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For Comment

Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

Office of Nuclear Regulatory Research

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U.S. NUCLEAR REGULATORY COMMISSION

REQUEST FOR COMMENTS ON NUREG DOCUMENT

REGULATORY IMPACT OF NUCLEAR REACTOR ACCIDENT

SOURCE TERM ASSUMPTIONS

The U.S. Nuclear Regulatory Commission has issued a report in its NUREG series. This series presents technical information gathered by the staff or its contractors and consultants on subjects related to the activities of the U.S. Nuclear Regulatory Commission and to its responsibilities.

Issued for comment, NUREG-0771, "Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions," discusses the impact of fission product iodine and fission product aerosols on past licensing practice, present regulations, and possible future licensing application. The information has been developed to help clarify the bases for the staff's current practices on the reactor accident source term assumptions, and to provide a resource base for the future staff reviews in this area.

Comments and suggestions in connection with the use of this information or the content of the document are encouraged at any time. Public comment on this issuance, if received by September 30, 1981, will be particularly useful for rulemaking considerations.

Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Office of Nuclear Regulatory Research.

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Dated at Bethesda, Maryland, this 10th day of June 1981.

THE U.S. NUCLEAR REGULATORY COMMISSION

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PREFACE

The motivation and objective for this report was set by the NRC staff group, the Degraded Cooling Steering Group, that had the responsibility for and coordinating the staff's efforts in a number of on-going rulemaking activities that address severe accident situations. The Steering Group requested preparation of an in-house report on the impact of fission product iodine and fission product aerosols on past licensing practice, present regulations, and possible future licensing application. It was also requested that this report be prepared in parallel with NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents."

The objective was to lay the necessary groundwork so that the findings of the Technical Bases Report can be appropriately incorporated into the regulatory and licensing process. This report therefore represents the in-house effort to carry out this objective.

PROLOGUE

Recent statements have been made concerning fission product release during an accident which question the appropriateness of current assumptions concerning the magnitude and chemical form of fission product iodine, and the potential for releasing aerosols to the environment. This concern has been expressed in a variety of forums including: communications to the Chairman of the Nuclear Regulatory Commission (see Appendix A for a reproduction); presentations to the Commission on November 18, 1980, presentations to the Nuclear Safety Oversight Committee (see Appendix B for a reproduction of a letter from NSOC to the President), and a number of papers presented at recent professional society meetings.[1,2]

Prompted by these concerns, the Nuclear Regulatory Commission undertook an examination of the technical bases of accident source terms, and the potential impact of changes of the current assumptions to the regulatory process. The former is addressed in NUREG-0772, Technical Bases for Estimating Fission Product Behavior During LWR Accidents, (hereafter called the Technical Bases Report), while the latter is the subject of this report. The purpose of this report, therefore, is to address the potential impact of changes in the assumptions concerning the postulated accidental fission product release on:

- Licensing practice, particularly as related to Engineered Safety Features (ESF) design and evaluation;
- Proposed rulemakings (Emergency Preparedness, Siting, Degraded Core, Minimum Engineered Safety Features); and
- Environmental and Risk Assessments.

1. SUMMARY

The current licensing process has evolved over the past 20 years. It is based upon the concept of the defense-in-depth in which power plant design, operation, siting, and emergency planning are the cornerstones of safety. From this foundation, specific safety requirements have evolved through a number of criteria, procedures and evaluations, as reflected in regulations, regulatory guides, standard review plans, technical specifications and license conditions and TID, WASH and NUREG documents. Interpretations of these criteria and evaluations form the basis for the licensing practices. Table 1.1 shows those current regulatory requirements relevant to the topic of this report.

Through this evolutionary process the concept of the design basis accident (DBA) has been formed. By interpreting the criteria and evaluations, the design basis concept has been molded into the licensing process. As reflected in Parts 50 and 100 of Title 10 of the Code of Federal Regulations (CFR), the defense-in-depth concept has focused on a limited number of accidents scenarios - notably the loss-of-coolant accident (LOCA). The development of the fission product source term used in the design basis LOCA is outlined in Appendix C.

As described in Appendix C, the Design Basis Accidents (DBAs) are a set of accidents which have been chosen to envelope the anticipated worst credible conditions in what was perceived to be a very conservative manner. Thus these accidents are not representative of expected or realistic conditions but have been judged to bound any credible accident. Specific research findings are difficult to apply to this non-mechanistic structure. In addition, the DBA-LOCA cannot be expected to be predictive of any specific accident situation. The DBA used as the representative accident for the consideration of fission product release, is the LOCA referred to in 10 CFR 100.11 for site selection purposes. This accident has also been used as the basis for engineered safety feature (ESF) design.

Under current assumptions the siting DBA-LOCA source term is dominated by noble gases (Xenon and Krypton) and iodine (primary in elemental form); these being the surrogates for all radionuclides and having been selected, in part, on volatility considerations. The Technical Bases Report has concluded that the dominant chemical form of iodine released to the containment for most Light Water Reactor (LWR) accident sequences would be cesium iodide (CsI), not elemental iodine (I_2). Since CsI is a highly water soluble and non-volatile compound one might expect that this should substantially reduce the magnitude of a potential release as compared to the relatively volatile elemental iodine form. The Technical Bases Report, however, also concluded that except for those accidents in which the fission products are released through water, the amount of iodine calculated to be released would not be subtantively reduced by the chemical form (I_2 or CsI).

Because the DBAs are non-mechanistic hypothetical events which are not intended to reflect reality in a best estimate way, changing one isolated factor (e.g., I_2 into CsI) is not necessarily justified solely on the grounds that it is more realistic. In order to properly account for specific research results, such as the recent information concerning CsI, a mechanistic analysis of the releases of all fission products is necessary. Since the mix of fission products

TABLE 1.1

REGULATORY REQUIREMENTS RELATED TO ACCIDENT SOURCE TERM ASSUMPTIONS

DEFENSE IN DEPTH	BASIC CRITERIA	ACCEPTANCE MEASURES
DESIGN		
• System Effectiveness (Engineered Safety Features)	10 CFR 50 (General Design Criteria) 10 CFR 100	Regulatory Guides - 1.3, 1.4, 1.5, 1.7, 1.25, 1.52, 1.77, 1.96, 1.97, 1.145
• Equipment Qualification		Standard Review Plan - 6.5, 9.4, 14
• Instrumentation		
• Shielding Requirements		
• Habitability Specifications		
OPERATION	10 CFR 50	Technical Specifications
SITING	10 CFR 100	Reg. Guide 1.3, 1.4, 4.7
EMERGENCY PLANNING	10 CFR 50, Appendix E	NUREG-0654

--------------------------INTERPRETATION AND PRACTICE------------------------

released varies with core, primary system, and containment conditions, a spectrum of accidents must be considered to properly account for the source term. Such considerations are already working their way into the regulatory process through probabilistic risk assessments, emergency planning requirements and environmental impact statement evaluations.

Over the past few years, a significant body of information has been developing which attempts to realistically describe the release characteristics that can be expected to occur during postulated accident conditions. Mechanistic models have been developed which attempt to represent the physical processes associated with the dynamic events that would be expected to occur during an accident. Realistic estimates have been made of the source terms for a spectrum of postulated conditions. These estimates as affirmed in the Technical Bases Report, allow for a better understanding of fission product behavior for a spectrum of accidents with consequences ranging from benign to severe.

In addition to attempting to realistically describe the release characteristics of the postulated accidents, probabilistic risk assessment techniques have permitted the assessment of the probabilities of a sequence of events which enables the development of risk dominant perspectives. These insights have begun to provide a basis for an improved assessment of reactor safety. Table 1.2

TABLE 1.2

REGULATORY ACCIDENT ASSUMPTIONS

	DESIGN BASIS ACCIDENTS	ACCIDENT SPECTRUM
DESCRIPTION	Conservative Non-Mechanistic Surrogates	Realistic Mechanistic Probabilistic
ACCIDENT TYPES	Steam Line Break Steam Generator Tube Rupture Fuel Handling Loss of Coolant Accidents 1. Emergency Core Cooling 2. Containment Structural Design 3. Siting/Engineered Safety Features (ESF)	Group 5 - DBA Equivalent Group 4 - TMI Like Group 3 - Containment Meltthrough Group 2 - Containment Failure Group 1 - Containment and ESF Failure
RELEASE CHARACTERISTICS	1) Coolant Activity 2) Gap Activity 3) 100% Noble Gases, 25% I_2 LOCA Environments	Curies Released Temperatures Pressures Chemical Environment Particle Sizes & Loadings Chemical Forms Dynamics & Timings Energies Daughter Products

summarizes the types of accidents and their release characteristics which have been used in the regulatory process. This table highlights the differences between the DBA-LOCA approach and a full spectrum of postulated accidents as is currently examined in the Technical Bases Report. To begin to account for the information that is now available, a reassessment of the design basis concept is in order.

A number of specific regulatory requirements could be affected by substantive changes in the assumptions concerning accident source terms. The regulatory requirements examined in this report with respect to potential impacts of accident source term assumptions include:

(a) Regulations (10 CFR Parts 50 and 100);
(b) Regulatory guides;
(c) Technical specifications (limiting conditions of operation);
(d) Emergency preparedness procedures; and
(e) Evaluation methods for assessing the environmental impact of the accident.

It is concluded that the appropriate method for incorporating new information concerning accident source terms, in the areas of emergency planning, siting, minimum engineered safety features, and degraded core, is via the on-going rulemaking process.

2. ACCIDENT CONSIDERATIONS

The two major hypotheses forming the basis for the conclusions reached by Stratton, et al. (see Appendix A) are that:

(1) Under LWR accident conditions (i.e., reducing environment) iodine is released from the core as cesium iodide (CsI); and

(2) This release occurs such that the CsI will be dissolved in water.

The first of these assertions will be addressed in the context of the source term variations considered in this report. The latter assumption is appropriately addressed by considering the variety of possible accident sequences.

Appendix C of the report details the past and present accident fission product release assumptions. As the discussion indicates, a complete spectrum of accidents has been postulated for various purposes, including accidents postulated as an aid in safety system design, and site selection (DBA's), and the spectrum of severe accidents analyzed by probabilistic risk assessment techniques (e.g., WASH-1400).[3]

2.1 Design Basis Accidents

The evaluation of the safety of a nuclear power plant includes analyses of the response of the plant to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. Such safety analyses provide a contribution to the selection of the design specifications for components and systems.

Design Basis Accidents (DBAs) are postulated as an aid in the design and evaluation of a variety of safety-related systems and equipment, and may involve a wide range of postulated fission product releases. Design Basis Accidents, which consider the release of substantial amounts of fission products, include: (1) accidents involving the release of activity normally circulating in the primary coolant (e.g., steam line break, steam generator tube rupture, instrument line break), (2) accidents involving the release of radioisotopes contained in the void space between fuel and cladding (e.g., rod ejection (PWR) or rod drop (BWR), fuel handling accidents), and (3) the Design Basis Accident postulated for site analysis (siting DBA-LOCA), involving the release of fission products from the fuel, in addition to coolant and gap activity.

The DBA postulated for purposes of site analysis in accordance with 10 CFR 100 involves the largest fission product source term postulated for any design basis accident. Paragraph 11 of Part 100 states in a foonote:

> "The fission product release assumed for these calculations should be based upon a major accident. Hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, 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."

Although this "maximum credible" accident, postulated for site analysis, is a non-mechanistic event, i.e., no specific accident sequence leading to the postulated releases is specified, the following accident characteristics are prescribed:

(a) substantial melting of the core is to be assumed implying a degraded performance of those systems designed to prevent core melting;

(b) containment integrity is to be assumed throughout the accident, implying that core damage is limited such that complete basemat melt-through, or other containment failure mechanisms are prevented; and

(c) functioning of enginered safety features (ESF's) designed to mitigate the consequences of this event is assumed.

The staff's interpretation of these criteria is contained in Regulatory Guides 1.3 and 1.4, which specify a loss-of-coolant accident environment in conjunction with this accident. Loss-of-coolant accident conditions may include a range of specific LOCA sequences, ranging from a small break in which the primary system remains pressurized and at least partially filled with water for minutes or hours, to a large, double-ended "guillotine" break of a main coolant recirculation pipe with a complete blowdown of the primary coolant in a matter of seconds. Since the latter is considered more severe, in terms of potential overheating and fission product release, the Regulatory Guides 1.3 and 1.4 conditions correspond to the large LOCA conditions, in which the only water left in the primary system at the time of the fission product release is superheated steam.

2.2 Accident Spectrum

These accidents cover a spectrum of releases which range from those accidents falling within the design basis envelope, as described in the previous section, to those accidents which have been calculated to release to the atmosphere significant fractions of the available radioactive material in the reactor core. For core accidents, the lower range of the spectrum would include accidents in which a core "melt-through" of the containment would occur. The upper range of the core-melt accidents is categorized by those in which the containment catastrophically fails and releases large quantities of radioactive materials directly to the atmosphere, e.g., containment over-pressurization. Radioactive materials which are calculated to be released include noble gases, organic iodine, elemental iodines, and particulate material such as cesium, tellerium and ruthenium. Thus, there is a full spectrum of potential releases between the lower and upper range with all of these releases involving some combination of atmospheric and/or melt-through accidents.

The physical conditions characterized by the spectrum of accidents includes wide variations in temperature, pressure, chemical environment (e.g., oxidizing or reducing) and timing of various critical events (e.g., core melt, vaporization, vessel melt-through). Probabilistic risk assessment, PRA, has been used to analyze these accidents. This process attempts to carry the accident sequences through to conclusions concerning the degraded state of the containment and assessments of the range of potential consequences and the associated probabilities.

By reviewing the accident sequence data, five distinct divisions, or groupings, of severe accidents can be classified. These groupings are based upon varying degrees of significant core and containment safety feature failures and therefore can represent any design in a generalized manner. The brief descriptions characterize the accident groups:

- Group 5 - Limited core damage. No failures of engineered safety features beyond those postulated by the various design basis accidents are assumed. The most severe accident in this group includes substantial core melt, but containment functions as designed (siting DBA equivalent).
- Group 4 - Limited to modest core damage. Containment systems operate but in somewhat degraded mode (TMI-2 equivalent)
- Group 3 - Severe core damage. Containment fails by basemat melt-through. All other release mitigation systems have functioned as designed (analogous to Reactor Safety Study Pressurized Water Reactor, PWR, Categories 6 and 7)
- Group 2 - Severe core damage. Containment fails to isolate. Fission product release mitigating systems (e.g., sprays, suppression pool, fan coolers) operate to reduce release (analagous to Reactor Safety Study PWR Categories 4 and 5)
- Group 1 - Severe core damage. Essentially involves loss of all installed safety features. Severe direct breach of containment (analogous to Reactor Safety Study PWR Categories 1 and 3)

The set of accident sequences in each group is large, and, to some degree is plant design specific. A representative selection of accident sequences drawn from a variety of plant designs were analyzed in the Technical Bases Report. They can be categorized into one of the five groups in the following manner:

- Group 5 - Terminated LOCA, AD with ECCS recovery
- Group 4 - TMI-2 equivalent
- Group 3 - Containment meltthrough
  $S_2D\varepsilon^1$, $TMLB'\varepsilon^1$
- Group 2 - Containment isolation failure
  $AD\text{-}\beta^1$, $AE\text{-}\delta^2$, $TQUV\text{-}\delta^2$, $TC\text{-}\delta^2$
- Group 1 - Containment and ESF failure
  $TMLB'\text{-}\delta^1$, V^1, $S_2HF\text{-}\delta^1$, $TC\text{-}\gamma^2$

NOTES: 1 refer to Table 2.1 for explanation of symbols.

2 refer to Table 2.2 for explanation of symbols.

TABLE 2.1

KEY TO PWR ACCIDENT SEQUENCE SYMBOLS

A	-	Intermediate to large LOCA.
B	-	Failure of electric power to ESFs.
B'	-	Failure to recover either onsite or offsite electric power within about 1 to 3 hours following an initiating transient which is a loss of offsite AC power.
C	-	Failure of the containment spray injection system.
D	-	Failure of the emergency core cooling injection system.
F	-	Failure of the containment spray recirculation system.
G	-	Failure of the containment heat removal system.
H	-	Failure of the emergency core cooling recirculation system.
K	-	Failure of the reactor protection system.
L	-	Failure of the secondary system steam relief valves and the auxiliary feedwater system.
M	-	Failure of the secondary system steam relief valves and the power conversion system.
Q	-	Failure of the primary system safety relief valves to reclose after opening.
R	-	Massive rupture of the reactor vessel.
S_1	-	A small LOCA with an equivalent diameter of about 2 to 6 inches.
S_2	-	A small LOCA with an equivalent diameter of about 1/2 to 2 inches.
T	-	Transient event.
V	-	LPIS check valve failure.
α	-	Containment rupture due to a reactor vessel steam explosion.
β	-	Containment failure resulting from inadequate isolation of containment openings and penetrations.
γ	-	Containment failure due to hydrogen burning.
δ	-	Containment failure due to overpressure.
ε	-	Containment vessel meltthrough.

TABLE 2.2

KEY TO BWR ACCIDENT SEQUENCE SYMBOLS

A	-	Rupture of reactor coolant boundary with an equivalent diameter of greater than six inches.
B	-	Failure of electric power to ESFs.
C	-	Failure of the reactor protection system.
D	-	Failure of vapor suppression.
E	-	Failure of emergency core cooling injection.
F	-	Failure of emergency core cooling functionability.
G	-	Failure of containment isolation to limit leakage to less than 100 volume percent per day.
H	-	Failure of core spray recirculation system.
I	-	Failure of low pressure recirculation system.
J	-	Failure of high pressure service water system.
M	-	Failure of safety/relief valves to open.
P	-	Failure of safety/relief valves to reclose after opening.
Q	-	Failure of normal feedwater system to provide core make-up water.
S_1	-	Small pipe break with an equivalent diameter of about 2 inches to 6 inches.
S_2	-	Small pipe break with an equivalent diameter of about 1/2 inches to 2 inches.
T	-	Transient event.
U	-	Failure of HPCI or RCIC to provide core make-up water.
V	-	Failure of low pressure ECCS to provide core make-up water.
W	-	Failure to remove residual core heat.
α	-	Containment failure due to steam explosion in vessel.
β	-	Containment failure due to steam explosion in containment.
γ	-	Containment failure due to overpressure - release through reactor building.
γ'	-	Containment failure due to overpressure - release direct to atmosphere.
δ	-	Containment isolation failure in drywell.
ε	-	Containment isolation failure in wetwell.
z	-	Containment leakage greater than 2400 volume percent per day.
η	-	Reactor building isolation failure.
θ	-	Standby gas treatment system failure.

3. ACCIDENT SOURCE TERMS

"Accident source term" represents the radioactive material released in a nuclear reactor accident. As used in this report, it is synonymous with "fission product release," "accident release," or similar terminology.

In this section the source term alternatives from the present assumptions will be discussed for the two categories of accidents described in the previous section, i.e., design basis accidents, and the severe accident spectrum.

3.1 DBA Source Terms

As noted in the previous chapter, the design basis accidents are non-mechanistic aids for safety system design and site selection. The source terms for these events, therefore, are based on assumed "worst case" conditions. Because of its volatility and biological concentration in the thyroid, iodine has been the fission product of primary concern in DBA analyses. As described in Appendix C, intentional conservatisms in the iodine source term have been used to create what is perceived as a substantial safety margin, which in turn compensates for uncertainties in the analysis, as well as non-conservative omissions made to simplify the day-to-day analyses used in the licensing process.

The most significant of these simplifications is the omission of all non-gaseous fission products other than iodine from the source term. The TID-14844 calculation of "site distance factors" referenced in 10 CFR 100.11 included the assumption of the release of one percent of the core inventory of solid fission products, dispersed into the containment as aerosols. Later codification of this source term in Regulatory Guides 1.3 and 1.4 emphasized the noble gas and iodine release and dropped the aerosol (solids) source term.

Similarly, the staff's routine calculations with this source term are performed for an adult thyroid, although it is recognized that a child's thyroid may be the limiting case. Once again, the staff looks upon the "safety margin" established by intentional conservatisms to compensate for this non-conservative assumption.

For these reasons the replacement of any one of the several assumptions constituting the DBA source term by a more accurate, or more "realistic" assumption would not necessarily result in an improved accident evaluation. Nevertheless, for the purpose of assessing the impact of source term modifications, a change of the assumed chemical form of iodine from I_2 to CsI is addressed below. The subsequent sections addresses an alternative which appears to be more appropriate in light of the above discussion, i.e., a change of <u>all</u> assumptions concerning DBA source terms to a best estimate basis.

3.1.1 Cesium Iodide

This supposition serves to identify the maximum impact of the hypothesis that iodine is released from the fuel in the cesium iodide form.

Table 3.1 summarizes the effects of this supposition on the major regulatory requirements resulting from varous postulated design basis accidents. No changes are identified for the accidents involving the release of fission products circulating in the primary system without additional fuel failures (i.e., steam line break, steam generator tube rupture (SGTR), and instrument line break). This is the result of the following characteristics common to these accidents:

(1) The iodine release mechanism is the blowdown or flashing of pressurized, hot primary or secondary coolant which carries the iodine in solution; and

(2) The release via this blowdown is either directly to the atmosphere, or via an unfiltered release path from a building outside the containment.

In current licensing calculations, the staff assumes that all of the iodine carried by small water droplets formed in the blowdown/flashing process is released. The quantity of iodine released, therefore depends primarily on the quantity of water dispersed in small droplet form. In the case of a SGTR the calculated iodine release also depends on the drop size, since larger drop sizes are removed by the steam drying portion of the steam generator. This process of droplet formation is independent of the chemical form of the iodine in solution and, therefore, would not differ from current assumptions if a cesium iodide release is postulated. Following evaporation of the water droplets, however, a CsI aerosol would have to be considered in lieu of iodine in vapor form.

For accidents involving fuel failure, and subsequent release of cesium iodide in aerosol form, a change in filtration requirements is the application of high efficiency particulate air (HEPA) filters and less emphasis on charcoal adsorber units (charcoal traps, however, may still be necessary for normal operation). The different deposition behavior of aerosols would have to be considered for in-plant equipment qualification. In addition, the specific properties associated with cesium (e.g., 0.67 MeV gamma ray energy) would have to be taken into consideration when designing the safety systems (e.g., shielding requirements).

The estimate of comparable or somewhat higher offsite doses for cesium iodide versus elemental iodine, shown in Table 3.1 for the DBA-LOCA, are based on preliminary results of the Technical Bases Report. The draft of that report, available at the time of this writing, also indicates comparable releases for either I_2 or CsI from the primary system for degraded core sequences, and comparable to somewhat lower effectiveness of CsI aerosols of small particle sizes (i.e., small mass loadings).

Consequently, short-term doses calculated at the site boundary would be comparable or somewhat higher when compared to calculations assuming I_2.

In contrast, long-term doses, typically calculated for 30-day periods at the LPZ distance, are dominated by equilibrium iodine partitioning for I_2 assumptions, resulting in an assumed upper limit for the effectiveness of iodine removal from the containment atmosphere. This limit would not apply for CsI aerosols, so that lower long-term doses would be calculated for the CsI case.

TABLE 3.1

EFFECT OF "CESIUM IODIDE" SUPPOSITION ON DESIGN BASIS ACCIDENTS

Accident	Resulting Reg. Requirement	Change If Only CsI is Released
Steam Line Break (BWR)	Steam line isolation within 5 sec. Coolant iodine concentration limit	No change (no fuel failure assumed)
Steam Line Break (PWR)	Secondary coolant iodine concentration limit	No change (no fuel failure assumed)
Steam Generator Tube Rupture (PWR)	Primary coolant iodine concentration limit	No change (no fuel failure assumed)
Instrument Line Break	Flow-limiting orifice in lines without isolation valves	No change (no fuel failure assumed)
Rod Ejection Accident (PWR)	Primary to secondary coolant leakage limit	No change
Rod Drop Accident (BWR)	MSIV leakage limits	No change
Fuel Handling Accident	Charcoal filters in Fuel Bldg. Ventilation System	Reduced need for charcoal filters. New model and experimental data would be necessary for fuel pool scrubbing efficiency.
DBA-LOCA:		
A. Containment leakage contribution to off-site dose	Containment ESFs	Comparable or somewhat larger doses would be calculated. ESFs not effective for particulates would not be necessary (see separate discussion of ESF effectiveness)
B. Leakage of contaminated water outside containment	Charcoal filters in Aux. Bldg.	Reduced need for charcoal filters. HEPA, rather than charcoal filters may be necessary for dose reduction purposes
C. Dose contribution from hydrogen purging	Filter train in purge flow,	Comparable to somewhat lower off-site dose would be calculated
Sum of DBA-LOCA dose contributions	Exclusion Area and LPZ distances adequate to meet 10 CFR 100	Comparable to somewhat large Exclusion Area distances would be required
Equipment Qualification/ Shielding Requirements	Current requirements include I_2 partitioning and plateout	Alternate deposition characteristics of CsI aerosol would have to be considered

The evaluations summarized in Table 3.1 demonstrate the effects of changing the assumed iodine form from I_2 to CsI in DBA analyses. The effects are seen to be minor. Based on the findings of the Technical Bases Report, a reduction of the CsI release from the primary system from the current estimates of comparability to the I_2 cases might be demonstrated with future model improvements. The order-of-magnitude reductions anticipated by Stratton et al. (see Appendix A), however, could not be realized by changing from I_2 to the CsI form in the existing structure of DBA assumptions.

3.1.2 Elemental Iodine Versus Cesium Iodide - Dispersion and Dosimetry Consequence Implications

The Technical Bases Report has shown that under current assumptions, it would be expected that releases of radioactive materials to the environment could include aerosols of both cesium iodide and cesium. This has implications with respect to atmospheric dispersion and dosimetry.

As far as atmospheric dispersion and depletion of the materials is concerned, elemental iodine has been treated as behaving like an aerosol in past analyses. This is because of the very large uncertainties associated with this question. The chemical form, therefore, has no direct impact on the manner in which the diffusion calculation is performed.

With respect to dosimetry even though cesium iodide is considerably more soluble than elemental iodine, they both would be expected to pass through the respiratory system of the body and into the blood in less than one day. Once in the blood, it is not anticipated that there would be any appreciable difference between the chemical forms since the cesium iodide would probably break down into cesium and iodide ions.

In terms of the radiological consequences, it is not expected that the difference in chemical form between elemental iodine and cesium iodide would be significant.

3.1.3 Release Spectrum

The exercise of the previous section demonstrates that the existing non-mechanistic DBA structure does not lend itself to easy adjustment to reflect specific research findings concerning one aspect of fission product release. An alternative approach is to perform the evaluations of postulated accidents on a realistic basis. The desired degree of conservatism (to account for uncertainties in the evaluation) can then be achieved by a "safety factor" multiplier applied to the end result.

A realistic treatment of accident consequences would necessitate the specification of important parameters and environmental conditions affecting actual fission product behavior (e.g., temperatures, pressures, timing of release, oxidation potential, particle size distributions, etc.). Such detail concerning the history and physical/chemical environment of the postulated release, in turn requires the specification of specific event sequences, as opposed to the non-mechanistic outer envelope concept embodied in the design basis fission product release accidents.

The primary basis for a realistic assessment of fission product release estimates at this time is the Technical Bases Report. Although substantial uncertainties persist, as identified in the Technical Bases Report, the current state of the technology does permit the calculation of fission product releases for various accident sequences, along with a reasonable quantification of the uncertainties involved.

The best estimate evaluations of an accident sequences comparable to the current DBA for siting and ESF design identify the expected dominant iodine form to be cesium iodide. Small fractions of elemental and organic iodine, however, cannot be ruled out. Because of the high effectiveness of the containment spray system for removal of large-particle aerosols, relatively small quantities (on the order of 1 percent) can become non-negligible contributors for total offsite doses calculated for large LOCA sequences.

A major difference identified in the Technical Bases Report's using a mechanistic, best-estimate analysis is the sizable aerosol mass predicted in the containment. This is an additional factor heretofore not considered in the siting DBA-LOCA. The fission products contained in this aerosol could have a substantial effect on environmental qualification of equipment and shielding requirements.

3.2 Severe Accidents Source Term Sensitivity

There are over five billion curies of radioactive materials in a large (1000 megawatt electric) power reactor at the end of a normal fueling cycle. These radioisotopes can be grouped into a few chemically similar categories for the purpose of calculating the amount of material which would be expected to be released. Table 3.2 gives the groups of radiologically important isotopes and the inventory and half-life for a typical reactor. For the five accident groups given previously, Table 3.3 summarizes the ranges of the percentages of the core inventory that would be expected to be released to the atmosphere, based on current assessments as indicated in Appendix C.

The difference in released source term between a Group 1, "worst case" release, and a Group 2, "spray functional" release, is about a factor of 100. The source term difference between a Group 2 release and a Group 3, "melt through" accident, is about a factor of 10. The difference between Group 3 and Group 4 is about a factor of 1000, and between Group 4 and Group 5 of another factor of 10. Therefore, the accident spectrum encompasses releases to the atmosphere of hundreds of curies through the release of hundreds of millions of curies.

As would be expected, the public health impact from such a variation in release would vary greatly, ranging from almost benign to very severe. To further highlight the variation between the release groups, Table 3.4 shows the ratio of results of each accident group to Group 1, the worst release, for a representative set of potential consequences.

In addition, a sensitivity study has been conducted to show the implications of reducing the release fractions of the worst release, Group 1, for the relatively more volatile fission product groups (I, Cs-Rb, Te-Sb). These isotope groups have been reduced by factors of 2 and 10, and than eliminated altogether. Table 3.5 presents the results of the study. The I,Cs category represents reductions in both the iodine and cesium groups; in addition to the previous factors,

TABLE 3.2

ACTIVITY OF RADIONUCLIDES IN A REACTOR CORE AT 3560 MWt

Group/Radionuclide	Radioactive Inventory in Millions of Curies	Half-Life (days)
A. NOBLE GASES		
Krypton-85	0.60	3,950
Krypton-85m	26	0.183
Krypton-87	51	0.0528
Krypton-88	73	0.117
Xenon-133	183	5.28
Xenon-135	37	0.384
B. IODINES		
Iodine-131	91	8.05
Iodine-132	129	0.0958
Iodine-133	183	0.875
Iodine-134	204	0.0366
Iodine-135	161	0.280
C. ALKALI METALS		
Rubidium-86	0.028	18.7
Cesium-134	8.1	750
Cesium-136	3.2	13.0
Cesium-137	5.1	11,000
D. TELLURIUM-ANTIMONY		
Tellurium-127	6.3	0.391
Tellurium-127m	1.2	109
Tellurium-129	33	0.048
Tellurium-129m	5.7	34.0
Tellurium-131m	14	1.25
Tellurium-132	129	3.25
Antimony-127	6.6	3.88
Antimony-129	35	0.179
E. ALKALINE EARTHS		
Strontium-89	101	52.1
Strontium-90	4.0	11,030
Strontium-91	118	0.403
Barium-140	172	12.8
F. NOBLE METALS & COBALT		
Cobalt-58	0.84	71.0
Cobalt-60	0.31	1,920
Molybdenum-99	172	2.8
Technetium-99m	151	0.25
Ruthenium-103	118	39.5
Ruthenium-105	77	0.185
Ruthenium-106	27	366
Rhodium-105	53	1.50

TABLE 3.2 (continued)

Group/Radionuclide	Radioactive Inventory in Millions of Curies	Half-Life (days)
G. RARE EARTHS, REFRACTORY OXIDES AND TRANSURANICS		
Yttrium-90	4.2	2.67
Yttrium-91	129	59.0
Zirconium-95	161	65.2
Zirconium-97	161	0.71
Niobium-95	161	35.0
Lanthanum-140	172	1.67
Cerium-141	161	32.3
Cerium-143	140	1.38
Cerium-144	91	284
Praseodymium-143	140	13.7
Neodymium-147	65	11.1
Neptunium-239	1800	2.35
Plutonium-238	0.061	32,500
Plutonium-239	0.023	8.9×10^6
Plutonium-240	0.023	2.4×10^6
Plutonium-241	3.7	5,350
Americium-241	0.0018	1.5×10^5
Curium-242	0.54	163
Curium-244	0.025	6,630

TABLE 3.3

POSTULATED RANGES OF ISOTOPES RELEASED FOR RELEASE GROUP SPECTRUM
(PERCENT)

Accident Spectrum	Kr-Xe	I	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
Group 1	100	30-70	30-70	30-70	1-10	1-40	.1-.5
Group 2	90	.1-1	.1-1	.1-1	.01-.1	.01-1	.001-.01
Group 3	.1-10	.01-.1	.01-.1	.01-.1	.001-.1	.001-.1	.00001-.0001
Group 4	$3x10^{-4}$	$1x10^{-5}$	$5x10^{-5}$	$1x10^{-7}$	$1x10^{-9}$	0	0
Group 5	$3x10^{-5}$	$1x10^{-6}$	$5x10^{-6}$	$1x10^{-8}$	$1x10^{-10}$	0	0

TABLE 3.4

SOURCE TERM ACCIDENT GROUP
EXPECTED CONSEQUENCE RATIOS*
(PERCENT)

Accident Spectrum	Early Fatalities	Early Illnesses	Cancer Fatalities	Dose Bone Marrow	Dose Thyroid	Property Damage
Group 1	100	100	100	100	100	100
Group 2	0.01	5.0	10	5.0	1.0	1.0
Group 3	0	0	0.02	0.04	0.02	0.02
Group 4	0	0	$1x10^{-4}$	$1x10^{-4}$	$3x10^{-5}$	-
Group 5	0	0	$1x10^{-5}$	$1x10^{-5}$	$3x10^{-6}$	-

*For explanations of consequences see WASH-1400 Appendix VI.

TABLE 3.5

VOLATILE ISOTOPE GROUP SENSITIVITY STUDY
EXPECTED CONSEQUENCE RATIOS (TO GROUP 1)
(Percent)

Sensitivity Group	Early Fatalities	Early Illness	Cancer Fatalities	Dose: Bone Marrow	Dose: Thyroid	Property Damage
Group 1	100	100	100	100	100	100
I-1/2	75	75	99	85	60	99
I-1/10	60	55	95	70	30	99
I-0	50	55	95	65	20	99
Cs-1/2	95	95	90	80	90	65
Cs-1/10	90	95	75	70	85	15
Cs-0	85	90	60	60	80	1
Te-1/2	75	65	95	80	90	100
Te-1/10	50	45	90	70	80	100
Te-0	45	40	90	65	75	100
I,Cs-1/2	70	70	90	70	60	65
I,Cs-1/10	45	55	70	60	30	15
I,Cs-1/20	40	50	60	55	25	10
I,Cs-1/100	40	50	55	55	20	5
I,Cs-0	40	50	55	55	20	1
I,Cs,Te-1/2	40	45	85	60	50	65

reductions of 20 and 100 are also shown. The final row represents reductions of a factor of 2 for all of these isotope groups.

From a comparison of Tables 3.4 and 3.5, it is apparent that even significant variations in an isotope group for the worst accident has relatively minor impact on the overall consequences. Whereas, the differences between the spectrum of accident groups, Group 1-Group 5, has dramatically larger reductions in consequences. From this sensitivity it follows that considerations of the entire source term spectrum is much more important than any variation that would be anticipated between the accident sequences that make up a group.

As can be seen from Table 3.5, the iodines contribute anywhere from nothing to about 80 percent of the consequences of a severe reactor accident. Depending on the consequence measures, cesium contributes from 10 percent to over 90 percent of the consequences. The tellurium group contributes from about five percent to 60 percent of the consequences. In addition, the ruthenium group is also very important in terms of potential consequences. Thus Table 3.5 shows that before significant reductions in consequences can be anticipated, all of the isotope group release fractions have to be substantially reduced. This is a very important conclusion since any isotope group, or accident sequence change, will have little impact on the overall results when considering the full accident spectrum.

4. IMPACT OF SOURCE TERM ALTERNATIVES

In this chapter the source term variations previously described are assessed with respect to their effect on current regulatory requirements, particularly the effects on the regulations, various regulatory guides, engineered safety features effectiveness and the potential impact on rulemaking assumptions. The spectrum of postulated accident sequences and radionuclide chemical forms, including noble gases, elemental iodine, methyl iodide, cesium iodide, and aerosols, will have effects on the existing regulatory assumptions, and a variety of regulatory approaches are possible. In the attempt to evaluate the impact on the regulatory process to determine to what extent the regulatory bases would have to be changed when considering the spectrum of accidents and spectrum of fission products and their various chemical forms, it should be recognized that it is not now possible to identify in detail how the various requirements would be changed, but only to determine where changes may be needed.

The first item of this review concerned the need for any modifications to Title 10 of the Code of Federal Regulations, which might be necessary to accommodate alternate accident source term assumptions. Based on this review of the regulations it was concluded that only those parts of the regulation already in the process of revision by declared intent of rulemaking were candidates for revisions arising from accident source term considerations. The effects of source term modifications on the four rulemakings (emergency planning, siting, minimum engineered safety features, and degraded core) are discussed in Section 4.4.

Another important consideration is the urgency with which changes of present regulations requirements are needed. This question could be phrased as follows: In the light of the additional information concerning fission product source terms available today, are the existing regulatory requirements adequate, or are immediate changes warranted? Based on our review of the regulatory requirements and safety system design discussed in this chapter and the consideration of the effects of the source term variations on accident consequences discussed in the previous chapter, as well as the Technical Bases Report, it is concluded that the appropriate time frame for consideration of source term modifications is the scheduled rulemaking process. The evaluation of the effectiveness of current-generation engineered safety features presented in Section 4.3 demonstrates the effectiveness of these systems for conditions extending beyond their design basis. It is concluded, therefore, that the current regulatory requirements do not represent unsafe design guidelines.

The accumulation of knowledge about fission product behavior during postulated accident sequences should serve as a technical base to be used for changing regulations through the rulemaking process. The Technical Bases Report represents the most comprehensive attempt to date to describe fission product behavior, transport and chemical form based on experience from limited experiments in conjunction with applied computer models. Conclusions provided in the Technical Bases Report should, however, be carefully scrutinized before use in the rulemaking process in attempted.

4.1 Technical Specification

Accident source term assumptions are reflected at various levels in the structure of the regulatory requirements, ranging from the Code of Federal Regulations to detailed technical specifications for the operation of a plant. The assumptions concerning fission product release and transport are directly reflected in the technical specification limits for the maximum permissible concentrations of iodine and noble gases in the primary and secondary coolant systems.

The accident analyses, which are used to determine limits on coolant activity and leakage, involve a release of fission products to the primary coolant system. The radioisotopes assumed to be released either originate from a small number of fuel rod cladding defects during normal operation, or originate from fuel to cladding gap failures. Any iodine released, therefore, would be in solution as dissociated iodine ions, independent of its chemical form during the release from the fuel. The question of the proper iodine species arises in the modeling of the potential releases from the primary coolant to the environment.

The staff has developed a detailed model of the iodine transport phenomena in a PWR steam generator during a tube rupture accident.[4] The predominant species are nonvolatile iodine ions, such as would result from a release of cesium iodide to the primary coolant. Application of this model shows that the variables controlling the iodine release arise from the heat and mass transfer phenomena involved (e.g., droplet carry-over in flashing processes, bubble and drop size characteristics, etc.). Although similar conclusions would probably be reached for coolant release processes outside the steam generator, such models have not yet been developed. In absence of such models, overly simplified assumptions have been used, such as the assumption that 10 percent of the iodine in coolant is released upon flashing or evaporation of the water. Refinements of the accident models to account for these phenomena, as well as the cesium iodide form, however, are expected to result in changes which are small by comparison with the variation of other parameters (e.g., thermal-hydraulic variables), so that the technical specifications for coolant activity and primary-to-secondary leakage would not change substantially.

4.2 Regulatory Guides and Standard Review Plans

Several existing Regulatory Guides (RG) and Standard Review Plans (SRP) express staff position and criteria based upon fission product source terms for DBA sequences (LOCA and selected fuel handling accidents). Revisions in source term assumptions would require revision and reissue of many of these guidance documents.

Such revisions would be warranted for any of the following reasons:*

*The magnitude of any damage following an accident is an important aspect that should be also considered in the NRC regulations. The TMI Unit 2 accident was, and still is, a good example. Followup procedures and alternatives, special operations associated with fuel removal and transport, damaged fuel accountability and criticality estimates are the only four highlights associated with clean-up and decontamination aspects after the accident occurred. Reexamination ought to be considered and attention should be focused on evaluation of possible modifications of 10 CFR Parts 51, 71 and 73, as well as regulatory guides in Division 5 - Materials and Plant Protection.

(a) The physical state characterized by variations in temperature, pressure, radiation, chemical atmosphere (oxidizing or reducing), and timing for one of these conditions, or combination of them, could produce more severe environment conditions for the emergency equipment and systems. The relevant regulatory assumptions pertaining to emergency equipment environmental qualification may need to be revised.

(b) A spectrum of accidents and fission products could produce more severe radiological consequences for some accident sequences and could lead to the reexamination of existing shielding assumptions and requirements.

(c) Presumed high aerosol concentration as an addition to other expected fission product presence in the containment atmosphere, would require to reexamine applicable regulatory guides to be reexamined specifying requirements for sampling and monitoring containment atmosphere, air filtration system design and effectiveness.

The relationships between the engineered safety features, General Design Criteria (GDC), and the licensing guidance documents (regulatory guides, standard review plan) are given in Table 4.1.

The potential impact of source term revisions on specific regulatory guide requirements are presented below:

Regulatory Guides 1.3 and 1.4: Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors and Pressurized Water Reactors. (6/74).

The assumptions in both regulatory guides, related to the release of fission products from the fuel and consequently to the containment, specify that: (1) 25 percent of the iodine inventory will be available for leakage from the containment, (2) 91 percent of this 25 percent is to be in the form of elemental iodine, (3) 4 percent of this 25 percent in the form of organic iodides, (4) 5 percent of this 25 percent in the form of particulate iodine, and (5) 100 percent of the noble gases inventory will be available for leakage from the containment.

Revision of source term assumptions would have potential impacts on the total amount and chemical form of iodine assumed to be available for leakage from the containment, but the specified noble gases inventory will not be affected. Specifically, the interplay between iodine in the elemental form, organic iodides form, and in the particulate form (aerosol) need to be reexamined. It should be expected that the amount of iodine in elemental form would be considerably reduced, and particulate form (aerosol) would be increased. The Technical Bases Report does not provide any new information on organic iodides form and concentration. Formation of organic iodides was observed in reactor incidents (see Appendix C), and its presence is observed during normal operations of nuclear power plants. Therefore it should be anticipated that changes in regulatory assumptions regarding organic iodides would be relatively small. It should also be pointed out that other chemical form of fission products released during postulated accidents need to be considered. Cesium iodide will need to be included in future accident evaluations.

TABLE 4.1

RELATIONSHIPS BETWEEN LICENSING GUIDANCE DOCUMENTS AND ENGINEERED SAFETY FEATURES

Engineered Safety Feature	Regulatory Guides	Standard Review Plan	General Design Criteria
1. Containment Sprays	1.3, 1.4, 1.7	6.5.2, 15.6.5.A	41
2. Containment Recirculation Filters	1.3, 1.4, 1.52	6.5.1, 15.6.5A	41
3. Auxiliary Building Filters	1.52	6.5.1, 9.4.2, 3, 4	-
4. Main Steam Isolation Valve Leakage Control	1.3, 1.96	1.5.6.5.D	-
5. Standby Gas Treatment	1.52	9.4.5, 15.6.5	-
6. Ice Condenser	-	6.5.2, 3, 4	-
7. Containment Leakage	1.3, 1.4	6.5.3	16
8. Dual Containment	1.4	6.5.3	50
9. Pressure Suppression Pool	-	6.5.3	-

Introducing other fission products and their chemical form (especially aerosols) would also have impact on the assumptions for atmospheric diffusion, dispersion and depletion of the effluent plume of radioactive iodine.

Regulatory Guide 1.5: Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Safety Guide 5) (02/71).

The fission product related assumptions are: (1) the radioactivity in the coolant is assumed to be the maximum amount specified in the technical specifications, and (2) all of the iodine and noble gases from the released coolant are released to the atmosphere.

Revised source term assumptions and chemical forms would impact on the retention of iodine in the reactor coolant, concentration and chemical form of iodine to be released to containment atmosphere during a postulated accident.

Regulatory Guide 1.7: Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (11/78).

This guide discusses acceptable assumptions to control hydrogen generated by the metal-water reaction assuming that: (1) 50 percent of the halogens and 1 percent of the solids present in the core are intimately mixed with the coolant water, (2) all noble gases are released to the containment, and (3) all other fission products remain in fuel rods.

The quantity of iodine dissolved in the reactor coolant, and the solids source term would have to be reevaluated for the purpose of computing water radiolysis.

Regulatory Guide 1.25: Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (3/72).

This guide provides assumptions to be used for calculating the radiological consequences of a DBA spent fuel handling accident. The fission product related assumptions are: (1) the gap activity in the damaged rods is released and consist of 10 percent of the total noble gases, 30 percent of krypton-85, and 10 percent of the total radioactive iodine, (2) for the purpose of sizing filters 30 percent of the iodine-127 and iodine-129 inventory is assumed to be released, (3) the iodine gap inventory is composed of inorganic species (99.75%) and organic species (.25%), (4) 99 percent of the total iodine is retained in the water, and (5) the removal efficiency by adsorbers should be 90 percent for inorganic species and 70 percent for organic. Even though only 1 percent to the iodine isotopes are assumed to be released from the pool during a fuel handling accident, calculations typically show they are dominant for the assumptions provided in this guide.

The percentages of iodine assumed to be released from the damaged fuel to the containment atmosphere need to be reexamined. The iodine retention in the water needs also to be reevaluated.

Regulatory Guide 1.52: Design, Testing, and Maintenance Criteria for Post-Accident Engineering-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (3/78).

This guide specifies typical environmental accident conditions to be used for ESF primary and secondary atmosphere cleanup systems design. The specified values of iodine buildup on adsorbers, airborne concentrations of elemental iodine, methyl iodide and particulate iodine are based on source term specified in Regulatory Guide 1.3 or 1.4.

The above design capacities will be revised. The direct impact would be the evaluation of size and effectiveness of adsorbers, which are presently designed to remove gaseous iodine (elemental iodine and organic iodides), taking into account aerosols present in the containment atmosphere. Prefilter and HEPA filter design requirements need to be reexamined based on revised fission product and their chemical form, especially the loading factor of aerosol postulated for selected accident sequences in the containment atmosphere.

Regulatory Guide 1.77: Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (5/74).

Appendix B of the guide spells out the fission product related assumptions for the radiological consequences: (1) all gaseous constituents in the fuel clad gaps to be released, (2) the amount of activity accumulated in the fuelgap in the damaged fuel should be assumed to be 10 percent of the iodines and 10 percent of noble gases, and (3) 25 percent of the iodines and 100 percent noble gases in melted fuel, if predicted to occur, should be assumed to be available for release as a gas.

A reevaluation of fission product related assumptions needs to be considered.

Regulatory Guide 1.97: Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident (12/80).

This guide specifies the minimum number of variables to be monitored by the control room operating personnel during and following an accident. The bases for establishing design and qualification criteria for the fission product related instrumentation are regulatory positions provided in Regulatory Guides 1.3 and 1.4 (refer to discussion for Regulatory Guides 1.3 and 1.4).

Assuming the spectrum of fission product releases, this regulatory guide would need to be revised by reexamining iodine-related variables to be monitored and sampled. The effects of the presence of substantial quantities of aerosols would have to be considered.

Regulatory Guide 1.145: Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (08/79).

Impact. Refer to discussion of Regulatory Guides 1.3 and 1.4.

4.3 Engineered Safety Feature Effectiveness Evaluation

Engineered Safety Features (ESF) have in the past been designed on the basis of a substantial iodine source term. Other environmental challenges imposed on the ESFs have been selected from LOCA sequences. Thus, whereas the iodine source term is consistent with a severely degraded core, aerosol loadings and other fission products are not specifically included in design requirements for ESFs. The impact of the source term on the design and effectiveness of ESFs for a spectrum of degraded core sequences is discussed below in the format of "cesium iodide" and "release spectrum" source term variations, as defined in Chapter 3.

Containment Leakage Requirements. The allowable leak rate for containment systems is based on the airborne concentration of iodine and noble gases in the containment atmosphere, and on downwind dispersion and dose calculations.

If iodine in the containment were present as CsI, containment leak-tightness requirements would depend in part on the rate of depletion of airborne CsI. Based on the Technical Bases Report, cesium iodide removal from the containment atmosphere is expected to be comparable, or somewhat slower than the removal of elemental iodine by either natural processes (plate-out) or containment spray washout. Therefore, under these assumptions comparable or somewhat more restrictive leakage requirements would be placed on the containment.

For most severe accident sequences, some type of breach in containment is encountered, and the release of radioactive material is not greatly affected by stringent leak-tightness requirements.

Containment Sprays. Containment sprays remove heat as well as airborne contaminants from the containment atmosphere.

CsI has a low vapor pressure at temperatures which are predicted for containment atmospheres, and therefore, if iodine were present in the containment atmosphere as cesium iodide, it would exist in a condensed form, i.e., an aerosol particle. Sprays are known to scrub particles but the rate is highly dependent on particle size. For particles smaller than approximately 1 micrometer in aerodynamic diameter, washout would be comparatively slow. If the mass of aerosol released from the core is small, this small mass release would lead to small particle sizes for CsI. Therefore, spray washout of CsI would likely be quite slow compared to that for elemental iodine. Aerosol masses released to the containment atmosphere under severe accident conditions are quite large, and particle sizes are expected to be large enough to result in rapid spray washout.

For severe accident sequences, which lead also to elemental iodine release, the removal rate would be slightly reduced unless a chemical additive system were used. Therefore, while chemical additive systems would be of reduced benefit in those sequences where CsI is dominant, they would still be necessary for accidents involving elemental iodine releases.

Organic iodides are not effectively removed by sprays, and therefore the overall effectiveness of removal by sprays may be limited by the abundance of the organic iodides.

In the severe sequences with very high aerosol mass loadings, the potential for clogging the containment recirculation sump, and damage of the spray pumps must be considered. Based on an examination of this question in the Technical Bases Report, it is concluded that mass loadings of this magnitude would not be expected for the spectrum of accident sequences considered.

It is concluded, therefore, that containment sprays would perform their heat removal and fission product scrubbing function for most accident sequences. Scrubbing of particulate iodide (CsI) would be comparable or somewhat less rapid than for elemental iodine for sequences with low aerosol concentrations, but for severe accidents, spray washout of aerosols would be at least as effective as has been predicted for elemental iodine.

Containment Recirculating Filter System. Recirculating filter systems are used in a few PWRs to remove fission products and to cool the containment atmosphere following a DBA. The systems employ in series: moisture separators, prefilters, HEPA filters, and charcoal traps.

The filter systems require electric power to operate and are located inside the containment. Therefore, they would not be operable for some severe accident sequences.

For the source term spectrum resulting from degraded core sequences, this type of filtration train would be challenged mainly by high aerosol loads. A review of the Technical Bases Report indicates that it would take only about two minutes to accumulate the 1 kg of aerosol per filter module which is sufficient to effectively plug the filter systems. It is concluded, therefore, that filter systems would perform their design function only under conditions where dilute aerosol concentrations are encountered. For accidents which involve core-melt, the copious quantities of aerosol produced would plug the filters in a short time and they would be ineffective for much of the accident.

Auxiliary Building Filter Systems. Auxiliary building ventilation systems employ filter trains having in series: prefilters, HEPA filters, charcoal traps, and a back-up stage of HEPA filters. The fans used with these filters require electric power, hence the system would be inoperative for the sequences in which power is not available.

If the leakage from fluid recirculation systems contains iodine in the form of dissolved cesium iodide, the potential for airborne iodine releases in the auxiliary building becomes a function of the physical process involved in the postulated leakage. For water leaking in the form of drops or small streams of liquid, such as would be expected from low temperature, low pressure systems, essentially no airborne release is expected. However, if the temperature of the coolant is high enough to cause flashing, or if the leakage is in the form of a spray, a CsI aerosol would be formed in a process analagous to an industrial spray-drying process. The CsI contained in the spray or mist droplets could form sub-micron sized aerosol particles upon evaporation of the droplets. The HEPA filters in the typical auxiliary building ESF filter systems would remove this aerosol from the building ventilation effluent with efficiencies comparable or exceeding that of the elemental iodine removal efficiencies of the charcoal bed.

The spectrum source term for the auxiliary building would include consideration of elemental iodine, cesium iodide, and a small fraction of organic iodides. The design of charcoal filters effectiveness depends on whether iodine is present as elemental iodine or as organic iodides. The amount of iodine escaping increases proportionately with the fractional abundance of organic iodides. In order to limit iodine releases, therefore, the present capabilities of the filters to trap elemental iodine and organic iodides, would have to be retained, in addition to their aerosol filtration capability.

In summary, filter trains of current design would effectively trap the modest quantities of cesium iodide aerosols, elemental iodine, and organic iodide source term identified for most accidents in the present study. For only one accident sequence is the auxiliary building identified as a dominant leak path. This is event V, where the blowdown occurs in the auxiliary building. However, for this accident, the filter system would not significantly mitigate the release of radioactive materials.

Main Steam Isolation Valve (MSIV) Leakage Control System (BWR). In BWR plants, leakage control systems are provided to trap gases leaked past the inboard main steam isolation valve. This is accomplished by exhausting the space between inboard and outboard isolation valves through a filter train. This is usually accomplished by connecting the exhaust to the standby gas treatment system (SGTS).

If iodine were present as CsI aerosol, the HEPA filters alone included in the filter train would capture the CsI aerosol. For the spectrum source term, elemental and organic iodides, in addition to CsI and other aerosols, would have to be considered, necessitating HEPA as well as charcoal filters to trap the elemental and organic iodide fractions.

This system would be of limited benefit for accidents beyond the design basis, because the dominant leakage paths to the environment would bypass the main steam line.

Pressure Suppression Pools (BWR). Pressure suppression pools are designed to condense the steam released to the BWR drywell (primary containment) during a LOCA. Although pressure suppression pools are not designed to remove fission products from the containment atmosphere, some scrubbing of contaminants in the air-steam mixture entering the suppression pool would occur. The iodine scrubbing effectiveness depends on a number of variables, including the air-steam ratio of the entering gas stream, bubble size, water temperature, etc. The absorption efficiency for elemental iodine could be enhanced substantially by controlling the water pH to assure an alkaline solution.

The scrubbing function of the pool would be different for CsI particles than for I_2 vapor. Particles smaller than approximately 1 micrometer would probably be less efficiently removed than elemental iodine. Larger particles would be removed more efficiently. Therefore, for accidents, where aerosol mass is low and where CsI is as an aerosol, scrubbing by the pool would be somewhat less effective than it would be where iodine is elemental vapor.

The above conclusions apply, with minor modifications, to the source term spectrum: Iodine scrubbing would probably be less effective for DBA-type of accidents. For severe accidents, the pool would remove CsI with an efficiency comparable to that for I_2.

The presence of a small fraction of organic iodides would diminish the overall scrubbing efficiency of the pool. For organic fractions smaller than a few percent, the overall effect is not major because it would only add slightly to the penetration of other forms. If the organic fractions were larger, overall pool scrubbing effectiveness would closely parallel the organic content since organic iodides are not efficiently trapped.

In summary, pressure suppression pools would perform the steam condensation function under DBA assumptions as well as for many severe accident sequences. Similarly to sprays, scrubbing of cesium iodide would be somewhat less efficient in sequences involving low concentrations of aerosols than for severe core melt sequences. In the severe sequences scrubbing efficiency would be at least as good as has been predicted for elemental iodine.

Standby Gas Treatment System (BWR). The standby gas treatment system (SGTS) is a ventilation control system that traps contaminants leaked from the primary containment and collected in the secondary containment or reactor building. The filter train includes in series: moisture separaters, heater, prefilter, HEPA filter, charcoal trap, and a HEPA filter. This train would be expected to achieve very high efficiencies for particulate contaminants (only the order of 99.99%), whereas a somewhat lower efficiency would be obtained for elemental iodine and methyl iodide.

The SGTS requires electric power for fan operation, hence would be unavailable for sequences which assume total loss of power.

The SGTS provides significant benefit primarily for DBA-type accidents. If iodine were present as CsI, the HEPA filters would efficiently trap the CsI aerosol.

For most severe accident sequences the dominant leak paths would by-pass the SGTS, rendering it ineffective, regardless of the fission product source term.

In severe accidents where containment failure allows venting through the reactor building it appears, based on discussions in the Technical Bases Report, that the gas flow rates would fail the walls of the building, rendering the SGTS ineffective.

Pressure Suppression by Ice. Similar to the suppression pools for BWR's, the ice condenser is a passive steam condensation device designed to lower the containment pressure resulting from the LOCA. Unlike the suppression pools, the iodine removal potential of the ice condenser has been incorporated in the design of the system by adjusting the pH of the ice with sodium hydroxide such that the ice melt solution achieves a pH of about 8 to 8.5.

The scrubbing effectiveness of the ice condenser for cesium iodide is similar to that of the pressure suppression pools. Scrubbing of cesium iodide can be expected as long as the ice beds have not melted out. The scrubbing efficiency for CsI is expected to be comparable to that for elemental iodine for large particles, and somewhat less for small particles. The low aerosol concentrations, therefore, would be scrubbed with somewhat lower efficiency than those resulting from the severe core melt accidents.

Summary of ESF Effectiveness Considerations. The evaluation of ESF effectiveness, based on the evaluations in the Technical Bases Report, and as summarized in this section, is tabulated in Table 4.2. This table is intended to provide an

TABLE 4.2

AN OVERVIEW OF ESF EFFECTIVENESS

ESF	Accident Spectrum				
	Group 5	Group 4	Group 3	Group 2	Group 1
Containment Leakage	high	high	medium	medium	low
Spray	medium	medium	high	high	low
Recirculation Filters	medium	medium	low	low	low
Auxiliary Building Filters	medium	high	medium	low	low
SGTS	high	high	medium	low	low
MSIV-Leakage Control	medium	medium	low	low	low
Suppression Pool	low	medium	high	high	medium
Ice Condenser	low	medium	high	high	medium

overview of these considerations, not a complete or systematic evaluation of these systems.

A few words of caution concerning this table are in order. The categories of accidents used in the table were chosen to span the spectrum of accidents, as described in Chapter 2. Within each group, a number of different specific sequences could result in substantially different performance of the various ESFs. Some sequences, in fact, may include the a priori assumption of the failure of the specific ESFs as a result of other factors such as failure of the power supply. Obviously the performance of the system is not evaluated for those cases where the system is assumed not to function. The entry in this table, therefore, would not accurately reflect the effectiveness of the system for all accidents in the group.

This difficulty of assessing an overall effectiveness for a number of sequences demonstrates the need for probabilistic risk assessment. Only by weighting the importance of each sequence by the probability of the sequence, can the overall effectiveness of the safety system be adequately assessed. Even with these shortcomings, however, Table 4.2 demonstrates that some engineered safety features have an effectiveness which extends far beyond the design basis of the system. This is particularly true for containment spray, suppression pool, and ice condenser systems.

Some ESFs, however, do not significantly contribute to offsite dose reduction for any sequences substantially beyond their iodine-removing design basis. Internal containment recirculation filter systems are examples of this type of system. For accidents within the DBA envelope, ESF effectiveness would vary somewhat with the fission product form assumed. While the effectiveness of some ESFs would be somewhat less for cesium iodide, as compared to elemental iodine, the effectiveness of other systems (e.g., HEPA filters) would be increased. For the complete set of ESFs used in current generation LWRs, no substantial reduction in overall effectiveness results for the source term variations considered. Future changes in the regulatory process (e.g., rulemakings), however, should emphasize those ESFs which provide a substantive measure of protection for a broader spectrum of accidents.

4.4 Rulemaking Assumptions and Implications

4.4.1 Emergency Planning

On August 19, 1980, the Commission issued revisions to 10 CFR Part 50, Appendix E-Emergency Planning and Preparedness for Production and Utilization Facilities; and 10 CFR Part 70 - Domestic Licensing of Special Nuclear Material. These regulations reflected improvements in the emergency planning requirements to assure that adequate protective measures could and would be taken in the event of a radiological emergency.

The final regulation contained the following elements:

(a) As a provision to operate, an applicant/licensee is required to submit emergency plans to NRC;

(b) Emergency planning considerations must be extended to "Emergency Planning Zones (EPZ)"; and

(c) Detailed emergency planning implementing procedures must be submitted to NRC for review.

In addition, Appendix E of 10 CFR Part 50 was expanded to include:

(a) Specification of Emergency Action Levels;

(b) Dissemination of public information;

(c) Development of a public rapid notification system;

(d) Onsite technical support center and near site emergency operations facilities;

(e) Redundant communications systems;

(f) Specialized training; and

(g) Emergency plan maintenance provisions.

The source term issues could potentially impact the following elements of the emergency planning rule:

1. Emergency Planning Zones;
2. Emergency Action Levels; and
3. Public Rapid Notification System.

Each of these elements was based in part upon considerations of the consequences associated with potential accidental releases.

Emergency Planning Zones. The concept of Emergency Planning Zones (EPZ) was developed by the NRC and EPA task force on emergency planning in NUREG-0396, EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans In Support of Light Water Nuclear Power Plants," dated December 1978.

The task force was chartered to "make a determination of the most severe accident basis for which radiological emergency response plans should be developed..." It concluded that no specific accident could be used as the planning basis; therefore a spectrum of accidents was considered including those discussed in environmental reports, accidents postulated for purposes of evaluating plant designs, and the accidents involving severe core damage from the Reactor Safety Study.

Consequence and accident characteristics were used to define the following elements needed to scope the planning effort:

(a) Distance from the plant for initiation of predetermined protective actions;

(b) Time dependent characteristics of potential releases and exposures; and

(c) Types of radioactive materials which could be potentially released to the environment.

Emergency Planning Zones were established of about 10 miles for the short term "plume exposure pathways," and about 50 miles for the longer term "ingestion exposure pathways." The size of these zones was determined after considerable discussions and deliberations, and was based in large part upon source term considerations. The following considerations were used in deciding upon the 10 mile plume exposure EPZ:

(a) Projected doses from the design basis accident used for siting purposes (DBA-LOCA) would not exceed Protective Action Guide (PAG) levels outside about 10 miles;

(b) Projected doses from most postulated core melt accidents wouldn't exceed PAG levels outside about 10 miles;

(c) Even for the postulated worst case core melt accident sequences, immediate life threatening doses would generally not occur outside the zone;

(d) Planning for about 10 miles would provide a substantial base for expansion of response efforts if deemed necessary.

The following considerations were used in deciding upon the 50-mile ingestion exposure EPZ:

(a) Because of wind shifts, contamination above PAGs will generally be limited to about 50 miles;

(b) Iodine chemical form and state is very uncertain and would limit transport;

(c) Significant deposition of aerosols would be expected within about 50 miles; and

(d) Planning for 50 miles would provide the ability to respond to any situation.

The plume exposure Emergency Planning Zone (EPZ) considered a representative spectrum of potential accident sequences and radionuclide chemical forms including noble gases, elemental iodine, methyl iodide, and aerosols. As such the rationale used to justify the EPZ includes consideration of all of the elements specified by the spectrum of accidents and chemical forms given previously. The accident spectrum therefore encompasses the range of potential source term releases as analyzed in the Technical Bases Report. Thus, the choice of 10 miles was not based solely on one consideration, but was developed for a selection of potential accidents which represent distinct ranges of conditions within the spectrum. The emergency planning philosophy that was adopted with the 10-mile EPZ was one of establishing changing criteria based upon the severity of the accident as the measure to justify the planning basis. For the small releases, the lower ranges of the Protective Action Guides (PAG) would be used as the appropriate measure to base planning. For somewhat larger releases, the criteria shift to the upper ranges of the PAG and levels of exposures which would still be less than medically detectable. For intermediate level releases (e.g., Group 2 core sprays expected to work but containment fails) early injuries would be used as the measure to base the EPZ distance judgment on. Finally, for the most severe accidents, early fatalities become the immediate concern and therefore the measure to base the criteria upon. This same concept is used for all emergency response situations and is predicated upon the medical triage philosophy in which the most serious cases are always given priority. The system therefore is designed to minimize the impact of the accident by channeling resources directly to the most serious problems.

It became evident that the 10-mile planning distance was sufficient to substantially reduce the critical health effects under most potential accident conditions. However, it was recognized in the development of the EPZ that there were very large uncertainties in the source term and for precisely this reason the only comprehensive justification for the criteria was to assume a full spectrum of accidents. Further the distance was chosen in as realistic a manner as possible while given the acknowledged uncertainties of probabilistic risk assessment techniques.

Concerning the 50-mile ingestion exposure EPZ, the chemical form is immaterial to the distance assuming that either iodine or aerosols will be released. Most of the arguments to support the rationale focus on the uncertainties in the chemical form and are therefore responsive to the source term issues. But the consideration of a spectrum of accidents and impacts, again shows that a firm foundation for the decision is justifiable based upon our current knowledge of the source term issues. It should be noted that the last consideration (i.e., responsiveness to any situation) was probably the most important for the 50-mile ingestion exposure judgment.

Iodine was definitely used in the rationale, but even if there was no iodine released, convincing arguments could be made to retain about these EPZ distances based upon other radionuclide considerations (e.g., noble gases and particulates).

When cesium iodide is substituted for elemental iodine, the impact would be negligible. The DBA calculations would be forced to consider aerosols in general, and the cesium contribution to whole body dose in particular. Thus, aerosols could have substantial impact on potential dose curves. The conclusions reached concerning the postulated core melt accidents would not be significantly changed by assuming all of the elemental iodine was cesium iodide since both of the isotopes are explicitly considered and the chemical form would not substantially impact the considerations.

In addition to the distances for the EPZ, the planning basis also established the time dependent characteristics and types of radioactive materials which are associated with the spectrum of source terms. The only impact the source term issues would have on these factors would be the inclusion of cesium iodide as well as cesium and elemental iodine separately. Since the compound is very soluble in water it might improve the efficiencies of ad-hoc respiratory measures (e.g., breathing through a wet towel or handkerchief) to limit inhalation exposures. However, such factors would not change the general conclusions concerning the planning basis.

Concerning the use of Potassium Iodide as a medical prophylactic to limit radioactive iodine uptake in the thyroid, the chemical form postulated in a release (element iodine versus cesium iodide) would not be expected to change current perspectives on this issue other than to highlight the uncertainties inherent in such a program. It does help focus on the impacts that other aerosol materials (e.g., cesium, tellurium, ruthenium) would have during an accident. Potassium-iodide would be totally ineffective in mitigating the exposures form these other isotopes. Emphasis should be placed on other more comprehensive emergency protective measures such as shelter, evacuation, or respiratory protection.

Emergency Action Levels. The most important element of the Emergency Planning Rule in terms of health and safety is establishing consistent Emergency Action Levels (EAL) based upon the full spectrum of accident sequences. (See NUREG-0654, Appendix 1.) Source term issues could potentially impact the development of the equipment and means used in determining the course of the accident and potential magnitude of release. Proper instrumentation to follow the course of an accident would have to be tailored to specific chemical species and environments that would be expected to be present during a release. Further consideration of emergency operations for the spectrum of accident sequences is necessary to develop detail requirement for the Emergency Operation Facility, Technical Support Center and Nuclear Data Link are premature. Since this aspect of the rule is still in the developmental stages, the principle efforts that are undertaken should focus on the uncertainties in the source term and should assure that the spectrum of accidents and chemical forms have been adequately considered.

Public Rapid Notification System. Finally, the emergency planning rule required the capability to quickly notify the public within the plume exposure EPZ. Systems designed to meet the requirements of a prompt notification system would assure better notification than could otherwise be assumed utilizing ad hoc warning measures. The rationale for the prompt notification capability is extensively discussed in the supplementary information to the emergency planning rule revisions. The rationale is not effected by the source term issues.

In conclusion, the Emergency Planning rule should not be directly impacted by those source term issues discussed herein. The rationale to support the rule was based on a spectrum of accidents and on considerations of the uncertainties of the source terms. As such, the emergency planning basis has a solid foundation with respect to these matters.

4.4.2 Siting, Degraded Core Cooling, Minimum Engineered Safety Features

At present, the staff is developing the rationale to support the Siting, Minimum Engineered Safety Features, and Degraded Core Cooling rulemakings. At the outset of this effort, the importance of the source term was recognized and factored into all considerations. As was done for the emergency planning rulemaking, a spectrum of accident conditions has been proposed which will form the foundation for the rationale to support the rules. In this way a consistent relationship is established between the overall set of regulations. Further, the relative importances of the parts of the regulations can be "measured" in terms of core damage or public risk, thus giving a useful integral perspective to regulatory philosophy.

Such a spectrum of accidents was presented in Chapter 2 and describes five groups of events which pose significantly different threats to the public health and safety. When translating the five groups into accident source terms, specific accident sequences must be selected to represent a group. In selecting accident sequences which will represent the spectrum of accidents, it is important to clearly state the objective or purpose that such a set of source terms will serve. It is not clear that one set of source terms will be sufficient for all purposes. The accidents that are needed to establish siting or emergency planning regulations would probably not be appropriate as the set needed for establishing minimum engineered safety features or degraded core cooling requirements. This is because siting and emergency response planning could be regulated on a generic basis, whereas the minimum safety requirements would be much more dependent upon specific plant design considerations.

For minimum design features or degraded core cooling purposes, design specific sets of source terms are needed to assure that challenges to the specific features of the plant have been realistically described and are tuned to focus on risk dominant concerns. For example, the ice condenser and BWR containments, being a lower free volume and lower design pressure, are more vulnerable to significant damage and potential rupture from hydrogen deflagration. Explicit consideration of this problem is required as a minimum design feature for these designs; but in a large dry containment, hydrogen shouldn't as much of a concern. Thus, a rationale is needed to indicate how important hydrogen is for a specific design in terms of risk. Only in this manner can a consistent perspective be obtained of the benefits and costs associated with regulations. There will be significant design tradeoffs which must be considered on their individual merit. A realistic assessment is therefore needed to weigh the pros and cons of changes to a design. Additionally, there needs to be explicit recognition of the benefits afforded by accident prevention as well as by accident mitigation.

Siting and emergency planning could be considered in a generic sense. By reviewing all of the accident sequence data, it becomes clear that they can be categorized into the five groups. The groups would no longer represent a specific reactor design, but would represent an "average" or representative design. Therefore the source term spectrum would be a surrogate for current light-water reactor power plants, and because of the large uncertainties in such analyses, could be considered as "enveloping" any reactor.

The source term issues focus our attention on the uncertainties and variations in magnitude and chemical form which can be expected. This perspective emphasizes the need to consider the entire spectrum of situations and develop a range of arguments to support the conclusions. If this is done, the rationale to support the rules will have a solid foundation and the regulatory process will be internally consistent and therefore more scrutable.

4.5 Environmental and Risk Considerations

On June 13, 1980, the NRC published a Statement of Interim Policy on the Environmental Impact of Postulated Accidents requiring considerations of severe accidents as part of the Final Environmental Impact Statement. As part of the new requirements, the probabilities, source terms and consequences of a full spectrum of potential accidents, from the Design Basis through the most severe accidents considered in probabilistic risk assessment, have to be described. As such, the environmental and risk assessment calculation can be considered in the same light. Accident sequences which are to be used for both analyses can be structured to be representative of one of the five source term groups given in Chapter 2 for either generic or specific power plant design considerations. Since these calculations include a spectrum of accidents and chemical forms as indicated in the Technical Bases Report, the source term issues can only impact the uncertainties associated with specific accident sequences, but not the overall perspective given by the spectrum in its entirety.

5. CONCLUSIONS

This report has addressed the accident source term implications for the following regulatory requirements: accident evaluations; regulations; regulatory requirements; engineered safety features; emergency planning; other rulemaking; environmental impact statements and probabilistic risk assessments; and licensing practice.

Source Terms and Accident Evaluation. The report on the Technical Bases for Estimating Fission Product Behavior During LWR Accidents (NUREG-0772) concludes that cesium iodide (CsI) is the expected predominant form of iodine released in the event of an accident in a light water reactor. We conclude that this research result should be included in future regulatory requirements. Based on the uncertainties stated in the Technical Bases Report, and from an examination of past accident experience (see Appendix C), we conclude that elemental iodine (I_2) and methyl iodide (CH_3I) cannot be excluded from accident source term considerations. In addition to CsI, I_2, and CH_3I, aerosols of other important isotopes (e.g., cesium, tellurium, strontium, and ruthenium) should be included in accident source terms.

Within the current licensing framework, the Design Basis Accident (DBA) serves as the basis for engineering design, operation, and accident evaluation. The Design Basis Accidents have evolved as non-mechanistic hypothetical events with surrogate fission product source terms which are thought to be conservative. To begin to account for the current information on source terms, as contained in the Technical Bases Report, a spectrum of accident scenarios is needed to realistically estimate the source terms resulting from a range of potential accident conditions (i.e., curies released, temperatures, pressures, chemical environments, particle sizes and loadings, chemical forms, dynamics and timings, energies and daughter products).

There are large uncertainties associated with the accident source term information. Further research is needed to understand such things as: aerosol formation and deposition in the primary system; aerosol particle size distributions; and containment failure mechanisms. Within a framework of realistic accident assessment, the regulatory requirements can be modified, as appropriate, as more information is acquired in these areas.

Regulations. Both 10 CFR Parts 50 and 100 have incorporated the design basis loss-of-coolant accident (LOCA) concept. At the present time, there are significant efforts underway to revise the regulations for siting, minimum engineered safety features and degraded core cooling. This rulemaking process provides the mechanism to allow these issues to be properly addressed and incorporated as part of regulatory process.

Regulatory Requirements. Accident source term assumptions are reflected in a variety of documents providing regulatory guidance and implementing licensing requirements, ranging from regulatory guides and standard review plan sections to a facility's technical specifications. In conjunction with the rulemaking considerations the following regulatory guides would require major changes to better account for the spectrum of accident conditions: 1.3, 1.4, 1.96, 1.97, 1.145. Regulatory Guides 1.7 and 1.25 would require moderate changes and Regulatory Guides 1.5, 1.52, and 1.77 would require relatively minor changes. A thorough review of the standard review plan sections addressing these requirements would also be required to incorporate the appropriate changes. In light of the findings of the Technical Bases Report equipment qualification, instrumentation, shielding and habitability requirements should be reexamined. No major impact on currently effective technical specifications for plant operation was identified.

Engineered Safety Features. The concerns that past regulatory emphasis on iodine may have resulted in a distortion of engineered safety feature design has received particular emphasis in this review. ESFs used in current LWR designs were found to be effective for all postulated combinations of iodine source terms under Design Basis Accident (DBA) conditions. In addition, most EFSs proved to be functional for postulated accidents substantially more severe than the DBA, with the single exception of the containment internal recirculating filter systems employed at a few older plants. However, there is substantial variation in the fission product removal effectiveness of the various systems under conditions exceeding their design basis. The containment spray, ice condenser, and suppression pool systems proved most effective for a broad accident spectrum. Quantification of the fission product removal effectiveness under conditions exceeding their design basis requires additional data and model development. Changes in the licensing practice would be required to emphasize those systems which would provide a substantive measure of protection for a broad spectrum of accidents.

Emergency Planning. The emergency planning requirments have been based on considerations associated with a spectrum of accident conditions. As such, emergency planning requirements do not require additional modifications.

Other Rulemaking. Revised siting, minimum engineered safety features and degraded core cooling regulations are being developed as part of the rulemaking process which explicitly considers a spectrum of accident conditions. The source term issues highlight the need for this effort and focuses on the concept of considerations which are pertinent across the spectrum of conditions.

Environmental Impact Statement and Probabilistic Risk Assessments. The Technical Bases Report concludes that current analyses do not support the contention that the predicted consequences for the risk dominant accidents have been overpredicted by orders of magnitude in past studies, and that best estimates would indicate that as much as fifty percent of the core inventory of iodine (as CsI) could be released to the environment. Therefore, environmental impact statements and probabilistic risk asssessments would not be impacted by the source term issues except in terms of a better understanding of the uncertainties in such analyses.

Licensing Practice. Based upon the foregoing insights, it is concluded that there is substantial latitude in current power plant design to cope with the spectrum of accidents and associated conditions. The current rulemaking process is addressing the source term issues in a manner which will consider the impacts in a realistic mode consisted with research findings. Additional interim measures to correct specific deficiencies in the current regulatory framework should not be contemplated at this time. In the use of current licensing requirements in the evaluation of engineered safety features design, however, emphasis should be placed on those ESF systems effective for a spectrum of accidents and source terms.

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*Available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.
**Available for purchase from the National Technical Information Service, Springfield, VA 22161.

APPENDIX A

August 14, 1980

Chairman John Ahearne
U.S. Nuclear Regulatory Commission
1717 H Street
Washington, D.C. 20555

Dear Chairman Ahearne:

We wish to bring to your attention a matter that may be a very important development in reactor safety analysis. We believe that sufficient evidence has accumulated to show that the behavior of iodine during nuclear reactor accidents is not correctly described by existing NRC models and Regulatory Guides. Iodine volatility is grossly overestimated by these models for accidents in which substantial amounts of water are present, and escape of iodine to the environment will be extremely small (as it was at Three Mile Island) as long as reasonable containment integrity is also maintained. As a consequence, the risk to the general public presented by iodine is lower than estimated, perhaps by orders of magnitude.

Our concern with this issue originated with our involvement in the several Technical Staff Analyses for the President's Commission on the Accident at Three Mile Island. The mechanism for the behavior of iodine that we propose here was derived from those analyses, from further examination of experimental and theoretical studies involving the chemistry of iodine and cesium fission products in light water reactor fuel and systems, and from the observed behavior of iodine subsequent to fuel failures during accidents and incidents at other reactor sites. We believe that the explanation presented here will change the present concepts of the hazards involved during and subsequent to reactor accidents and, therefore, will require a critical reexamination of how these hazards and risks are calculated, and the criteria to which engineered safeguards are designed and installed.

Although the Three Mile Island (TMI) reactor core inventories of xenon-133 and iodine-131 were comparable, between 2.4 and 13 million curies of xenon escaped to the environment during the accident, while only 13 to 18 curies of iodine similarly escaped! This great disparity was identified as a matter of crucial importance early in the investigation by the President's Commission, and an effort was made to find the explanation. It was clear that we could not claim to understand the accident until this discrepancy (a factor of 10^5 to 10^6) was explained satisfactorily. Further, it was recognized that the physical and chemical conditions during the accident at TMI may not have been unique. (We note that, generally, radioiodine is the controlling fission product species with respect to site safety analysis as well as the design and operation of certain engineered safeguards.

The explanation for the very low escape of iodine that developed during the investigation by the President's Commission was that, as the temperature of the core increased, iodine diffused out of the fuel rods through the failed cladding and vaporized. The iodine escaping, if not already in the iodide form, then encountered a chemically reducing environment which converted it to iodide. The iodide subsequently went into solution as iodide ion when it contacted water. It was recognized that additional experimental work was needed to provide a quantitative description of the iodine behavior. Nevertheless, this explanation accounted for the much smaller escape of iodine that was observed at TMI compared to the amount predicted to escape if elemental iodide had been present. as is assumed in the Regulatory Guides.

We believe that this description can be strengthened and made more definitive. Although the present data are not absolutely conclusive, we believe that iodine emerged from the fuel as cesium iodide, already reduced to iodide. The reactor system environment then sustained this chemical state. Furthermore, it would have converted other iodine species, should they have been present, to iodide. Cesium iodide would be expected to condense or "plate-out" when it reached metal surfaces at temperatures at or below 400 to 500°C, and it would finally enter into solution as iodide ion as soon as water or condensing steam was encountered. The reactions of iodine species in water, and the fact that iodide ion is the dominant species, ensure that iodine volatility will be very small (compared to that implied by the Regulatory Guides, for example). A reaction causing oxidation of iodide would be necessary to increase the volatility of iodine. Additional experimental work is required to provide a quantitative description of iodine behavior, but this qualitative picture is consistent with the small escape of iodine observed in a number of incidents when water was present, such as at TMI.

This mechanism is supported by the following observations, as well as by measurements made at TMI:

1. Iodine and cesium are released congruently from PWR leakers during power transients (the iodine spiking phenomenon).

2. Thermodynamic calculations performed at several sites indicate that CsI is the stable form of iodine in LWR fuel. Further, the fission yield of cesium is larger than that of iodine, and cesium is always present in great (about tenfold) excess over iodine.

3. Irradiated fuel has been caused to fail in experiments performed under simulated accident conditions, and the iodine released is recovered predominantly as CsI rather than as molecular I_2.

4. The chemistry of iodine is such that, if water is accessible, iodine will interact with the water so that its concentration in the gas phase will be much smaller than its concentration in the water.

5. In other incidents that have led to the destruction of fuel in water systems (NRX, Spert-1, Snaptran-3, SL-1, MTR, ORR, and PRTR), we understand that a much smaller amount of iodine escaped from the systems than would be projected by the existing models. Data are hard to come by for many of these accidents and experiments, and our investigation is continuing. In marked contrast, a large fraction (20,000 curies) of the iodine escaped to the environment during the Windscale accident, which occurred under oxidizing conditions and in the absence of water.

The significance of this mechanism for iodine escape and transport can hardly be overemphasized. We assert that the unexpectedly low release of radioiodine in the TMI-2 accident is now understood and can be generalized to other postulated accidents and to other designs of water reactors. We believe that an accident involving hot fuel and a water or steam-water environment will have the same controlling chemical conditions as did the TMI-2 core and primary system. The iodine will emerge as CsI (and possibly some other iodides) and enter into the solution as soon as wet steam or water is encountered. It will persist in solution as non-volatile iodide ion as long as oxidizing conditions do not prevail.

Although we feel that the evidence is sufficiently strong to justify this letter, it is important to qualify our position. Iodine chemistry is very complex, and definitive experimental and analytical studies of iodine behavior during and following loss-of-coolant accidents are lacking. Nonetheless, it is clear that the behavior projected from the existing Regulatory Guides is wrong The current NRC assumption, that elemental iodine is the chemical form of the radioiodine released, is regarded as a conservatism, but in this case the assumption of a wrong chemical form must be regarded as an error which has compounding effects.

If, after due consideration, the NRC is satisfied that our description of iodine behavior is valid, we recommend that an urgent study and assessment be made of all available information, and appropriate actions be undertaken. With due respect we point out four consequences should our position be correct:

1. The frequently quoted fission product escape assumptions (from TID-14844 in 1962 to the more recent Regulatory Guides 1.3 and 1.4, and the Reactor Safety Study, WASH-1400) should be reexamined. The present assumptions grossly overstate iodine release from a reactor site in many types of loss-of-coolant accident, and safety criteria based on these assumptions should be reevaluated.

2. The dispersal of radioiodine in the biosphere may no longer dominate and control consideration of accidents and the design of safety systems.

3. Many, if not most, accident sequences must be reexamined in detail. The iodine risk to the general public may, in fact, be lower than previously estimated, possibly by orders of magnitude. The impact of a reduction of iodine risk on the requirements for evacuation is particularly important at this time.

4. The engineered safeguards designed for iodine control should be reexamined to assure effectiveness and optimization for the actual iodine behavior rather than the behavior currently assumed.

Finally, we realize that a major revision of NRC assumptions relative to accident analyses, dose calculations, and design of safeguards should not take place without an adequate base of technology from both experiment and theory, and especially until the Commission itself is convinced that it is appropriate to accept a revised physical and chemical description of iodine transport from fuel to the environment. On the other hand, the impact of wrong assumptions is so serious that an intensive effort should be made to establish the facts.

We are ready to offer more detailed information or further assistance should the NRC request it. We will be pleased to brief the NRC staff or any review committees you may appoint.

Sincerely,

W. R. Stratton

W. R. Stratton
Los Alamos Scientific Laboratory

A. P. Malinauskas

A. P. Malinauskas
Oak Ridge National Laboratory

D. O. Campbell

D. O. Campbell
Oak Ridge National Laboratory

cc: [illegible] [illegible] Cunningham, DOE-WASH
D. M. Kerr, LASL
H. Postma, ORNL

APPENDIX B

Nuclear Safety Oversight Committee

APPENDIX (B)

December 21, 1980

The Honorable Jimmy Carter
President of the United States
The White House
Washington, DC 20500

Dear Mr. President:

Your Nuclear Safety Oversight Committee has recently examined an issue that could have a significant impact on nuclear safety and regulation. In sum, there is evidence suggesting that radiological consequences of some nuclear accidents may be substantially less than previously assumed.

Scientists at Los Alamos and Oak Ridge National Laboratories have recently examined the unexpectedly low air-borne release of Iodine 131 at Three Mile Island and also studied the pattern of iodine releases in past reactor accidents in this country and abroad.

This research suggests that in light water reactor accidents, radioactive iodine fission products may not be released as a gas as previously assumed in the Reactor Safety Study (WASH 1400) and other studies. In the reducing atmosphere likely to be present in most light water reactor accidents, the new studies suggest that radioactive iodine would combine with cesium and enter into water solution.

If this assessment, which to our knowledge has not been refuted, proves correct, it would have major implications for such regulatory issues as plant siting and emergency planning, because the potential exposure of the neighboring population in the event of a major accident would be much lower than previously assumed.

In our view, the Nuclear Regulatory Commission and the Department of Energy should be responding more aggressively to this important development. There are outstanding technical questions surrounding the hypothesis that can and should be answered by analysis and experimentation. In our judgment, you should press for a coordinated research effort that would verify or refute this hypothesis about iodine behavior. This technical question should be resolved on an expedited basis

1133 15th Street, NW, Third Floor, Washington, DC 20005 for it bears directly on fundamental assumptions underlying some of the most important regulatory issues facing the nation.

The iodine release question is part of a broader constellation of issues involving source term estimates of the amount of radioactivity that should be expected in the event of a major accident. We believe that the entire set of issues, including fission product chemistry and aerosol formation and behavior in accident environs, deserves increased attention as well.

It would be helpful if you would designate someone from your staff to discuss this matter with us.

Respectfully,

Bruce Babbitt
Chairman

John Deutch
Committee Member

Marvin Goldberger
Committee Member

Harold Lewis
Committee Member

BB:kae

APPENDIX C

NUCLEAR REACTOR ACCIDENT SOURCE TERM ASSUMPTIONS: A HISTORICAL PERSPECTIVE

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The fission product release assumptions presently used in the regulatory framework have developed over a period of about twenty years. In order to gain a better understanding of the underlying bases for these assumptions, it is instructive to examine their historical development. This appendix summarizes the experimental bases, accident experience, and past regulatory practice related to accident source term considerations.

C.1 EXPERIMENTAL BASES

Most of the available empirical information on the release of fission products from nuclear fuels under severely degraded cooling conditions has been derived from relatively small scale experiments.

The primary source of the experimental data at the time of the development of the fission product release assumption reflected in today's regulatory framework, was the early work of Parker, et al.[1] at ORNL. In these experiments small pieces of bare UO_2 fuel were heated to the melting point in an inert (helium) atmosphere. Iodine release fractions approaching 90 percent were observed for irradiated fuel fragments heated to temperatures of about 2,200°C. Not insignificant release rates from the fuel were observed at temperatures of about 1000°C, and were greater for higher burnup fuel.

These release fractions were recognized to be applicable only to the hottest areas of the core, with substantially lower releases from fuel pellets located on the periphery, so that a core-wide release fraction of 50 percent for iodine was thought to be reasonably conservative.[2]

Experimental studies of fission product release from overheated UO_2 fuel elements which were performed prior to 1974 were reviewed by Ritzman, et al.[3] in the Reactor Safety Study. The experiments reviewed supported the following generalizations regarding fission product release:

(a) Noble gases, iodine, cesium, and tellurium were among the elements released to the greatest extent.

(b) Nearly total release (80-100 percent) of these elements would be expected if the fuel were heated to the melting point.

(c) The behavior of ruthenium was quite variable because it can form volatile oxides.

(d) Experiments seemed to indicate that iodine would be present mainly in elemental form even though thermodynamic calculations indicated that CsI was the favored species.

Experiments on fission product release done as part of the Containment Systems Experiment (CSE)[4] yielded results which are consistent with the Reactor Safety Study (RSS) findings. These results are consistent because RSS used CORRAL I as a code, using semi-empirical fits to the CSE data. Most of the noble gases, iodine, cesium, and tellurium was released when UO_2 pellets were heated to the melting point. The evolved iodine was mainly in the elemental form when the atmosphere was air, whereas the iodine was mostly attached to particles when a pure steam atmosphere was employed.

A number of experiments have been performed at several laboratories in the U.S. and in the U.K. to determine the fractional conversion of iodine to organic species. In some of the tests iodine was produced by heating irradiated fuel, in some iodine was released in the elemental form, and in one case iodine was released in a reactor incident which involved molten fuel. A 1972 review of 67 iodine transport experiments[5] resulted in the following conclusions:

(a) The fractional conversion to organic iodides was greatest for the lowest iodine concentrations.

(b) A conservative upper limit to the organic fraction applicable to a design basis LOCA was 0.04 of the iodine released to the containment atmosphere.

The same data set used to obtain the regulatory limit of 4 percent was used in the Reactor Safety Study in the attempt to obtain a realistic estimate of organic iodide formation. It was concluded that a more realistic estimate was 0.7 percent for cases where containment sprays did not operate, and 0.4 percent for cases in which iodine was removed rapidly by successful spray operation.

Recent fission product tests at Oak Ridge[6,7,8,9] were concentrated on the release of key elements at temperatures covering the range of 500°C to 1300°C. For these tests the clad failed, but the UO_2 remained far below the melting point. In general, the release of volatile elements (noble gases, I, Cs, Te) was found to be the sum of a burst release which varied with time and temperature. Another conclusion of the recent ORNL tests was that evolved iodine was mainly present as CsI.

C.2 ACCIDENT EXPERIENCE

Recent reviews of accident experience[10,11] have attempted to correlate the release of non-inert fission products, particularly iodine, with the presence of water and oxygen during the event. It is implied that the current regulatory assumptions lack recognition of the effect of these two variables on accidental releases. A brief review of accidents which might have contributed to the development of source term assumptions is in order.

Fission product releases which occurred as a result of reactor incidents are subject to large uncertainties resulting from a lack of appropriate instrumentation available at the time of the accident, the time lag between the event and subsequent sampling and analyses, and an understandable reluctance on the part of those responsible for the facility to publicize the event and its consequences.

Four accidents for which the fission product release has been comparatively well characterized are the Windscale,[12] the Stationary Low Power Reactor No. 1,[13] the Plutonium Recycle Test Reactor[14,15] and the TMI-2 accidents. Although these and similar events may have contributed to the regulator's concerns about potential accidental fission product releases, the information from these accidents has not served as the basis for the regulatory framework for accident source terms. A single exception is the very limited use of organic iodine data from the PRTR accident, as described in the PRTR accident discussion below.

The Windscale Accident

The Windscale No. 1 Pile was an air-cooled graphite-moderated, plutonium production reactor using natural uranium fuel slugs.

The accident occurred during a controlled release of stored Wigner energy (i.e., energy stored in graphite as a result radiation damage) from the graphite of the pile.

In September 1957, a spontaneous release of Wigner energy occurred while the pile was shut down. This led to a rise of temperature of the graphite, but the rise was not dangerous and there were no harmful effects. A procedure was initiated for controlling the release of Wigner energy and the first nuclear heating was applied. In this process the graphite is heated slowly and the stored energy is gradually released as additional heat, consequently the radiation damage in a large measure is annealed out. The situation, however, is unstable and can cause or intensify a safety problem in graphite-moderated reactors operating below 300°C. After that, because of dropping temperature of the graphite, the second nuclear heating was applied.

The immediate cause of the accident was the second nuclear heating. It was thought that the Wigner release was dying away and that parts of the graphite structure were being annealed. The second nuclear heating led to the accident by causing the failure of one or more cartridges whose contents then oxidized slowly*, eventually leading to the fire. By far the most likely possibility is that the second nuclear heating caused the failure of one or more uranium fuel cartridges. A second possibility, which cannot be entirely eliminated,

*In the absence of water, and the presence of air.

is that it was a lithium - magnesium cartridge which failed. The exposed uranium gradually led to the failure of other uranium cartridges and their combustion, and to the combustion of graphite. In a few days, the fire had spread and was affecting about 150 channels. The pile, as a result of this accident, was a total loss. The accident is considered the only reactor accident to date that caused contamination of large areas of the environment.

The physical conditions during the accident, characterized by the presence of abundant amounts of oxygen (air cooled) and lack of water, helped the formation of elemental iodine, and consequently the majority of iodine was released in gaseous form and iodine on particulates.

It has been estimated than the released iodine represented of about 12 percent of the total available iodine inventory. The stack filter removed the particulate iodine (between 20,000 to 50,000 curies), but that the gaseous iodine (20,000 to 30,000 curies) was released to the atmosphere from the 400 foot tall stack.

Windscale No. 1	October 1957
• Reactor type:	Once - through, air cooled, uranium - graphite
• Reactor fuel:	Magnox-clad,* natural - U
• Type of accident:	Fuel burning and melting
• Cause of accident:	Local overheating in reactor during Wigner energy release
• Extent of contamination:	Widespread; Iodine-131 contamination of milk supply of large area.
• Major Fission products	I-131, Te-152, Cs-137, Sr-89/90, Ru-106, Ce-144
• Amounts of other fission products released (Curies):	Te-132: 12,000 Cs-137: 600 Sr- 89: 80 Sr- 90: 2 Ru-106: 80 Ce-144: 80

*The magnesium cladding, in the form of the alloy Magnox (Magnox 12 contains 0.8% aluminum and 0.01% beryllium) was employed as cladding in the British Calder Hall (or similar) reactors, and adopted for low-temperature gas-cooled reactor to avoid the problem of fuel-cladding interaction.

The magnesium-uranium combination not only does not form intermetallic forms, but it does not interact in any way that would create a metallurgical bond. Reliance is placed on mechanical bonding.

The Stationary Low Power Reactor No. 1 Accident (SL-1)

The SL-1 Reactor was a natural recirculation highly enriched boiling water reactor designed for use at remote military installations as a power and heat source.

At the time of the accident the reactor was shutdown and three reactor operators were carrying out a routine maintenance operation. For this reason virtually no recording instrumentation was operating during this time, and there will always be uncertainty about the sequence of events leading to the accident.

As a part of the very extensive post-accident analysis, General Electric undertook a computer analysis of the transient which occurred at the SL-1 to: (1) establish the most probable sequence of events, and (2) identify those features which might differentiate this type of reactor and this accident from other transients studied.

General Electric has estimated that the withdrawal of the central rod would make the reactor critical, and that the power rose on a period of approximately 4 msec until steam voids and core vaporization terminated the accident. In this accident it was evident that some metal vaporization certainly occurred, and large a fraction of the core was damaged.

As a result of the accident about 20 percent of the total core plate area, containing about 40 percent of the fission products, was destroyed. It appears that 5-10 percent of the total inventory escaped from the reactor vessel. Less than 0.5 percent of the iodine-131, and a negligible fraction of the non-volatile inventory, was found in the surrounding area. It should also be emphasized that the reactor building was an ordinary type of construction, and was not designed as containment type structure.

SL-1	January 1961
• Reactor type:	Boiling water, water moderated
• Reactor fuel:	93 percent enriched U-235, Al-U plate type
• Type of accident:	Fuel melting
• Cause of accident:	Manual withdrawal of central control rod
• Extent of contamination:	Minimal
• Major fission products released:	I-131, Sr-90, Cs-137
• Amounts of approximate fission products released (Curies):	I-131 : 80* Sr-90 : 0.1 Cs-137 : 0.5

*10 Ci was released in approximately 16 hrs, and the remaining 70 Ci was released over the next 30 days.

The Plutonium Recycle Test Reactor Accident

The following synopsis of the fuel melting incident at the Plutonium Recycle Test Reactor (PRTR) is based mainly on studies of fission product transport reported by Perkins et al.[14] A companion report by Freshley et al.[15] provides a detailed thermal-hydraulic analysis and a metallographic study of the incident.

The plutonium recycle test reactor (PRTR) was a heavy water moderated - heavy water cooled test reactor. The fuel elements, made of mixed oxides of uranium and plutonium, were centered in process tubes and were cooled with recirculating heavy water which served as primary coolant. Surrounding the process tube was a shroud tube, which separated it from heavy water moderator contained in the reactor calandria. Reactor gas (helium) was flowing in the space between the process tube and shroud tube in a low pressure dry gas system which contained rupture discs to allow venting to the containment vessel atmosphere in case of overpressurization.

The center process tube of the reactor was used as a final element rupture test facility, and was cooled with light water supplied through an independent cooling system. One rod of the fuel element had been purposely defected by drilling a 1/16-inch hole through the zirconium cladding to the surface of the fuel.

On September 29, 1965 the testing fuel element failed in an unexpected manner, and a release of fission products to the containment atmosphere occurred. Overheating of the process tube at the rupture location produced a small hole (about 1/2-inch diameter) in the process tube. Superheated water flashed through the hole in the process tube into the low pressure helium system, and then escaped through ruptured discs to the reactor containment vessel. A total of about 700 grams of fuel material (about 39 percent) in the rod was lost. Approximately 100 grams of this material were found in the bottom of the shroud tube.

Some of the fuel element material, which was lost in the rupture, was dissolved or suspended in the superheated water; and, on flashing of this water into the helium filled space, an aerosol of fission products and fuel element material was formed. On entering the containment vessel particulate material began to deposit slowly on the vessel surfaces while some of the radioiodine (and noble gases) remained in the atmosphere.

Studies of fission product transport made by Perkins, et al.[14] provides the following information on iodine behavior:

(a) Approximately 40 percent of the fuel, 50 percent of the noble gas inventory and 30 percent of the iodine inventory was released from the fuel element.

(b) Of the 205 curies of I-131 released from the fuel, approximately 7 curies, or 3.4 percent were found airborne in the containment atmosphere, and only a small fraction were deposited on the containment vessel surfaces.

(c) The iodine in the containment atmosphere, several hours after the incident, was mainly in the form of organic iodides. The estimated .64 Ci of airborne iodide comprised 9.1 percent of the seven curies of iodine which was initially airborne in the containment atmosphere. Samples of airborne iodine, taken subsequent to flashing the containment atmosphere (3-11 days), showed 60 to 80 percent organic, and the remainder in the form of inorganic vapor or particulate.

(d) Most of the iodine, 70 percent, and 99 percent of the fission product aerosol were removed from the atmosphere by recirculating air coolers.

(e) The total deposition of I-131 on surfaces in the containment was about 0.05 curies. The deposition of other short-lived fission products ranged up to a few tenths of a curie. Iodine plateout on wall and ceilings was much greater (about 10-fold higher) than on floor surfaces, suggesting that a fraction of the iodine was initially in elemental form.

(f) The dominant iodine pathway to the environment was via the discharge of condensate from the air coolers.

The organic iodide fraction specified in Regulatory Guides 1.3 and 1.4, i.e., 4 percent, was based on a review of all available experimental data on organic iodide formation, including the PRTR incident results. As described in WASH-1233[5] a conservative upper bound for organic iodide fraction was determined to be 3.2 percent for core melt accidents. This percentage conversion is lower than the value actually observed in the PRTR incident by approximately a factor of 3. The PRTR experience was not considered to be directly applicable to severe accidents because the mass concentration of airborne iodine in PRTR was much lower (by orders of magnitude) than is predicted for severe core melt accidents.

The Three Mile Island Unit 2 Accident

On March 28, 1979, the Three Mile Island Unit 2 (TMI-2) nuclear power plant experienced a loss of feedwater transient that led to a series of events culminating in a partially mitigated loss-of-coolant accident (LOCA) with significant core damage. The sequence of events that led to core damage involved equipment malfunctions, design deficiencies, and human errors, each of which contributed in varying degrees to the ultimate consequences of the accident.

The President's Commission appointed to conduct a comprehensive study and investigation of the accident, and to provide a technical assessment of the events and their cause, has provided in its final document (Kemeny Report)[19] the following findings related to the accidental release of fission product to the environment:

"(a) The total release of radioactivity to the environment from March 28 through April 27 has been established as 13 to 17 curies of iodine and 2.4 million to 13 million curies of noble gases.

(b) Five hundred thousand times as much radioactive iodine (7.5 million curies) was retained in the primary loop. On April 1, 10.6 million curies of iodine were retained in the containment building's water and about 36,000 curies in the containment atmosphere. Four million curies were in the auxiliary building tanks. Almost all of the radioactive iodine released from the fuel was retained in the primary system, containment, and the auxiliary building.

(c) No detectable amounts of the long-lived radioactive cesium and strontium escaped to the environment, although considerable quantities of each escaped from the fuel to the water of the primary system, the containment building, and the auxiliary building tanks.

(d) Most radioactivity escaping to the environment was in the form of fission gases transported through the coolant let-down/make-up system into the auxiliary building and through the building filters and the vent header to the outside atmosphre.

(e) The major release of radioactivity on the morning of March 30 was caused by the controlled, planned venting of the make-up tank into the vent header. The header was known to have a leak."

The first of these findings, i.e., the quantities of iodine and xenon (13-17 curies and 2.4 million curies respectively), has been described as "unexpected." In the August 14 letter to the Commissioners[10] (see Appendix A), it is described as a "great disparity" for which an explanation was sought.

It should be noted that there are a number of reasons for the relatively small amount of iodine-131 released compared to very large release of xenon-133.

First, noble gases would be expected to be released in greater fraction than any other elements, in any accident. By their nature, noble gases are volatile, insoluble in water, and inert.

Second, of the iodine that was released from the fuel, the majority was absorbed by the primary coolant water in the core and subsequently flowed to the containment sump. Sodium hydroxide automatically injected into the containment enhanced the iodine absorption by raising the alkalinity of the water. Those phenomena were not only predictable, but the sodium hydroxide injection was specifically designed for the purpose of retaining iodine in the liquid phase following an accident.

Third, general environmental chemical conditions formed during the accident (water, steam, hydrogen) helped formation of nonelemental form of iodine. This was also predictable given the circumstances.

Fourth, about 90 percent of the iodine airborne released from the auxiliary building was trapped by the filters. This was also predictable for the given environment (steam).

Fifth, the xenon-133 that was released from the auxiliary building came from xenon-133 dissolved in the water in the building, and from the decay of iodine-133 trapped in the water.

Thus, the relative quantities of iodines and noble gases does not represent "a great disparity." It is recognized, however, that there is a substantial difference between the course of events during an accident and the regulatory assumptions concerning fission product releases.

A summary of previously published accident evaluations for TMI-2, together with observed or estimated behavior following the accident is given in Table C-1. Although substantial differences can be seen in the estimated consequences, it

TABLE C.1

A COMPARISON OF REGULATORY ASSUMPTIONS WITH TMI-2

Parameter	Regulatory Assumptions Small LOCA EIS[1]	Large LOCA SER[2]	TMI-2[3,4]
Noble gas release from fuel	1%	100%	50-70%
Iodine release from fuel	0.5%	50%**	40%
Solids release from fuel	0%	1%**	2-3%
Noble gases in containment atmosphere	1%	100%	40-60%
Iodine in containment atmosphere prior to spray	-	25%	Unknown
Iodine in containment atmosphere after spray	0.25%	1.25%	.001%
Solids in containment atmosphere after spray	0%	0%	Negligible
Noble gas escaped from containment* (30 days)	.003%	3%	10%
Iodine escaped from containment* (30 days)	.00001%	.04%	3.4%
Solids escaped from containment*	0%	0%	0.1-0.3%
Noble gas released to atmosphere	.003%	3%	10%
Iodine released to atmosphere	.00001%	.04%	.00002%
Solids released to atmosphere	0%	0%	Negligible
Highest dose to individual			
whole-body (30 days), rem	<.0005	2.1	<0.1
thyroid (30 days), rem	<.005	108	<0.004
Population dose (50 miles), person-rem	<0.1	-	2000-3500

Note: All percentages are normalized to core inventory.

[1]Final Environmental Statement, NUREG-0112, Final Supplement, Dec. 1976.

[2]Safety Evaluation Report, NUREG-0107, Sept. 1976, and Supplements 1 and 2.

[3]Rogovin, M., Director, Special Inquiry Group, NRC, Three Mile Island, Vol. II.

[4]Kemeny, et al., President's Commission on TMI-2, Vol. II, "Report of the Technical Assessment Task Force on Chemistry," Oct. 1979.

*Includes fission products carried in liquid effluents from containment.

**According to TID-14844. This step is actually omitted in the SER analysis, which starts with the assumption of 25% and no solids in containment.

is not clear that the cause of any such differences can be isolated from the many different variables. It is important to note, for example, that many of these differences are the result of a comparison of different postulated accidents. The TMI-2 event was neither a small loss of coolant accident (LOCA) without substantial fuel failures (as described in the Environmental Impact Statement, nor a large (rapid-depressurization) LOCA analyzed in the Safety Evaluation Report. Also, it should be reiterated that Design Basis Accidents are postulated as an aid in the design and evaluation of safety system performance and should not be interpreted to represent a prediction of actual accident consequences.

C.3 HISTORIC PERSPECTIVE OF REGULATORY PRACTICE

Regulatory Assumptions Concerning Source Terms

Since the earliest days of reactor power plant development, attempts have been made to define the probabilities, source terms, and consequences associated with potential reactor accidents. In 1957, Brookhaven National Laboratory published WASH-740, "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants."[16] That report presented three reactor accident scenarios as being "typical" cases. The first scenerio, "the contained case," is one in which no activity is released but a gamma shine dose from the contained source is calculated. In the second scenario, "the volatile release case," significant fractions of the volatile fission products (e.g., noble gases, halogens) are released. The third scenario, "the 50 percent release case," is an accident in which 50 percent of all fission products from the fuel are presumed to be released from the containment to the atmosphere. No explicit probabilities were assigned to these scenarios except that a probability range of between one in 100,000 to one in a billion was discussed. Calculated consequences ranged from none to 3,400 fatalities, 43,000 injuries, and 2.3 billion dollars in property damage.

With WASH-740 in mind, in 1961 regulations for site selection were developed as part of 10 CFR Part 100, Reactor Site Criteria. In conjunction with Part 100 the concept of a maximum credible accident was developed as the mechanism to evaluate the acceptability of the potential site and required engineered safety features of the containment. The Maximum Credible Accident concept was devised to place a limit on the considerations of siting and containment design. In 1962, the maximum credible accident was codified in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."[17] The TID source term, as it is known, postulated a loss-of-coolant accident (LOCA) upon complete rupture of a major coolant pipe, followed by a meltdown of the fuel and partial release of fission product inventory to the atmosphere of the reactor building (containment).* One hundred percent of the noble gases; fifty percent of the radioiodines; and one percent of the other particulate materials (solids) in the fission product inventory was assumed to be released from the fuel. The containment was assumed to leak at one tenth of one percent (0.001) per day, indicating that the containment structure was assumed to be fully effective in meeting its design leak rate. Any leakage and possible dispersion, however, of the solid fission products was neglected.

Since Part 100 provided for offsetting unfavorable site characteristics with engineered safeguards, these release assumptions came to be used as the design basis for safety systems in the containment, engineered to mitigate the release of fission products to the environs. As this practice evolved, the presumed iodine releases into the containment atmosphere were recognized as being highly conservative but this was felt to compensate for the uncertainty in, and possible nonconservatism resulting from, the neglect of detailed analysis of the release and transport of other fission products included in the 1 percent solids assumptions.

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*The cause attributed to the release to the containment was not important, although often considered to be.

A decade later (early 1970s), a design basis was established for control of hydrogen evolved by metal-water reaction based on an assumption of localized overheating of the core, but not melting. This basis was modified (reduced) some years after the issuance of Appendix K to 10 CFR Part 50, but still assumed localized overheating.

The increased thermal margins provided by Appendix K led to the definition of design bases for many systems, particularly for auxiliary systems, which by assuming no core damage downplayed the safety significance of those systems or understated the service conditions for which they should be qualified.

The stylized and intentionally conservative (i.e., overestimating the release) analysis of the release of fission product iodine was incorporated in the mid-60s into Safety Guides 3 and 4, later re-named Regulatory Guides 1.3 and 1.4. These regulatory guides have incorporated a reduction of the iodine source term by a factor of two to account, very conservatively, for natural deposition of elemental iodine on primary system and containment interior surfaces. In addition, these regulatory guides codified the staff's judgment concerning organic and particulate forms of iodine, as discussed above.

Environmental Impact Assessment

In 1971 the Atomic Energy Commission issued for comment, a set of assumptions for a "realistic" assessment of the environmental impact of accidents at nuclear power reactors, pursuant to the requirements of the National Environmental Policy Act (NEPA) of 1969. These assumptions, specified in a proposed Annex to Appendix D of 10 CFR Part 50 included a system for classifying accidents according to a graded scale of severity and probability of occurrence. Nine classes of accidents were defined, ranging from trivial to very serious. It directed that "for each class, except classes 1 and 9, the environmental consequences shall be evaluated as indicated." Class 1 events were not to be considered because of their trivial consequences. Class 9 events were recognized to have the potential for severe consequences, but their probability of occurrence was presumed to be so low that their estimated low environmental risk not to be considered in environmental impact assessment. Although this classification scheme was intended for environmental statements only, the term "class 9" accident has been used widely, and is often used synonimously with core melt accidents with severe consequences. It should be noted, however, that the proposed Annex defined this category of accidents as those sequences of "postulated successive failures more severe than those postulated for the design basis for protective systems and engineered safety features."

In contrast to the intentionally conservative treatment of accidents in the NRC's Safety Evaluations, the environmental impact assessments were intended to be realistic[18] (i.e., best estimate). A comparison of similar accidents in the staff's safety evaluation and the Environmental Impact Statement for a given plant, therefore, provides some indication of the staff's own perception of the degree of conservatism included in it's Safety Evaluations. Such a comparison is shown for the TMI-2 plant in the first two columns of Table C.1. The range of the parameters assumed in these evaluations is from one to three orders of magnitude. The final results, i.e., estimated doses, differ by factors of 4,000 and 20,000 for whole-body and thyroid doses.

On June 13, 1980, the NRC published a statement of interim policy on the treatment of accidents in environmental impact statements. This statement specified that future environmental impact statements would address the risk arising from all power reactor accidents in the context of their probability of occurrence.

The accident sequences addressed under the new policy include but are not limited to those that can reasonably be expected to occur. In-plant accident sequences that can lead to a spectrum of releases are discussed and include sequences that can result in inadequate cooling of reactor fuel and to melting of the reactor core. The extent to which events arising from causes external to the plant which are considered possible contributors to the risk associated with the particular plant is also discussed.

The environmental consequences of releases whose probability of occurrence has been estimated are discussed in probabilistic terms. Such consequences are characterized in terms of potential radiological exposures to individuals, to population groups, and, where applicable, to biota. Health and safety risks that may be associated with exposures to people are discussed in a manner that fairly reflects the current state of knowledge regarding such risks.

The Reactor Safety Study

In 1975, the Reactor Safety Study (WASH-1400)[3] was published. The study concluded that the risk from reactor accidents was small, and that accidents more severe than the "maximum credible accident" (termed Class 9 accidents) dominate the risk. Such accidents involve not only core melt, but deterioration of the capability of the containment to limit the release of radioactive materials to the environment. This study was the first attempt to use quantitative assessment techniques to estimate the probabilities, source terms, and consequences associated with potential reactor accidents. Event tree and fault tree reliability assessment techniques were used to evolve the sequences of events that would be required to lead to core damage and to assess the probabilities associated with such sequences. In addition, models of the physical processes associated with the sequences were developed to assess the magnitudes and timings associated with the release, transport, and deposition of the radioactive materials from the core, through the primary system and the containment, to the environment. Consequence models were also developed to disperse the radioactive materials into the environment and assess the distributions of risk (probabilities and consequences) associated with such accidents.

Two specific reactor designs were analyzed in the WASH-1400: a 3-loop Westinghouse Pressurized Water Reactor (PWR) with a subatmospheric containment design (Surry); and a General Electric Boiling Water Reactor (BWR) with a Mark I design, vapor suppression containment (Peach Bottom). A spectrum of accident release characteristics (source terms) and associated probabilities was generated for each reactor. A detailed description of the accidents and associated processes are given in Appendices V, VI, VII and VIII of WASH-1400.[3]

Uncertainties in Reactor Safety Study

In 1977, the Nuclear Regulatory Commission sponsored an Independent Risk Assessment Review Group[20] to review the accomplishments and limitations of the Reactor Safety Study (RSS). The general conclusion that was reached was that the uncertainties associated with the absolute values used in WASH-1400 were large and caution should be applied whenever using the techniques, but that Probabilistic Risk Assessment (PRA) does and should have a place in the licensing process.

Significant research efforts have been ongoing to improve and formalize the risk assessment techniques developed in WASH-1400. All aspects of the work have undergone extensive study to determine sensitivities of various assumptions, and to revise approaches that were criticized. Improvements in data bases and in the experience in using the techniques has given greater levels of understanding to their application. Of dominant interest to this report is the improvements which have occurred in the modeling of the physical processes associated with the release. Significant progress has been made to improve the analytical methods of meltdown analysis and containment response over the RSS approaches. The Meltdown Accident Response Characteristics,[21] MARCH code, model has been developed to provide analyses of various thermal-hydraulic processes during reactor meltdown accidents. Additionally, the Containment of Radionuclides Released After LOCA,[22] CORRAL code, has been modified and generalized for use in these other studies.

The principal physical phenomena and accident parameters that are analyzed in the MARCH and CORRAL process are:

(a) The time scale of the accident, particularly the time for the start and completion of core melting, and releases to the containment.

(b) The time required for the molten core to fail the reactor vessel bottom head.

(c) Possible energetic interactions when the core debris fall to the floor of the reactor cavity, including the likelihood of containment failure due to such interactions.

(d) Long-term pressure-time history within the reactor containment, including the likelihood and time of containment failure due to overpressure.

(e) The probability and consequences of hydrogen burning or detonation within the containment building.

(f) The interaction of the core debris with the concrete foundation.

(g) The magnitude and timing of fission product release from the fuel to the containment atmosphere.

(h) The transport and removal of the various fission product species in the containment building atmosphere.

(i) Time-dependent leak rate from the containment building, including the airborne fission products.

Rebaselined Reactor Safety Study

In response to a request by the Commission to address the question of continued operation of the Indian Point Units 2 and 3, a task force was formed to compare the risks from Indian Point with other reactors.[23] As part of this effort the release categories of the Reactor Safety Study were reevaluated using current techniques (see Appendix B of NUREG-0715, Rebaselining of the RSS Results). Primarily, the rebaselined results reflect the use of the advanced accident process models (MARCH and CORRAL). In general, the changes decrease the release magnitudes of iodine isotopes and increase the release magnitudes of the cesium and tellurium isotopes. In addition to the physical process changes, the following modifications were made to the RSS results:

(a) Probability of reactor vessel steam explosion containment failure mode (α) was reduced by a factor of 100 to account for new research information;

(b) The "smoothing technique" between release categories was eliminated;

(c) Actual accident sequence characteristics were used to represent a release category instead of the synthesized conservative representation of the group of accident sequence categories.

(d) Erroneous RSS assumptions concerning some key sequences (e.g., TCγ') were corrected.

<u>Probabilistic Risk Assessments (Post WASH-1400)</u>

Since the completion of the Reactor Safety Study (RSS), additional reactor risk assessments have been made under the direction of the Nuclear Regulatory Commission's Office of Research. It was recognized that the two designs analyzed in the RSS did not necessarily represent the various reactor power plant designs in operation and construction. Therefore, a research program was begun to analyze other significant variations in design. This program, titled "Reactor Safety Study Methodology Application Program," RSSMAP, was designed to analyze a plant using the insights from the RSS on the dominant accident sequences, the RSS system fault and event tree analyses. Four reactors were analyzed: Sequoyah, a Westinghouse 4-loop PWR with ice condenser containment; Oconee, a Babcock and Wilcox 2-loop PWR with dry containment; Calvert Cliffs, a Combustion Engineering 2-loop PWR with dry containment; and Grand Gulf, a General Electric BWR with Mark III containment.

In addition to RSSMAP, another research program, the Interim Reliability Evaluation Program (IREP), was begun to develop a workable approach toward evaluating the reliability of power plants as part of the regulatory process. The first plant analyzed in that program was Crystal River 3, a Babcock and Wilcox 2-loop PWR with dry containment. Four additional plants: Millstone 1; Arkansas 1; Calvert Cliffs; and Browns Ferry are currently being analyzed.

In addition to the studies sponsored by the NRC, other power plants have been analyzed using probabilistic risk assessment techniques. For example, the German Reactor Safety Study analyzed the Biblis-B PWR. Diablo Canyon was analyzed to assess the seismic vulnerabilities. Recently Indian Point, Zion, and Limerick were ordered by the NRC to perform risk assessments because of their high population densities.

C.4 TMI-2 EXPERIENCE

The Lessons Learned Task Force[24,25] was established to identify and evaluate those safety concerns originating with TMI-2 accident that require licensing actions.

The recommendations were intended to constitute a set of short-term of proposed changes and/or modifications to ensure the safety of plants already licensed to operate, and those to be licensed for operation in the future. The fission product release assumptions and their consequences are recognized as direct impact from the TMI-2.

The fission products assumptions and recommendations are:

(a) Several of the engineered safety features (ESF) and auxiliary systems external to the containment building that would contain radioactive material showed some imperfections concerning operational leakage characteristics, and shielding provisions (2.1.6). Some of these systems which were used during the accident experienced releases of radioactive materials to the auxiliary building ventilation systems. These releases are believed to have resulted from leaking of various systems, mainly through relief valves, seals, and process valves, but the leakage rate was not known. Lessons Learned recommended that a program should be implemented to reduce leakage from systems outside containment. The leakage rate tests should be performed on systems that process primary coolant and could contain high-level radioactive materials. The design review of the shielding of systems processing primary coolant outside containment should also be implemented to protect areas, or equipment, that are vital for post-accident occupancy and operation. The radiation source terms for the purpose of conducting the shielding review may approximate those of Regulatory Guides 1.3 and 1.4.

(b) Special features and instruments to be provided to aid in accident diagnosis and control (2.1.8). A design and operational review of the radiological analysis should be performed to determine the capability to promptly quantify, in less than 2 hours, radioisotopes that are indicators of the degree of core damage, such as noble gases, iodines and cesiums. The accident conditions should assume a release of fission products as provided in Regulatory Guidse 1.3 and 1.4. This should include improvement of post-accident sampling capability to obtain samples from the reactor coolant system and containmant atmosphere under high radioactivity conditions, increase high range radiation monitors for noble gases in effluent lines and in the containment, and provide instrumentation for determining in-plant airborne radioiodine concentration.

The long-term recommendations are concentrated on evaluation of the TMI-2 accident by considering broader and more fundamental aspects of the design and operation of nuclear power plants, and associated licensing process.

From the potential release of fission products a central issue, as it was stressed out in the document, will be the evaluation and possibly a modification of current design basis events, or even depart from the existing concept. Analysis of design basis accidents could be modified to include core uncovery situation or core melting core. As it was pointed out, extensive core damage and, as a consequence, release of fission products, and production of a large quantity of hydrogen from the reaction of zircalloy cladding and steam were excluded from the design basis, since plant safety features are designed and provided to prevent such occurrences.

The Action Plan[26] was developed as a comprehensive and integrated plan for the actions which were judged necessary by the Nuclear Regulatory Commission, to correct or improve the regulation based on the experience gained from the accident at TMI-2, and as the results of official studies and investigations of the accident.

Action Plan items related to fission product release issue are:

(a) To review plant shielding providing access to vital areas and protect safety equipment for post-accident operation (II.B.2). The intention of this action is to ensure that certain facilities, equipments and compartments, under post-accident conditions that may contain abnormally high levels of radioactivity, will not be degraded and be fully operable to mitigate the accident consequences. The evaluation of the equipment should include personnel protection equipment (e.g., respirators).

(b) To provide post-accident sampling (II.B.3) of the reactor coolant and containment atmosphere to analyze concentration of radioactive noble gases, iodines, cesiums, and nonvolatile isotopes withing 2 hours after the accident.

(c) To reduce risk for operating reactors at sites with high population densities (II.B.6) by conducting a review to evaluate the consideration of severe accident mitigation features such as filtered containment venting and core retention systems. Design studies should be performed to determine if the above features, or combination of them, could be employed to mitigate the effects of core degradation and core melt accidents.

(d) To provide instrumentation to monitor variables during and following an accident (II.F.A) for obtaining information to: (1) determine the potential for a breach of the barriers to radioactivity release (i.e., fuel cladding, reactor coolant pressure boundary, and containment) and if a barrier has been breached, and (2) allow for early indication of the need to initiate action necessary to protect the public, and for an estimate of the magnitude of the impending threat. Additional accident monitoring instrumentation should be provided to measure containment radiation intensity (high range, high-range noble gas effluents for each potential release point, including PWR steam safety and atmospheric-steam-dump valves), and accident monitoring system with expanded ranges to cover source term that considers a damaged core.

(e) To improve the reliability and capability of nuclear power plant containment structure to reduce the radiological consequences from design basis events, degraded core and core melt accidents (II.E.4) by imposing additional limits on containment purging and venting to reduce potential accidental releases of iodine, and other fission products.

(f) To promptly improve and upgrade emergency preparedness by requiring improvements to adequately respond to and manage an accident.

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16. ABSTRACT (200 words or less)

This report addresses the reactor accident source term implications on accident evaluations, regulations and regulatory requirements, engineered safety features, emergency planning, probabilistic risk assessment, and licensing practice. Assessment of the impact of source term modifications and evaluation of the effects in Design Basis Accident analyses, assuming a change of the chemical form of iodine from elemental to cesium iodide, has been provided. Engineered safety features used in current LWR designs are found to be effective for all postulated combinations of iodine source terms under DBA conditions. In terms of potential accident consequences, it is not expected that the difference in chemical form between elemental iodine and cesium iodide would be significant.
In order to account for the current information on source terms, a spectrum of accident scenerios is discussed to realistically estimate the source terms resulting from a range of potential accident conditions.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

Accident Source Terms
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17b. IDENTIFIERS/OPEN-ENDED TERMS

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