxation.

In draft NUREG-1465, the staff indicated that fission-product gap activity release was estimated for a large-break LOCA to commence no earlier than 10 to 30 seconds for PWRs and 30 seconds for BWRs. It also indicated that fission-product early in-vessel release was estimated to commence no earlier than 0.5 and 1.0 hour for PWRs and BWRs, respectively. In contrast to the instantaneous releases that were postulated in Regulatory Guides 1.3 and 1.4, analyses of severe accident sequences have shown that fission-product releases can be generally categorized in terms of phenomenological phases associated with fuel failure, melting, and relocation despite differences in reactor design and accident sequence.

The single values for release timing and durations shown in draft NUREG-1465 have been chosen on the basis of simplicity, consistency, and conservatism. An accurate determination of the release timing and durations will depend not only on the reactor design but also on the selected accident sequences. The staff will review an evolutionary or passive reactor vendor's design and accident sequence as a basis for fission-product release timing to determine if its values are different from those provided in draft NUREG-1465. For its reviews of evolutionary and passive LWR designs, the staff will use the fission-product release timing values for the radiological consequence assessments of (1) DBAs, (2) the operation of control room habitability systems, (3) the operation of ESF filtration systems, (4) containment and MSIV closure, (5) containment purge/vent isolation valve closure, and (6) emergency diesel generator start time.

In draft NUREG-1465, the staff stated that the coolant activity release starts at the beginning of the accident, the gap activity release at no earlier than 10 minutes into the accident (assuming approval of leak-before-break methodology by the staff), and the early in-vessel release no earlier than 30 and 60 minutes into the accident for PWRs and BWRs, respectively. This timing is based on a large-break LOCA of current LWR design. The staff realizes that fission-product release timing is dependent upon reactor type, design, and the bounding reactor design basis accident sequences chosen for the source term application.

Therefore, the staff will evaluate each proposed ALWR design on a case-specific basis to determine appropriate fission-product release timing. The staff will discuss the application of fission-product release timing in each evolutionary and passive LWR design review in its forthcoming safety evaluation reports.

ISSUE 7: **Aerosol Deposition in Primary Containment (for AP600 and SBWR)**

The principal means of removing the airborne fission products from the containment atmosphere in the LWR design traditionally included use of active containment atmosphere cleanup systems such as containment spray, ESF filtration, and pressure-suppression pool scrubbing. In the TID-14844 source terms, the staff assumed that 50 percent of radioactive iodine (91 percent in the form of elemental iodine) released into the containment was immediately removed by plateout (diffusion mechanism). No fission products in aerosol form were assumed to be present. In the passive LWRs, an active containment atmosphere cleanup system is not provided. Reliance is placed solely on such natural aerosol removal processes in the containment as holdup (for decay), sedimentation (for settling), diffusion (for plateout), and leakage (for depletion).

In response to the staff's request for additional information, EPRI stated that because a containment spray system is not used for the passive plant, the means by which aerosol is removed will be dominated by such natural removal effects as diffusion and sedimentation.

The most complete mechanistic treatment of aerosol behavior in the reactor containment is found in CONTAIN, a computer code developed at Sandia National Laboratories under NRC sponsorship for the analysis of containment aerosol behavior under severe accident conditions. The other computer code, NAUA, which is very similar to the aerosol portion of CONTAIN, was developed at Kernforschungszentrum, Karlsruhe, Federal Republic of Germany, and was used for aerosol treatment in the NRC Source Term Code Package. There are a number of other aerosol behavior computer codes, but these two codes are the most widely used and accepted throughout the international nuclear safety community. Either code is acceptable to the staff for the evaluation of aerosol fission-product behavior in the primary containment following a severe reactor accident. However, most of these computer codes use lumped parameter methods, which assume that the aerosols are well mixed in the containment. When the condition of mixing cannot be established, the calculated results using the above computer codes may not be valid.

The staff will consider two natural processes for removing aerosols from the containment atmosphere over the entire period of an accident (30 days): (1) sedimentation mechanism of gravitational settling, including aerosol agglomeration, and (2) diffusion mechanisms of diffusiophoresis and thermophoresis. Diffusiophoresis is the diffusion of particles into interior heat sink surfaces, such as the containment walls on which steam is condensing. Thermophoresis is a similar effect due to a temperature gradient near the surface.

In consideration of these two natural processes for removing aerosols from the containment atmosphere, the staff will evaluate the thermal hydraulic conditions and production of nonradioactive aerosols since they can strongly influence the behavior of radioactive aerosols in containment following a DBA. The thermal hydraulic conditions include the containment pressure, relative humidity, and steam condensation and heat removal rates of the containment structure. These thermal hydraulic parameters, as well as the amounts of non-radioactive aerosol produced, differ with the specific reactor accident sequences and with the accident mitigation features provided (e.g., isolation condensers and primary containment cooling system for the SBWR design, containment shell cooling for the AP600 design, and containment spray for the CE System 80+ design).

In a paper on the source term for passive plants supplemented by a draft report titled "Passive ALWR Containment Natural Aerosol Removal" (April 1993), EPRI has provided radioactive aerosol removal rates based on sedimentation and diffusion of aerosols in the containment atmosphere using the EPRI version of the NAUA code (NAUAHYGROS).

The staff agrees with EPRI on the physical processes associated with natural aerosol removal that could be taken into account in establishing the airborne fission-product concentrations in the containment atmosphere. Therefore, the staff accepts the aerosol removal mechanisms proposed by EPRI in the requirements documents for evolutionary and passive plant designs. However, the containment aerosol removal rates are plant design specific and will vary, depending on, but not limited to, the containment geometry, containment size, surface area, steam quality, and containment cooling mechanisms.

Because the AP600 and SBWR designs have been submitted and are under review, the staff does not intend to develop and promulgate specific guidance relative to the impact of thermal hydraulic conditions and production of nonradioactive aerosols on the behavior of radioactive aerosols in containment at this time. The staff will, however, communicate on these matters with Westinghouse and General Electric via Requests for Additional Information (RAI) or similar licensing communication vehicles during the development of the draft safety evaluation reports. In the discussion which follows, the staff has summarized the current status of the three ALWR design reviews which are based on the revised accident source terms.

General Electric proposed a group of reference accident sequences for the SBWR design that lead to core damage but terminate with the reactor pressure vessel still intact to determine the thermal hydraulic conditions in the containment following the design basis accident. GE, in turn, used these reference accident sequences to generate the thermal hydraulic conditions using an SBWR version of the MAAP code. The staff is currently reviewing the GE proposal and will either select its own representative group of accident sequences or, after appropriate review, use the GE-selected accident sequence for the staff's independent determination of the thermal hydraulic conditions and nonradioactive aerosol generation. The staff will use this information along with the following parameters for evaluating the fission-product behavior and determining the aerosol removal rates in the SBWR containment following a design-basis accident:

1) containment geometry
2) aerosol characteristics
3) aerosol removal by isolation condensers and primary containment cooling systems

Westinghouse has neither provided design basis accident sequences nor the thermal hydraulic conditions in the containment for the AP600 design at this time. The staff is currently addressing this matter with Westinghouse via RAIs.

The CE System 80+ evolutionary plant design includes a safety-grade containment spray system; therefore, the thermal hydraulic conditions and the amounts of nonradioactive aerosol are less significant for the purpose for determining radioactive aerosol behavior and its removal rates in the containment. Nevertheless, the staff used in its evaluation, a heat removal rate of 100 moles per second (typical value from NUREG-1150) and 350 kilograms of nonradioactive aerosol (a value given in draft NUREG-1465 as information) in its evaluation of the System 80+ design.

ISSUE 8: Aerosol Removal by BWR Suppression Pool (for ABWR and SBWR)

The BWR suppression pools are designed primarily as containment pressure and temperature suppression mechanisms for reactor pressure vessel blowdown. However, they can also serve as a medium for scrubbing radioactive fission products except noble gases and iodine in organic forms. The scrubbing (attenuation) of fission products in suppression pools is usually expressed as a "decontamination factor" (DF), which is defined as the ratio of the radioactive material injected into the pool divided by the airborne radioactive material that leaves the surface of the pool water.

Regulatory Guide 1.3, issued in 1974, does not recommend credit for fission-product scrubbing by BWR suppression pools. However, the Reactor Safety Study (WASH-1400), issued in 1975, assumed a DF of 100 for subcooled suppression pools and 1.0 for steam-saturated pools. Since 1975, the staff and EPRI have developed detailed models for the analysis of radioactive aerosol scrubbing by the suppression pool. Accordingly, in 1988, the staff issued revised SRP Section 6.5.5, "Pressure Suppression Pool as a Fission Product Cleanup System," stating that ignoring the large volume of research data supporting pool credit would be an undue degree of conservatism. The revised SRP section allowed a maximum DF of 10 for particulate and elemental iodines for Mark II and Mark III containments and a DF of 5 for a Mark I containment. These DF values were based on the TID-14844 report, which stipulates an instantaneous release of fission products from the reactor coolant system to the containment atmosphere. In the revised SRP section, this was further subjected to the fraction of the drywell atmosphere bypassing the suppression pool by leaking through drywell penetrations to obtain the overall DF.

GE utilized the TID-14844 source terms for its ABWR design. The staff conservatively assumed in its radiological consequence assessment a DF of 2 by the ABWR suppression pool (equivalent to suppression pool steam bypass of 50 percent) for airborne radioactive iodine in elemental and particulate forms. The staff has accepted the drywell leakage values [0.005 m^2 (0.05 ft^2) of effective leakage area pathway] proposed by GE for the ABWR. The ABWR containment sprays in the drywell or wetwell, or both, also would reduce the effect of suppression pool bypass leakage on containment performance. Therefore, a DF of 2 used in the staff's analysis is conservative. For the ABWR design, the staff will review the plant-specific technical specifications, which require periodic inspections to confirm suppression pool depth, and surveillance tests to confirm drywell leaktightness on a case-by-case basis.

ISSUE 9: Fission-Product Removal by Containment Spray (For AP600)

General Design Criteria (GDC) 41, 42, and 43 of Appendix A to 10 CFR Part 50 require systems to control fission products to reduce the concentration that may be released to the environment. The containment spray system reduces containment pressure and temperature and removes airborne radioactive fission products in the containment atmosphere following a LOCA.

The EPRI requirements document for evolutionary plant designs requires a containment spray system. ABB-CE has provided an acceptable spray system for the System 80+ design. For its ABWR design, GE provided a safety-related spray system in the containment to reduce pressure and temperature following a LOCA, but has not claimed nor has the staff provided, any credit for the removal of airborne fission products for the purpose of radiological consequence assessments.

The EPRI requirements document for passive plant designs requires neither an active containment atmosphere cleanup system nor a containment spray system. This document states that fission-product control should be accomplished passively by reliance on natural deposition (plateout) and holdup (decay) and slower fission-product release timing. The Westinghouse AP600 design relies on plateout and holdup in the primary containment. In the SBWR design, GE relies on plateout in the reactor primary containment and holdup in the reactor primary containment and in the safety envelope. The safety envelope is a reinforced-concrete structure within the reactor building that forms an envelope completely surrounding the containment. It is designed to be periodically tested for leaktightness. The SBWR design includes a non-safety-grade spray system for both the upper drywell and suppression chamber. However, GE has not claimed, nor has the staff provided, fission-product removal credit for the non-safety-grade spray system in the SBWR design. The AP600 design does not provide a non-safety-grade containment spray.

The passive LWR vendors have submitted radiological consequence assessments to demonstrate that without containment spray, the passive LWR designs can meet the dose reference values of 10 CFR Part 100 and the control room operator dose limits in GDC 19 using the fission-product release timing based on the specific designs and selected accident sequences and using their best estimate of fission-product removal efficiencies. Unlike the current generation of operating PWRs, the Westinghouse AP600 design does not include containment sprays. The staff will review the AP600 radiological consequence assessment to determine if the dose reference values in 10 CFR Part 100 and the control room operator dose criteria of GDC 19 can be met without a containment spray system. That determination will involve evaluation of aerosol removal rates by natural deposition, containment leak rates, and assurance of pH control.

ISSUE 10: Radioactive Aerosol and Iodine Removal by Engineered Safety Features (ESF) Atmosphere Cleanup System (for AP600 and SBWR)

Containment atmosphere cleanup systems are to be provided as necessary to reduce the amount of radioactive material released to the environment following a postulated design-basis accident in accordance with GDC 41, 42, and 43. These GDC also require that these systems be designed to permit appropriate periodic inspection and testing to ensure their integrity, capability, and operability.

The EPRI requirements document for passive plant designs does not require an ESF atmosphere cleanup system, and the passive LWR vendors have not provided these systems in the AP600 and SBWR designs. The System 80+ design includes an ESF atmosphere cleanup system with charcoal adsorbers for the annulus building ventilation system. However, ABB-CE did not claim any fission-product removal credit for charcoal adsorbers; therefore, they will not be required to be in the CE System 80+ technical specifications. The GE ABWR design includes an ESF atmosphere cleanup system with HEPA filters and charcoal adsorbers for the reactor building (standby gas treatment system) and control room habitability system. Since the chemical forms of iodine will be predominantly in particulate form, HEPA filters rather than charcoal adsorbers will play a major role in removing airborne fission products following an accident.

The passive LWR vendors have submitted the radiological consequence assessments to demonstrate that without an ESF atmosphere cleanup system, the passive LWR designs can meet the dose reference values of 10 CFR Part 100 and the control room operator dose limits in GDC 19 using the fission-product release timing based on its specific passive LWR design and selected accident sequences and using their best-estimate fission-product removal efficiencies. The staff will review these assessments and provide its conclusions in the safety evaluation reports.

Therefore, the staff will not require an ESF atmosphere cleanup system, provided the passive LWR vendor demonstrates that the passive LWR design, including adequate pH control, can meet the dose reference values in 10 CFR Part 100 and the control room operator dose limits in GDC 19.

ISSUE 11: Atmospheric Diffusion Model for Control Room Habitability Assessment (for CE System 80+, AP600 and SBWR)

The staff has developed with contractor assistance a new model for determining atmospheric relative concentrations (χ/Q) in building wakes to be used for control room habitability assessment. The new model is applicable for either TID-14844 or revised accident source terms and may be used by the evolutionary and passive LWR vendors and existing licensees. The staff will use this model in its review of evolutionary and passive LWR designs using the given building layouts and standard plant site design parameters.

The current guidelines for determining χ/Q are given in Regulatory Guides 1.3, 1.4, and 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and in SRP Section 6.4, "Control Room Habitability Systems." The new model, in general, is more realistic in determining the atmospheric dispersion factors for assessing concentrations of radioactive material at the control room emergency air intake except for the reactor sites with a high average wind speed (e.g., sites around the Great Lakes in the United States). This model is not proposed for backfitting on existing plants. However, because it reflects the current understanding of building wake behavior, the staff plans to make this new model available to existing licensees who may voluntarily propose to use it in future license amendments. The staff will use this new model for reviewing future reactor designs and their siting.

The new model differs from the current model in that it assumes the wind will transport radioactive materials directly from the release point to the control room air intake only if the wind direction is within ± 30 degrees of the direction needed to transport radioactive material directly to the intake. It also determines 2-hour and 4-hour average concentrations from hourly centerline χ/Q values. It will use hourly, sector-average χ/Q values to calculate average concentrations for periods of 24, 48, 96, 168, 360, and 720 hours. All averages will be computed as moving (overlapping) averages of the appropriate hourly χ/Q values.

In May 1994, NRC contractor (Pacific Northwest Laboratories) conducted a peer review group meeting for NRC to discuss and resolve, by consensus, any comments on this new model. The peer review group consisted of experts having demonstrated expertise in meteorology and atmospheric dispersion modeling. The peer group did not raise any major concerns, and the staff is considering their comments for incorporation into the model.

The staff will use the new model in determining the atmospheric dispersion factors for assessing concentrations of radioactive material at the control room emergency air intake following a design-basis accident for evolutionary and passive LWR design reviews.

ISSUE 12: Failure of Heat Exchanger Tubes in the Passive Containment Cooling System (for SBWR)

The SBWR passive containment cooling system (PCCS) will remove the core decay heat rejected to the containment after a LOCA. It consists of three independent closed loops that are extensions of the containment. Each loop contains a heat exchanger that condenses steam on the tube side and transfers heat to the water in a large pool located outside the containment. The PCCS loops are operated at a low pressure and are initially driven by the pressure difference created between the containment drywell and suppression pool and then by gravity drainage of steam condensed in the tubes. Three PCCS loops are an extension of the containment and do not have isolation valves. The potential for containment bypass in this manner only applies to the SBWR design.

Should the PCCS heat exchanger tubes fail, the PCCS will provide a bypass pathway for the SBWR containment, releasing radioactive fission products from the containment atmosphere to the reactor building through the passive containment cooling (PCC) pool water. The staff expects the radioactive aerosol scrubbing by the PCC pool water to be similar to that by the BWR pressure-suppression pools.

The staff has not completed its review of the SBWR PCCS design, nor has it determined the important characteristics (such as carrier fluid conditions, failure configuration, pathway evaluation, and pool thermal and hydraulic characteristics) affecting the radioactive aerosol removal (source term behavior) by the PCC pool water. The staff's evaluation of this potential containment bypass with its source term behavior will be given in the safety evaluation report for the SBWR. The discussion in this section provides information relative to staff commitments identified in SECY-93-087.