June 30, 1998 SECY-98-154

FOR: The Commissioners

FROM: L. Joseph Callan /s/
Executive Director for Operations

SUBJECT: RESULTS OF THE REVISED (NUREG-1465) SOURCE TERM REBASELINING FOR OPERATING REACTORS

PURPOSE:

To provide the Commission with the results and findings of an evaluation of the impact of implementing the revised source term described in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," for operating reactors.

BACKGROUND:

The Commission's reactor site criteria, 10 CFR Part 100, requires that a fission product release into containment be postulated and that offsite radiological consequences be evaluated against the guideline dose values given in Part 100. Other Commission regulations, in 10 CFR Part 50, GDC 19, address regulatory requirements on the accident radiological doses for the control room. The evaluation of the release of fission products into containment (called "source term") is used for judging the acceptability of both the plant site and the effectiveness of engineered safety features. The original source term, which was based on releases from a severely damaged core, was published in 1962 by the U.S. Atomic Energy Commission in Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactors." Since that time there have been significant advances in our understanding of the timing, magnitude and chemical forms of the fission product release from severe reactor accidents. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," was published in February 1995, and reflects that extensive research and experience culminating in the development of a new or revised source term. The development of the revised source term was originally intended for initial application to advanced reactors though it was recognized that

Contact: C. Tinkler, AEB/DST/RES
415-6770
J. Schaperow, AEB/DST/RES
415-5907

current reactors may want to utilize the revised source term in licensing actions. The impetus for operating reactors to adopt the revised source term is that through its more realistic characterization of the source term, plants may modify existing restrictive plant features, (e.g., component actuation times, leakage control systems).

The staff proposed, in SECY-96-242, “Use of the NUREG-1465 Source Term at Operating Reactors,” dated November 25, 1996, an approach to allow use of the revised source term for operating plants. As part of its approach the staff indicated its plans to undertake a comprehensive assessment, referred to as rebaselining, of two plants to further evaluate issues attendant with applying the revised source term to operating plants. The staff also indicated its plans to undertake the review of five pilot plant submittals that address a wide range of revised source term applications.

In an SRM dated February 12, 1997, the Commission approved the staff’s plan outlined in SECY-96-242 and directed the staff to commence rulemaking upon completion of the source term rebaselining and concurrent with the pilot plant evaluations. The Commission further stipulated that implementation of the NUREG-1465 source term at operating reactors should include the revised Part 20 dose methodology (total effective dose equivalent or TEDE criterion) and should include consideration of the dose over the worst two hour interval after the accident. In a memorandum to the Commission dated September 9, 1997, the staff provided the status of its plans to implement the revised source term at operating reactors and outlined a schedule of November 30, 1998, for completion of rebaselining analyses and the preparation of a rulemaking plan. Subsequently, the staff shifted its assignment of responsibilities on rebaselining to expedite the completion of those tasks to June 1998. In addition, a joint RES/NRR management committee was formed to meet regularly to oversee progress on rebaselining activities.

DISCUSSION:

The objective of the rebaselining effort was to develop a better understanding of the impacts of implementing the revised source term (NUREG-1465) at operating reactors. The major areas examined were:

- i) effect on calculation of individual offsite and control room dose
- ii) effect on calculation of dose for equipment qualification
- iii) effect of potential plant modifications, including assessment of changes to plant risk

It has been recognized since the development of the revised source term that changes in the prescription of the source term from that originally described in TID-14844 would influence the major areas of dose analysis and could prompt plant, technical specification and procedure modifications. The most significant changes in the source term are the treatment of the fission product release as a time dependent process and the release of radioiodine primarily as an aerosol. In the revised source term the fission product release is assumed to occur over roughly two hours as opposed to the TID source term which assumed the release of the entire source term occurs instantaneously at time zero. In addition, in the revised source term, 95% of the radioiodine is assumed to be released as an aerosol, CsI, with the remaining 5% as a combination of inorganic and organic vapors. This is in contrast to the original TID source term which prescribed the opposite ratio, 95% of the iodine as vapor and 5% as aerosol. Therefore, plant systems originally provided to mitigate an instantaneous source term by very rapid actuation would not be required to perform under such stringent requirements with the revised source term. Likewise, systems needed to remove iodine vapors are less important under conditions where iodine is an aerosol. As part of rebaselining these issues and a number of other more subtle differences between the source terms, as well as the impact of improved modeling of fission product processes, were explored to position the staff for review of pilot plants and rulemaking.

At the outset it was planned to study the impacts in depth for two plants, Grand Gulf and Surry, a BWR and PWR respectively. These plants were chosen to represent broad classes of reactors; they also represented plants which had been studied under NUREG-1150 and for which the risk profile had been carefully examined. In addition, in the early planning of this activity these plants' participation in providing detailed site data and information was important to the completion of a thorough examination. Shortly after initiation of the rebaselining project, the set of analyses was expanded to include calculations for the Zion plant, which is representative of a large class of PWR dry containment reactors. Rebaselining was the vehicle for assessment of the likely dominant issues associated with implementation of the revised source term as revealed by analysis of representative plants and as such provides the technical bases for development of regulatory guide criteria.

In the formulation of the rebaselining initiative the activities were divided into four phases which allowed for progression of analyses and insights to be factored into ongoing assessments. Phase I addressed DBA dose calculations using both the TID and NUREG-1465 source terms to evaluate the impact of the revised source term on individual dose. Analyses addressed a range of design basis accidents including a loss of coolant accident, fuel handling, main steam line break and rod drop accidents. Calculations were performed to determine the accident dose at the exclusion area boundary (EAB), low population zone (LPZ) boundary and for the control room. The approach taken in Phase I was to perform calculations as they were conducted as part of the staff's licensing review documented in the safety evaluation report.

In Phase II, similar calculations were performed as in Phase I, but, in this phase, calculations were performed using the approach and methodologies adopted by the licensees in their analyses, as described in the Final Safety Analysis Report (FSAR). Since the licensees often utilize different assumptions and techniques, these calculations were undertaken to examine the impact of the new source term under conditions representative of licensee analyses. In Phase II, the staff also performed equipment qualification dose calculations to assess the impact of the revised source term. In both Phases I and II, in addition to evaluating the impact of the revised source term versus the TID source term, the rebaselining analyses also addressed use of the revised dose acceptance criterion of 25 rem TEDE and use of the worst 2 hour interval for dose analysis.

In Phase III, the staff undertook a number of diverse analyses to investigate, in detail, technical issues associated with implementation of the revised source term. In conjunction with the development of the revised source term, the staff also reevaluated the modeling of fission product removal mechanisms associated with containment sprays and suppression pools. As a result of a previous effort, new models for fission product removal were developed, with the intent of quantifying uncertainty as well as representing current understanding. In Phase III, DBA dose calculations were repeated for the revised source term using the existing Standard Review Plan (SRP) treatment of the removal mechanisms and with the updated models. Also, equipment qualification dose calculations were repeated with both Regulatory Guide 1.89 and updated models. In addition to the DBA dose calculations, analyses were also performed using the MELCOR code, an integrated severe accident code which is used to calculate both thermal hydraulic and fission product behavior in the reactor and containment. MELCOR was used to perform a best estimate assessment of offsite doses to provide insights into the margins still inherent in the revised source term methodology. Because of MELCOR's capabilities, it was also used to investigate the consistency and margins between the treatment of thermal hydraulic conditions in the DBA dose calculations and those calculated for a postulated core damage accident with recovery of emergency core cooling systems (ECCS). Another technical issue examined under Phase III was the revaporization of iodine from the sump or suppression pool. It is known that radiolysis in acidic water pools can cause revaporization of dissolved iodine. Inherent to the relatively low vapor concentrations of iodine prescribed in NUREG-1465 is the assumption that sump/suppression pool pH is controlled at a value of 7 or greater to effectively inhibit the revaporization of dissolved iodine. Many PWRs already control pH of the post accident water inventory in containment. Analysis was performed to examine the extent to which pH control would be needed for BWRs.

In Phase IV the staff focussed on assessment of potential plant changes that may be proposed in conjunction with implementation of the revised source term. Dose calculations were performed with specific changes proposed in the pilot plants applications. These plant modification applications were submitted in conjunction with submittal of the generic industry proposal, EPRI Technical Report TR-105909, "Generic Framework for Application of Revised Accident Source Term to Operating Plants," November 1995. In Phase IV, the staff also factored in an evaluation of the risk impacts of potential plant modifications. A study of such risk impacts was conducted for the NUREG-1150 plants considering the risk impacts from relaxation of requirements related to design containment leak rate, containment spray operation, reactor building drawdown, subatmospheric containment operation and filtration systems.

A more detailed discussion of the rebaselining analyses, which involved a large number of calculations and assessments is provided in the attachment along with a more detailed description of the results. A summary of the general results is provided below.

<u>RESULTS</u>:

The overall impact of implementing the revised source term in the majority of cases is to produce lower calculated doses, ranging from a slight reduction up to an order of magnitude decrease, for an individual, whether for the EAB, LPZ or control room. In addition, in assessing the impact of implementing the revised source term and comparing new calculated doses against earlier (and occasionally much older) analyses, it was confirmed that changes in the calculated dose may also occur for reasons not directly related to the source term itself. For example, in older calculations, the dose to an individual was calculated using dose conversion factors taken from International Commission on Radiation Protection (ICRP) Publication 2. Current analyses including those implementing the revised source term would use updated dose conversion factors taken from EPA Federal Guidance Reports (FGR) 11 and 12. The use of updated dose conversion factors alone will produce reductions in the dose by up to 40%.

The extent of further reduction in calculated doses is also influenced by several factors, two of which are connected to differences between the TID and revised source term. As noted previously, the revised source term treats the release of fission products as a time dependent release, thus reduction of doses will be strongly influenced by safety features which are timing sensitive. In the case of Surry, a subatmospheric design, all containment leakage ceases after one hour, thus with the revised source term (released over 1.8 hrs) only a fraction of the source term is available for leakage (approximately 1/4 of the eventual release). Similarly, if dose mitigation is provided by a standby gas treatment system (e.g., Grand Gulf) which filters all releases after several minutes, then the dose release during the period of unfiltered release is relatively inconsequential using the revised source term. In the area of chemical form of radionuclides, cited previously, in the case of Zion, the TID dose calculation is heavily influenced by the assumption of a large release of iodine as organic vapor. Treatment of iodine primarily as an aerosol in the revised source term resulted in a substantial reduction of the dose. Lastly, comparing calculated doses from different analyses (SER and FSAR) which utilize different assumptions occasionally produces observed changes in doses which are driven by modeling assumptions which are plant specific.

Another finding from the rebaselining activity was that the time dependent release of the new source term, coupled with plant characteristics, can result in a substantial shift in the 2 hour interval associated with the maximum dose. In Grand Gulf LOCA dose analyses, the worst two hour interval for the TEDE dose of 6.8 rem began at 2.2 hrs. By comparison the calculated dose for the first two hours was 2.0 rem.

An evaluation of the equipment qualification dose using the TID and NUREG-1465 source terms revealed that the containment atmosphere dose using the revised source term was similar since most of the dose is from noble gases, for which the two source terms are identical. With respect to doses for equipment exposed to containment sump water, the revised source term again produces similar results. However, the revised source term produces somewhat higher doses later in time (after approximately 1 week in the case of the Surry analysis) due to a much higher inventory of cesium. These results confirm the overall trend of an assessment previously performed, (Memorandum to the Commission dated February 23, 1993, from J.M. Taylor, “Impact of New Source Term on Safety Related Equipment”). The significance of any higher dose in the containment sump will be considered in the pilot plant reviews.

In Phase III, analyses were also performed with the MELCOR code both to evaluate the extent of margins maintained with the revised source term calculations and to evaluate related thermal hydraulic issues. The MELCOR analyses indicated that the DBA dose calculations still have substantial margin (a factor of 2 or greater) even though the dose may be well below the earlier TID analysis. This margin often stems from the integral coupling of dose analysis and thermal hydraulic analysis in the MELCOR calculations. In the DBA dose analysis, a constant containment leak rate, associated with leakage at the peak containment pressure, is assumed for the 24 hours after the accident, whereas the MELCOR analysis varied leakage in accordance with the predicted containment pressure transient. Additionally, best estimate treatment of fission product removal in MELCOR yields a further reduction in doses against the updated DBA models.

In Phase IV, an evaluation of potential plant changes was undertaken with the objective of assessing the impact on the DBA dose calculation and the impacts on plant risk. The general conclusion from these studies was that indeed many of the types of changes proposed could be made and the DBA dose would remain within acceptance limits, though the plant specific changes will need to be reviewed. Furthermore, analysis indicated that since most of the systems being contemplated for modification are not involved in risk significant sequences their modification is not likely to have any substantial offsite risk impact, using a measure such as large early release frequency, as established in Regulatory Guide 1.174.

Having concluded the rebaselining initiative, the staff did not identify any issues that would prevent implementation of the revised source term at operating reactors. Further, the rebaselining activities have provided a technical basis to support rulemaking and changes to associated regulatory guides. The staff is therefore proceeding in accordance with Commission direction, in the SRM dated February 12, 1997, to commence rulemaking. A rulemaking plan is being submitted to the Commission by a separate paper. The staff is also proceeding with the evaluation of implementation of the revised source term for the pilot plant applications.

L. Joseph Callan
Executive Director
for Operations

Attachment: As stated

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Fission Product Source Term Rebaselining

Introduction

The assessment of a nuclear plant response to Design Basis Accidents (DBAs) includes the analysis of plant and offsite consequences of a substantial release of fission products to the containment as a result of a postulated accident involving significant damage to the reactor core. Historically, regulatory guidance on the release of fission products into containment, or source term, has been based on TID-14884, “Calculation of Distance Factors for Power and Test Reactors,” published by the U.S. Atomic Energy Commission in 1962. The source term described in this document has been used to evaluate compliance with the reactor site criteria in 10 CFR Part 100, and the regulatory requirements for the plant described in 10 CFR Part 50. Additional regulatory guidance for application of the TID source term is found in Regulatory Guides 1.3 and 1.4 and NUREG-0800, the Standard Review Plan.

The TID source term is an instantaneous release of noble gases and iodine, which is assumed to be predominantly in elemental vapor form. For equipment qualification, additional nuclides of lower volatile species, in aerosol form, are assumed. The existing regulatory framework for operating reactors prescribes the calculation of offsite doses at the exclusion area boundary (EAB) for the first two hours after the accident and at the low population zone (LPZ) boundary for the course of the accident, which for dose analysis is assumed to continue for 30 days. Control room doses are also calculated for a 30 day period. Although the existing requirements involve the calculation of whole body and thyroid doses, in practice the thyroid dose, resulting from the iodine, has generally been the limiting dose for plants.

The development of a revised source term described in NUREG-1465, “Accident Source Terms for Light Water Nuclear Power Plants,” which was published in February 1995, represents the substantial improvement in our understanding of fission product releases from severe nuclear power plant accidents garnered over the past three decades since the prescription of the TID source term. A general comparison of the TID and NUREG-1465 source terms is provided in Table 1. The NUREG-1465 source term depicted is that for a PWR (BWR releases are similar); the TID source term did not distinguish between PWRs and BWRs. Additionally, implementation of the revised source term is limited to the release from the first three phases of a severe accident (coolant, gap and early in-vessel) described in NUREG-1465, and which are appropriate for DBA evaluations. An examination of the two source terms indicates that the two source terms are comparable in magnitude, with the notable exception of cesium which is substantially higher in the revised source term. The revised source term is markedly different in that there is a more mechanistic and realistic treatment of the rate of release of fission products and their chemical and physical form. The revised source term is a time dependent release of fission products with nuclides, other than noble gases, primarily in aerosol form.

	TID-14844 Source Term	NUREG-1465 Source Term

Core fractions released into containment	Noble gases - 100% Iodine - 50% (with half of this 50% plated out) Solids - 1%	Noble gases - 100% Iodine - 40% Cesium - 30% Tellurium - 5% Barium - 2% Others - .02% to .2%
Rate of release	Released instantaneously	Released over 1.8 hours
Iodine chemical and physical form	91% inorganic vapor 4% organic vapor 5% aerosol	4.85% inorganic vapor .15% organic vapor 95% aerosol
Solids	ignored in offsite and control room dose assessment	treated as an aerosol

Table 1. Comparison of TID-14844 and NUREG-1465 Source Terms for a PWR

In conjunction with the implementation of the revised source term, the staff is considering new accompanying regulatory criteria addressing dose acceptance limits. In place of dose acceptance limits of 300 rem thyroid and 25 rem whole body, the staff has proposed to use the total effective dose equivalent (TEDE) methodology in 10 CFR Part 20 with a limit of 25 rem TEDE for offsite doses and 5 rem TEDE for control room doses. Implementation of the revised source term will also be accompanied by the related requirement that doses for the EAB be calculated over the worst two hour period.

In addition to developing a more realistic source term, recent research has been directed at development of updated models for fission product removal mechanisms including models for fission product removal by containment sprays and suppression pools. These updated models described in NUREG/CR-5966, “A Simplified Model of Aerosol Removal by Containment Sprays,” and NUREG/CR-6153, “A Simplified Model of Decontamination by BWR Steam Suppression Pools,” are intended to serve as part of the staff’s supporting guidance in implementing the revised source term for operating reactors.

In order to more fully evaluate issues associated with implementation of the revised source term at operating reactors, including assessment of the impact of revised dose acceptance criteria, the staff undertook a systematic evaluation of dose analyses for a range of accidents at representative plants. This evaluation, termed rebaselining, was intended to explore significant trends, evaluate generic implications, and where possible, identify types of plant specific dependencies. The principal technical issues of rebaselining concerned the effect of the revised source term on calculation of individual offsite (and control room) doses, the effect on the dose calculation used for equipment qualification and the impact of potential plant changes both on dose analyses and on severe accident risk.

In implementing the rebaselining activity, the work was broken down into four phases, which divided the work according to the types of analyses and allowed insights gained to be incorporated into subsequent phases. In Phase I, analyses were performed to calculate DBA doses using both the TID and revised source terms to assess the impact of the revised source term (and newer dose conversion factors) on individual dose. Calculations were performed for a spectrum of design based accidents including loss of coolant, fuel handling, main steam line break and rod drop accidents. Analyses in Phase I were performed using the analytical assumptions employed in the staff’s confirmatory analyses and documented in the safety evaluation report.

Analyses of the main steam line break accident do not specifically involve the revised source term, however calculations were performed in this case to assess the revised dose acceptance criteria. Also, in the case of the fuel handling accident, the entire TID and NUREG-1465 source terms are not involved but rather the fission product release is limited to gap activity.

Phase II addressed the same types of calculations as Phase I (i.e., DBA dose calculations) but in Phase II the calculations were performed using the analytical models and assumptions employed by the licensees in their analyses described in the FSAR. The intent here was to evaluate the representative impact of the revised source term using industry methods to determine if different impacts would be seen. The staff also included in Phase II equipment qualification dose calculations using both the TID and NUREG-1465 source terms. As noted previously, the staff, as part of implementing the revised source term, will apply revised dose acceptance criteria (TEDE methodology) and a modification to the two hour interval for determining EAB doses. Rebaselining in Phases I and II also explored the impact of those revisions.

In Phase III, the staff addressed the impact of updated models for fission product removal mechanisms by comparison with calculations using equivalent SRP models. (Phase I calculations using SER assumptions employed treatments which in some areas predate the SRP methods). All Phase III analyses utilized the revised source term. Phase III analyses also considered the calculation of doses using the MELCOR severe accident code. MELCOR, which is an integrated code containing models for thermal hydraulic and fission product behavior, was used to assess margins in the DBA dose calculation. As part of Phase III, a study was also undertaken to examine the long term revolatilization of iodine from the containment sump. As noted, the revised source term prescribes a relatively modest amount of iodine in vapor form. A condition which, in theory, could challenge the validity of this assumption is the long term revolatilization of iodine from water pools which are or have become acidic following an accident. Analyses were performed then to determine the extent of revolatilization of iodine, using the TRENDS code, a containment water chemistry code. These analyses predict both the transient pH of the sump/suppression pool and the amount of iodine revolatilization.

In Phase IV, a study was conducted to assess potential plant changes that may be feasible with the use of the revised source term. Some specific plant modifications have been proposed by pilot plants in conjunction with a generic industry proposal described in EPRI Technical Report, TR-105909, “Generic Framework for Application of Revised Source Term to Operating Plants,” November 1995. In addition to the specific plant modifications described in pilot plant submittals, the staff also evaluated other generic candidates such as containment leak rates. Finally, modification of plant systems, while acceptable from a design basis regulatory standpoint, may have severe accident risk impacts. Therefore, as part of Phase IV, we included the results of a study to evaluate severe accident risk impacts of potential plant changes which may arise as a result of implementing the revised source term.

Phase I

The objective of Phase I was to determine the effect on individual dose based on SER assumptions of substituting the NUREG-1465 source term for the TID source term and the effect of the new dose acceptance criteria. To achieve this objective, Phase I involved calculating offsite and control room DBA doses with SER assumptions using both the TID and NUREG-1465 source terms. The SER assumptions are the assumptions used by the NRC in its independent analysis of offsite and control room doses.

Based on a review of SERs for Surry and Grand Gulf, a set of accidents were identified for which doses would be calculated. This set of accidents, a subset of the accidents evaluated in Chapter 15 of the Standard Review Plan, is shown in Table 2. The main steam line break accident, which is not affected by the NUREG-

1465 source term, was chosen to assess the impact of the new dose acceptance criteria. The assumptions in the most recent SER for each of these accidents were used in the rebaselining dose calculations. Also, the current and proposed dose limits for these accidents are given in Table 3.

Plant	Accident	Fission Product Release	Dose Location
Surry	LOCA	gap and in-vessel	EAB/LPZ/Control Room
	Fuel Handling	gap	EAB/LPZ
	Main Steam Line Break	iodine spike	EAB/LPZ
Grand Gulf	LOCA	gap and in-vessel	EAB/LPZ/Control Room
	Fuel Handling	gap	EAB
	Rod Drop	gap and in-vessel	EAB/LPZ

Table 2. Accidents Evaluated in Phase I of Rebaselining

In Phase I, dose calculations were first performed using SER assumptions with the TID source term. In the LOCA dose calculations, there are two types of leakage to the environment. The first type is leakage of nuclides suspended in the containment air (called containment leakage). The second type is leakage of nuclides suspended in the sump water through Emergency Core Cooling System (ECCS) components located outside of containment during the recirculation phase. In the LOCA dose calculation, the TID source term is 100% of the core inventory of noble gases and 50% of the iodine (half of which is assumed to instantaneously plate out) for releases to the containment atmosphere and 50% of the iodine for releases to the containment sump. The results of these calculations were compared with results reported in the SERs to ensure that the assumptions used in the original SER analysis are captured in the rebaselining study.

Accident	Dose Location	Current Dose Limit	Proposed Dose Limit
LOCA	EAB	300 rem thyroid, 25 rem whole body, first two hours	25 rem TEDE, worst two hours
	LPZ	300 rem thyroid, 25 rem whole body, first 30 days	25 rem TEDE, first 30 days
Fuel Handling	EAB/LPZ	25% of current LOCA dose limit	25% of proposed LOCA dose limit
Main Steam Line Break - Pre-existing Iodine Spike -Accident-Initiated Iodine Spike	EAB/LPZ EAB/LPZ	100% of current LOCA dose limit 10% of current LOCA dose limit	100% of proposed LOCA dose limit 10% of proposed LOCA dose limit
Rod Drop	EAB/LPZ	25% of current LOCA dose limit	25% of proposed LOCA dose limit

All Accidents	Control Room	30 rem thyroid, 5 rem whole body, first 30 days	5 rem TEDE, first 30 days

Table 3. Current and Proposed Dose Limits

Dose calculations were then performed using SER assumptions with the NUREG-1465 source term. In the LOCA dose calculations, the NUREG-1465 source term is the gap and early in-vessel releases of NUREG-1465 for releases to the containment atmosphere and for releases to the containment sump. The results of these calculations were compared with the results of the dose calculations with the TID source term. With the exception of the main steam line break accident, the limiting DBA analyses included as part of the rebaselining study do not include coolant activity since it is a small contribution to doses. The results are given below for each DBA.

Surry DBA Dose Analysis

LOCA (Offsite Doses)

The SER analysis considered the offsite dose due to containment leakage. For the offsite dose, leakage of the ECCS during the recirculation phase was not included in the SER analysis. In many earlier analyses, ECCS leakage was considered to be a small contributor to offsite dose, in part due to the magnitude and conservatism in the containment leakage dose calculations coupled with the overall conservatism of the TID source term. The containment was assumed to leak at its design rate (.1%/day) until pressure fell below 14.7 psia which was assumed to occur at one hour into the accident. The results of these calculations are shown in Table 4.

Source Term/ Dose Conversion Factors	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
TID-14844/ ICRP 2	232	3.09	NA	20.3	.270	NA
NUREG-1465/ ICRP 2	66	.48	NA	5.7	.042	NA
NUREG-1465/ FGR 11& 12	39	.39	3.7	3.4	.034	.32

Table 4. Surry LOCA EAB and LPZ Doses - Phase I

The original SER dose calculations with TID were performed with dose conversion factors based on ICRP Publication 2. The rebaselining dose calculations with NUREG-1465 were performed with dose conversion factors in Federal Guidance Reports 11 and 12, which are the most recent dose conversion factors that the staff plans to use. Additional dose calculations were performed with NUREG-1465 with dose conversion factors based on ICRP Publication 2 to allow evaluation of the dose changes resulting solely from changing from ICRP Publication 2 to Federal Guidance Reports 11 and 12. The results of these additional calculations are also shown in Table 4. The results show a significant dose reduction as a result of the updated dose conversion factors and related nuclide data.

Because containment leakage stops after one hour (subatmospheric containment), the worst two hours are the first two hours. Also, the non-instantaneous NUREG-1465 source term coupled with no leakage after one hour resulted in a lower dose than with the TID source term. A comparison of the iodine release of the TID and NUREG-1465 source terms used in the Surry LOCA analysis is shown in Figure 1. Figure 1 shows the time-dependent nature of the NUREG-1465 source term. For the first half hour of the accident, the gap inventory of iodine is released at a constant rate up to a total amount of 5% of the core inventory. For the next 1.3 hours, the in-vessel release occurs at a constant rate up to a total amount of 40% of core inventory. The other nuclides in the NUREG-1465 source term are also released at constant rates. The Surry subatmospheric containment leaks at a constant rate of .1%/day for one hour. After one hour the containment becomes subatmospheric and all outward leakage ends. Because of the constant leak rate, the offsite and control room doses from iodine (i.e., thyroid doses) are proportional to the areas under the curves in Figure 1.

<u>LOCA (Control Room Doses)</u>

The SER assumed both containment leakage and ECCS leakage contributed to control room dose. Again, the containment was assumed to leak at its design rate (.1%/day) until pressure fell below 14.7 psia which was assumed to occur at one hour into the accident.

The results of these calculations are shown in Table 5. A small removal rate for particulate iodine was assumed in the SER. Although this had a small impact on the containment leakage dose with the TID source term (5% particulate), it has a significant impact on the NUREG-1465 source term (95% particulate). The reduction in thyroid dose from no containment leakage after one hour is offset by the unrealistically low removal rate assumed for spray removal of particulate iodine.

Source Term/ Dose Conversion Factors	Control Room		
	Thyroid	Whole Body	TEDE
TID-14844/ ICRP 2	30.6	.59	NA
NUREG-1465/ ICRP 2	31.7	.13	NA
NUREG-1465/ FGR 11& 12	19.4	.11	.82

Table 5. Surry LOCA Control Room Doses (rem) - Phase I

<u>Fuel Handling Accident</u>

The SER assumed that, as a result of a dropping a fuel assembly, the fuel rods in one assembly broke and released their gap inventory. The gap inventory the SER used was 10% of the inventory of noble gases and iodine. The gap inventory used for the rebaselining calculation with the NUREG-1465 source term was 3% of the core inventory of noble gases and iodine. As shown in Table 6, the smaller gap release of NUREG-1465 results in a proportionally smaller dose.

Source Term/ Dose Conversion Factors	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
TID-14844/ ICRP 2	27.8	.62	NA	2.43	.054	NA
NUREG-1465/ ICRP 2	8.5	.18	NA	.74	.015	NA
NUREG-1465/ FGR 11& 12	6.1	.11	.30	.53	.009	.026

Table 6. Surry Fuel Handling Accident Doses (rem) - Phase I

Main Steam Line Break Accident

Doses for the main steam line break accident are calculated in two cases. The first case is that of a pre-existing iodine spike; that is, the accident is assumed to occur following another event that resulted in an elevated iodine concentration in the reactor coolant system. In this case, the SER assumes that the reactor coolant system iodine concentration is 60 µCi/g when the accident occurs. The second case is that of an accident-induced spike. When an accident occurs, it is typically accompanied by a power or pressure transient that produces an increase in the iodine concentration or spike. In this case, the SER assumes that the iodine concentration is 1 µCi/g prior to the accident and increases at the rate of 7400 Ci/hr for four hours (release rate of iodine from the fuel is generally assumed to be 500 times greater than the rate to sustain 1 µCi/g).

The NUREG-1465 source term addresses accidents where cladding fails and significant fuel damage occurs. The main steam line break accident is an accident involving only coolant activity (iodine). Therefore the NUREG-1465 source term does not apply in the main steam line break accident dose assessment. Also, the SER does not provide the assumptions for the reactor coolant system noble gas concentrations needed for the calculation of whole body and TEDE doses. Therefore, the Phase I rebaselining analysis of the main steam line break accident was limited to an evaluation of the worst two hours criterion and a comparison of thyroid doses with ICRP Publication 2 and Federal Guidance Reports 11 and 12 dose conversion factors. Use of the TEDE dose criteria was investigated in the Phase II dose assessment with FSAR assumptions where assumptions for reactor coolant system noble gas concentrations are available. The results of the Phase I analysis of the main steam line break accident given in Table 7 show smaller doses for the first two hours with the Federal Guidance Report 11 and 12 dose conversion factors. For the accident-initiated spike case, the worst two hours dose which occurs from two to four hours is much higher than the first two hours dose. (For this document, when the worst two hours doses differ from the first two hours doses, the start time of the worst two hours is included in parenthesis immediately following the worst two hours dose.)

Source Term/Dose Conversion Factors	EAB	LPZ

	Thyroid	Thyroid
Pre-existing spike/ICRP 2	2.44	.371
Pre-existing Spike/FGR 11	1.81	.275
Accident-Initiated Spike/ICRP 2	3.54 11.8 (2h)	1.92
Accident-Initiated Spike/FGR 11	2.62 8.72 (2h)	1.43

Table 7. Surry Main Steam Line Break Accident Doses (rem) - Phase I

Grand Gulf DBA Dose Analysis

LOCA

Grand Gulf has a Mark III containment where the reactor is located in a drywell which is surrounded by a wetwell, which is in turn surrounded by auxiliary and enclosure buildings. During a LOCA, the Standby Gas Treatment System (SGTS) is activated, which is assumed to drawdown the auxiliary and enclosure buildings to a negative pressure in two minutes after which all leakage from the drywell and wetwell to the environment is assumed mixed in the enclosure building and filtered by the SGTS. The SER assumed the source term was homogeneously mixed throughout the drywell and wetwell. The wetwell was assumed to leak at its design rate (.35%/day) for the duration of the accident. Drywell leakage is assumed to bypass the wetwell at a rate of .21% of the drywell volume/day for the first two hours and at .89%/day for the remainder of the accident; this bypass is caused by main steam isolation valve leakage. No credit is given for spray or suppression pool scrubbing of airborne nuclides. Also, the SER analysis only considered the offsite dose due to containment leakage; leakage of the ECCS during the recirculation phase was not included.

The results of these calculations are shown in Table 8. For the EAB, the non-instantaneous source term coupled with no bypass after two minutes resulted in a much lower dose for the first two hours. For the LPZ and control room, similar doses are seen with the revised source term, because the limited duration of bypass has a limited impact on a 30 day dose. The rebaselining calculations performed, in general, reasonably reproduced the results of our original SER analysis. One exception was the Grand Gulf LOCA control room dose in which rebaselining calculations produced higher doses. The control room doses shown in Table 8 are not representative of the design basis for Grand Gulf and are illustrative only for indicating the relative impact of the revised source term and the updated dose conversion factors.

Source Term/ Dose Conversion Factor	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
TID-14844/ ICRP 2	166	9.38 14.3(2.5h)	NA	151	10.8	NA	64.9	1.3	NA
NUREG-1465/ ICRP 2	11.3 104(7.5h)	1.13 8(3.3h)	NA	155	5.0	NA	70.8	.66	NA

NUREG-1465/ FGR 11&12	6.9 69.9(7.9h)	.98 5.9(3.1h)	8.9(3.9h)	111	4.5	11.7	50.3	.57	3.89

Table 8. Grand Gulf LOCA Doses (rem) - Phase I

Fuel Handling Accident

The SER broke down the analysis of the fuel handling accident for Grand Gulf into two cases.   In the first case, the fuel handling accident occurs in the containment. After seven seconds, the containment is isolated and the release to the environment ends. In the second case, the fuel handling accident occurs outside of the containment. In this case, the release to the environment is filtered by the SGTS. The same SER and NUREG-1465 source terms were used as in the Surry fuel handling accident analysis. Also, the same trend was seen as in the Surry fuel handling accident analysis, that is, the smaller gap release of NUREG-1465 resulted in a proportionally smaller dose.

Rod Drop Accident

When a control rod is dropped, fuel rods in the immediate vicinity of the control rod are assumed to fail. Some of the fuel rod failures are cladding failures, while some result in partial melting of the fuel. Fission products that are released from the failed fuel travel to the condenser which is assumed to leak at a rate of 1%/day for one day. The fission product release from fuel rods with cladding failures is the same as that used for the Surry fuel handling accident analysis. The fission product release from the fuel rods with melting is 100% of the noble gases and 50% of the iodine for the TID source term and 100% of the noble gases and 30% of the iodine for the NUREG-1465 source term. As shown in Table 9, the smaller gap and in-vessel release of NUREG-1465 results in a smaller dose.

Source Term/ Dose Conversion Factor	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
TID-14844/ ICRP 2	8.20	1.16	NA	5.65	.47	NA
NUREG-1465/ ICRP 2	4.19	.62	NA	2.87	.21	NA
NUREG-1465/ FGR 11& 12	2.42	.50	.58	1.78	.18	.24

Table 9. Grand Gulf Rod Drop Accident Doses (rem) - Phase I

Phase II

There were two objectives for Phase II. The first objective was to determine the effect on individual dose using FSAR assumptions of substituting the NUREG-1465 source term for the TID source term and the effect of the new dose acceptance criteria. The second objective was to determine the effect on equipment qualification dose using FSAR assumptions of substituting the NUREG-1465 source term for the TID source term. The FSAR assumptions are the assumptions used by the licensee in its analysis of offsite, control room, and equipment qualification doses. The set of accidents analyzed in Phase II was identical to the set analyzed in Phase I with the exception that the Zion LOCA analysis was added. Zion analysis was added to the matrix of calculations to provide insight into the impact of the NUREG-1465 source term on dose calculations for a large dry containment, representative of a significant class of reactors.

As a first step in Phase II, dose calculations were performed using FSAR assumptions with the TID source term. The results of these calculations were compared with results reported in the FSARs to ensure that the assumptions used in the FSAR analysis are captured in the current rebaselining study. Dose calculations were then performed using FSAR assumptions with the NUREG-1465 source term. The results are given below.

Surry DBA Dose Analysis for the LOCA

The FSAR assumed both containment leakage and ECCS leakage contributed to offsite dose. The containment was assumed to leak at its design rate (.1%/day) until pressure fell below 14.7 psia which was assumed to occur at one hour into the accident. In the FSAR, the ECCS leak rate was also included and was .003% of the sump volume/day from 0 to 29 minutes and .014%/day after 29 minutes.

The results of these calculations are shown in Table 10. The whole body dose decreased due to source term timing. The decrease is similar to that seen in the Phase I (SER) results. The EAB and LPZ thyroid doses are mainly from containment leakage. The dose reduction for NUREG-1465 due to timing is offset by a dose increase due to a very low spray rate constant assumed in the FSAR analysis for particulates. For the control room thyroid dose, half of the dose is from ECCS leakage, because bottled air is used during the first hour while the containment is leaking. Therefore, the dose reduction is partly due to the smaller iodine source term assumed for ECCS leakage (40% of core inventory vs. 50% in TID). The rebaselining calculations performed, in general, reasonably reproduced the results of the plants' FSAR analysis. An exception was the Surry LOCA control room dose in which rebaselining calculations produced higher doses. The control room doses shown in Table 10 are not representative of the design basis for Surry and are illustrative only for indicating the relative impact of the revised source term and the updated dose conversion factors.

Source Term/ Dose Conversion Factor	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
TID-14844/ ICRP 2	225	3.36	NA	12	.15	NA	45	.66	NA
TID-14844/ FGR 11&12	129	2.35	NA	7.0	.1	NA	28	.46	NA
NUREG-1465/ FGR 11&12	108	.58	4.93	5.8	.025	.25	23	.12	.935

Table 10. Surry LOCA Doses (rem) - Phase II

Grand Gulf DBA Dose Analysis for the LOCA

The FSAR assumed both containment leakage and ECCS leakage contributed to offsite dose. The source term was put into the drywell and subjected to boundary conditions of the thermal-hydraulic design basis LOCA. The wetwell was assumed to leak at its design rate (.35%/day) for the duration of the accident. The drywell was assumed to leak past the wetwell at a rate of .22% of the drywell volume/day from 20 minutes to 2.9 hours and at .89%/day after 2.9 hours. Credit is given for spray and suppression pool scrubbing of airborne fission products. Also, for the first two minutes of the accident, all containment leakage bypasses the enclosure building. After two minutes, containment leakage mixes in the enclosure building and leaves to the environment through the SGTS. The ECCS leak rate was .16% of the suppression pool volume/day.

The results of these calculations are shown in Table 11. For the EAB, most of the thyroid dose with the TID source term is from the first two minutes, because of the instantaneous source term coupled with enclosure building bypass. For the NUREG-1465 source term, no dose is from the first two minutes. However, doses are still seen because of suppression pool bypass. Flow through the suppression pool is basically over after thirty seconds, which is before the NUREG-1465 source term is released. Also, the dose from the worst two hours is equal to the TID dose because of suppression pool bypass. For the EAB, the whole body dose with the TID source term is higher for the worst two hours than for the first two hours because of the dual containment. For the LPZ and control room thyroid doses, for the TID source term half of the dose is from containment leakage and half is from ECCS leakage. For NUREG-1465, the containment leakage contribution is smaller, because the organic iodine fraction is reduced from 4% to .15%. Also, the ECCS leakage contribution is smaller, because the iodine source term to the suppression pool is reduced from 50% to 30% of core inventory. Finally, because of the spray system scrubbing of iodine, the TEDE dose is mostly a noble gas dose.

Source Term/ Dose Conversion Factor	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TED E
TID-14844/ FGR 11&12	23.1	9.5 11.5(1.0h)	NA	40.1	7.57	NA	8.87	.726	NA
NUREG-1465/ FGR 11&12	10.6 22.6(1.5h)	1.5 5.89(2.3h)	1.95 6.78(2.2h)	19.5	4.05	4.73	3.94	.392	.532

Table 11. Grand Gulf LOCA Doses (rem) - Phase II

Zion DBA Dose Analysis for the LOCA

The FSAR assumed both containment leakage and ECCS leakage contributed to offsite dose. The containment was assumed to leak at its design rate (.1%/day) for the first day and then about half of its design rate for the remainder of the accident. ECCS leakage was .003%/day for the first hour and 0 thereafter.

The results of these calculations are shown in Table 12. Smaller thyroid doses are calculated with the NUREG-1465 source term for two reasons. First, NUREG-1465 has almost no organic iodine compared with what was assumed in the FSAR (.15% vs. 10%). Second, the particulate releases to the containment were assumed to be washed out by sprays at a high removal rate. Because of the high spray removal rate used by Zion, TEDE is mostly a noble gas dose.

Source Term/ Dose Conversion Factors	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
TID-14844/ ICRP 2	181	6.23	NA	161	3.31	NA
TID-14844/ FGR 11 & 12	105	4.30	NA	101	2.24	NA
NUREG-1465/ FGR 11& 12	14.3	1.01 1.71(1.3h)	1.62 2.07(.9h)	5.83	.93	1.15

Table 12. Zion LOCA EAB and LPZ Doses - Phase II

Surry and Grand Gulf DBA Dose Analysis for the Fuel Handling, Main Steam Line Break, and Rod Drop Accidents

In Phase II, offsite doses for fuel handling, main steam line break, and rod drop accidents were calculated using FSAR assumptions and both the TID and the NUREG-1465 source terms. The insights yielded by the calculations were the same as from the Phase I calculations for these accidents. For fuel handling and rod drop accidents the dose is again reduced because the magnitude of the NUREG-1465 source term is lower for these accidents.

Equipment Qualification Dose Analysis

To gain insight into the impact of the NUREG-1465 source term on equipment qualification doses, LOCA doses were calculated for equipment exposed to the containment atmosphere and for equipment exposed to sump water. Dose calculations were first performed using FSAR assumptions with the TID source term. The results of these calculations were compared with results reported in the FSARs to ensure that the assumptions used in the FSAR analysis are captured in the current rebaselining analysis. In general, calculated results with the TID source term compared well with the doses reported in the FSARs. Dose calculations were then performed using FSAR assumptions with the NUREG-1465 source term. The results of these calculations were compared with the results of the dose calculations with the TID source term.

Surry Assumptions and Results

For the airborne gamma dose, the FSAR assumed a single volume containment with no removal of airborne activity. For the airborne beta dose, the FSAR modeled the containment atmosphere as two volumes, one with removal of iodine by sprays and one without. The FSAR source term was 100% of the core inventory of noble gases and 50% of the iodine. For the NUREG-1465 source term, the gap and in-vessel releases of noble gases and iodine were used.

For the sump dose, the FSAR assumed an instantaneous source term of 50% of the iodine and 1% of the solids. For the NUREG-1465 source term, the gap and in-vessel releases of all nuclide groups, except noble gases which were assumed not to dissolve in the water, were used. The results of the rebaselining calculations with both the TID and NUREG-1465 source terms are shown in Figures 2 and 3.

Grand Gulf Assumptions and Results

For the airborne gamma and beta doses, the FSAR used a single volume containment (i.e., wetwell) with removal of iodine by settling. The source term used was 100% of the core inventory of noble gases and 50% of the iodine. For the NUREG-1465 source term, the gap and in-vessel releases of noble gases and iodine were used.

For the sump dose, the FSAR assumed an instantaneous source term of 50% of the iodine and 1% of the solids. For the NUREG-1465 source term, the gap and in-vessel releases of all nuclide groups, except noble gases which were assumed not to dissolve in the water, were used. The results of the rebaselining calculations with both the TID and NUREG-1465 source terms are shown in Figures 4 and 5.

Equipment Qualification Dose Conclusions

For airborne gamma and beta doses, the doses with the TID source term were about the same as those with the NUREG-1465 source term, because only noble gases and iodine were assumed to be airborne and the magnitudes of the noble gas and iodine releases of the source terms are similar. For the sump, the gamma doses are higher at later times for the NUREG-1465 source term, because of the large amount of cesium in the NUREG-1465 source term. The TID source term included 1% of the core inventory of cesium, while the NUREG-1465 source term includes 30% of the cesium.

Phase III

There were three objectives for Phase III. The first objective was to determine the effect on individual dose using updated removal mechanism models with the NUREG-1465 source term and the effect of the new dose acceptance criteria. The second objective was to determine the effect on equipment qualification dose using updated removal mechanism models with the NUREG-1465 source term. The updated models are those that are being considered for the update of Regulatory Guides 1.3 and 1.4 on evaluation of radiological consequences for the design basis LOCA. The third objective was to assess the margin in the licensing dose methodology for assessment of offsite releases. In Phase III, the only accident evaluated was the LOCA.

Surry DBA Dose Analysis for the LOCA - Updated Models

The FSAR breaks the containment down into two regions, a sprayed region and an unsprayed region. The FSAR assumes a small spray removal rate for particulates. In contrast, a greater removal rate is given in the Standard Review Plan. In Phase III, removal rates from the Standard Review Plan and from the updated models in NUREG/CR-5966 were used. The spray removal rates are designated in the following tables as "λ_p" for removal rate of particulate fission products. The updated models in NUREG/CR-5966 are a 50th percentile model and 10th and 90th percentile models giving a measure of uncertainty in the 50th percentile model.

The results for Surry for Phase III are given in Table 13. There are several insights that were obtained from this analysis. For the EAB, the first two hours are the worst two hours, because the subatmospheric containment does not leak after one hour. Also, the TEDE dose is mainly from iodine. With the exception of the EAB thyroid dose for the FSAR λ_p case, EAB and LPZ doses with the NUREG-1465 source term are a small fraction of the 300 rem thyroid and 25 rem whole body and TEDE limits. Also, control room thyroid doses are about one half of the 30 rem thyroid limit. Finally, ECCS leakage is important for LPZ and control room thyroid doses, but it is unimportant for other doses.

Spray Model	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
FSAR λ_p	108	.58	4.93	5.78	.025	.25	22.8	.12	.94
SRP λ_p	62	.40	2.95	3.81	.018	.16	17.5	.082	.69
10% λ_p	76	.46	3.55	4.41	.021	.19	19.2	.092	.76
50% λ_p	48	.35	2.33	3.21	.016	.14	16.0	.071	.62
90% λ_p	29	.27	1.47	2.37	.013	.10	13.8	.057	.51

Table 13. Surry LOCA Doses (rem) - Phase III

Grand Gulf DBA Dose Analysis for the LOCA - Updated Models

The FSAR models the containment as four volumes, a drywell and three wetwell volumes. One of the wetwell volumes is sprayed. The FSAR assumes a spray removal rate comparable to that in the Standard Review Plan. The FSAR also assumes the suppression pool decontamination factor given in the Standard Review Plan. In Phase III, removal rates and decontamination factors from the Standard Review Plan and from the updated models in NUREG/CR-5966 and NUREG/CR-6153 were used.

The results for the Grand Gulf for Phase III are given in Table 14. All rows in the table use the Standard Review Plan suppression pool decontamination factor (DF), except for the final row which uses the 50th percentile DF from NUREG/CR-6153. For the NUREG-1465 source term, the first two minutes are unimportant for overall dose, because the source term has only begun to appear in containment. Most of the TEDE dose is from noble gas, because of the containment sprays and the SGTS that filters releases from the secondary containment. However, sensitivity studies revealed that the most of the dose reduction is due to the SGTS. Suppression pool scrubbing is less important with the NUREG-1465 source term, because most of the flow from the drywell to the containment bypasses the suppression pool. Finally, ECCS leakage is important for LPZ and control room thyroid doses and unimportant for other doses.

Spray or Suppression Pool Model	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
FSAR λ_p	22.6(1.5h)	5.89(2.3h)	6.78(2.2h)	19.5	4.05	4.73	3.94	.39	.53
SRP λ_p	21.2(1.6)	5.89(2.3h)	6.72(2.2h)	19.1	4.05	4.72	3.85	.39	.528
10% λ_p	24.4(1.6h)	5.90(2.3h)	6.89(2.2h)	19.9	4.06	4.76	4.06	.39	.538
50% λ_p	20.0(1.6h)	5.86(2.3h)	6.65(2.2h)	18.7	4.05	4.69	3.77	.39	.524
90% λ_p	19.8(1.6h)	5.86(2.3h)	6.63(2.2h)	18.7	4.05	4.69	3.75	.39	.523
50% DF	22.0(1.6h)	5.89(2.3h)	6.78(2.2h)	19.2	4.05	4.72	3.91	.39	.531

Table 14. Grand Gulf LOCA Doses (rem) - Phase III

Zion DBA Dose Analysis for the LOCA - Updated Models

The FSAR models the containment as a single sprayed volume. The FSAR assumes a spray removal rate significantly larger than the Standard Review Plan. In Phase III, removal rates from the Standard Review Plan and from the updated models in NUREG/CR-5966 were used. The results for Zion for Phase III are given in Table 15. Because of the large FSAR λ_p, doses with the NUREG-1465 source term are very small and the TEDE doses are mainly from noble gases. Also, doses with the NUREG-1465 source term are a small fraction of the 300 rem thyroid and 25 rem whole body and TEDE limits.

Spray Model	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
FSAR λ_p	14.5(0h)	1.65(1.2h)	2.03(.8h)	6.02	.91	1.14
SRP λ_p	72.8(0h)	1.75(1.1h)	4.45(.5h)	23.2	.97	1.93
10% λ_p	145(.3h)	1.94(.9h)	8.02(.5h)	47.6	1.04	3.06
50% λ_p	48.1(0h)	1.69(1.1h)	3.40(.5h)	15.9	.943	1.60
90% λ_p	24.9(0h)	1.66(1.2h)	2.41(.6h)	9.08	.919	1.28

Table 15. Zion LOCA Doses (rem) - Phase III

Comparison of DBA Dose Analyses for Phases I, II, and III

To gain insight into the overall impact on dose calculations of changing from the TID source term to the NUREG-1465 source term, the difference in the Phases I, II, and III results for the DBA LOCA were analyzed. A comparison of the Surry results is shown in Table 16. For Surry, NUREG-1465 produced reduced doses, because containment leakage stopped at one hour while the source term continued to be released into containment until 1.8 hours.

Assumptions	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
SER (TID-14844)	232	3.09	NA	20.3	.270	NA	30.6	.59	NA
FSAR (TID-14844)	225	3.36	NA	12	.15	NA	45	.66	NA
FSAR (NUREG-1465)	76	.46	3.55	4.41	.021	.19	19.2	.092	.76

Table 16. Comparison of Phases I, II, and III Surry LOCA Doses (rem)

A comparison of the Grand Gulf results is shown in Table 17. For the TID source term, the thyroid doses are smaller for the FSAR, because credit is given for spray and suppression pool removal. Also for the TID source term, the whole body doses are different, because of different assumptions of flow between drywell and containment. The thyroid doses with NUREG-1465 are all lower except for the EAB thyroid dose. The EAB thyroid doses for FSAR-TID and FSAR-NUREG-1465 are comparable, because the NUREG-1465 source term bypasses the suppression pool and is instead washed out by the sprays while the TID release is instantaneous and is scrubbed by the suppression pool. The whole body doses are lower with NUREG-1465, because of the source term timing coupled with the short half-life of Kr-88 which is the dominant dose contributor.

Assumptions	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TED E
SER (TID-14844)	166.2	9.38 14.3(2.5h)	NA	151	10.8	NA	64.9	1.30	NA
FSAR (TID-14844)	23.1	9.52 11.5(1.0h)	NA	40.1	7.57	NA	8.87	.726	NA
FSAR (NUREG-1465)	24.4(1.6h)	5.90(2.3h)	6.89(2.2h)	19.9	4.06	4.76	4.06	.39	.538

Table 17. Comparison of Phases I, II, and III Grand Gulf LOCA Doses (rem)

A comparison of the Zion results is shown in Table 18. The FSAR used a larger atmospheric dispersion factor and a larger λ_p than the SER. The larger atmospheric dispersion factor counters the effect of the larger λ_p. For the EAB, it more than offsets the larger λ_p causing the thyroid dose to be increased. The LPZ thyroid dose in the NUREG-1465 case is much smaller than the other cases due to the reduced amount of organic iodine in the NUREG-1465 source term (.15% vs 10%).

Assumptions	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
SER (TID-14844)	146	3.56	NA	175	2.08	NA
FSAR (TID-14844)	181	6.23	NA	161	3.31	NA
FSAR (NUREG-1465)	145(.3h)	1.94(.9h)	8.02(.5h)	47.6	1.04	3.06

Table 18. Comparison of Phases I, II, and III Zion LOCA Doses (rem)

MELCOR Severe Accident Analysis

As part of Phase III, calculations were performed with the MELCOR code, the NRC's integral severe accident code, to evaluate the margin in the licensing dose methodology's estimation of releases to the environment for the Surry, Grand Gulf, and Zion LOCAs. This was done by calculating EAB and LPZ doses from the MELCOR release to the environment for a recovered core-melt accident which involved a fission product release from the fuel similar to the NUREG-1465 release. Because use of the traditional design basis accident with ECCS operational would not result in core damage with any significant release of fission products, a core damage accident with delayed recovery of ECCS was used to approximate the relative magnitude of the NUREG-1465 source term. In one set of calculations, the NUREG -1465 release was coupled to MELCOR thermal-hydraulics and fission product removal. The main results of the MELCOR calculations are described below along with their significance with respect to assessment of margin in the licensing dose methodology.

The MELCOR analysis consisted of integral calculations of plant thermal-hydraulics and fission product release and deposition. This analysis was also used to assess consistency of the thermal-hydraulic treatment in the licensing dose calculations. This analysis was also intended as an overall examination of a best-estimate dose calculation. MELCOR calculations were performed using both the MELCOR-generated source term and the NUREG-1465 source term. In the case of the MELCOR-generated source term, water was injected into the vessel at the time that 40% (30% for Grand Gulf) of the iodine was released into the containment. For the NUREG-1465 source term case, water was injected at the same time as for the MELCOR-generated source term case. MELCOR containment releases were coupled with the licensing dose code to produce offsite doses for comparison with results from the Phase III licensing dose calculations.

Surry

The MELCOR analysis indicates that the licensing dose assumption that it takes one hour to become subatmospheric is conservative. MELCOR predicts that subatmospheric conditions are achieved in about 20 minutes as shown in Figure 6. Another insight gained from the MELCOR analysis was that the offsite dose can be sensitive to assumptions regarding location of containment leakage, i.e., whether the leak is from a sprayed or an unsprayed region.

As mentioned above, the MELCOR-calculated fission product releases to the environment were coupled with the licensing dose code to produce offsite doses for comparison with results from the Phase III licensing-type dose calculations. The comparison of these MELCOR doses with the Phase III licensing-type dose calculations is given in Table 19.

Assumptions	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Containment and ECCS leakage						
10% λ_p	76	.46	3.55	4.41	.021	.19
MELCOR (NUREG-1465)	1.55	.006	.055	1.16	.001	.037
MELCOR (MELCOR)	1.60	.01	.061	1.16	.001	.037
Containment leakage only						
MELCOR (NUREG-1465)	.024	.001	.002	.003	.00008	.0002
MELCOR (MELCOR)	.07	.005	.008	.003	.0002	.0003

Table 19. Comparison of Surry Licensing Dose with MELCOR-based LOCA Doses (rem)

<u>Grand Gulf</u>

As shown in Figure 7, the FSAR treatment of thermal-hydraulics results in virtually all of the NUREG-1465 source term bypassing the suppression pool. The blowdown is over before the fission products are released into the drywell. The bypass flow dominates pool flow (30:1) during the period of fission product release into the drywell. Also as shown in Figure 7, the MELCOR treatment of thermal-hydraulics results in a larger flow of fission products through the suppression pool due the reflood assumption. Therefore, continued use of the FSAR DBA thermal-hydraulic profile will conservatively treat suppression pool scrubbing with the NUREG-1465 source term. Alternatively, the NUREG-1465 source term could be coupled with a simplified thermal-hydraulic profile simulating reflood at the end of the in-vessel release allowing suppression pool scrubbing for a short interval.

The MELCOR-calculated fission product releases to the environment were coupled with the licensing dose code to produce offsite doses for comparison with results from the Phase III licensing dose calculations. The comparison of these MELCOR doses with the Phase III licensing dose calculations is given in Table 20.

Assumptions	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Containment and ECCS leakage						
10% λ_p	24.4(1.6h)	5.90(2.3h)	6.89(2.2h)	19.9	4.06	4.76
MELCOR (NUREG-1465)	11.7(3.8h)	4.76(2.4h)	5.05(2.5h)	16.1	3.27	3.77
MELCOR (MELCOR)	8.21(.7h)	2.01(1.4h)	2.27(1.2h)	15.7	1.33	1.82
Containment leakage only						
MELCOR (NUREG-1465)	5.95(3.8h)	4.74 (2.4)	4.87 (2.5)	1.95	3.24	3.31
MELCOR (MELCOR)	5.98(.6h)	2.00(1.4h)	2.15(1.1h)	1.54	1.3	1.36

Table 20. Comparison of Grand Gulf Licensing Dose with MELCOR-based LOCA Doses (rem)

Zion

MELCOR results indicate that modeling of the thermal-hydraulics mechanistically produces a more rapid depressurization than the licensing dose calculation assumptions (i.e., containment pressure must be reduced to below half of design pressure within 24 hours). Another insight of the MELCOR analysis is that the containment volumetric leak rate will vary as a function of the pressure depending on the nature of the flow. The containment leak rate is largely invariant if choked flow persists, while the containment leak rate is a direct function of the pressure under subsonic flow conditions. The net effect of modeling the Zion leak path as an orifice (with initial choked-flow conditions) is to produce lower leakage than assumed in the licensing dose calculations. Modeling the Zion leak path as having a large pressure drop along its length so that no choked flow conditions exist would produce an even lower leakage.

The MELCOR-calculated fission product releases to the environment were coupled with the licensing dose code to produce offsite doses for comparison with results from the Phase III licensing dose calculations. The comparison of these MELCOR doses with the Phase III licensing dose calculations is given in Table 21.

Assumptions	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Containment and ECCS leakage						
10% λ_p	145(.3h)	1.94(.9h)	8.02(.5h)	47.6	1.04	3.06
MELCOR (NUREG-1465)	.79(.1h)	1.03(1.2h)	1.03(1.1h)	.23	.51	.521
MELCOR (MELCOR)	.61(0h)	1.42(.4h)	1.4(.4h)	.18	.67	.68
Containment leakage only						
MELCOR (NUREG-1465)	.48(.2h)	1.03(1.2h)	1.03(1.1h)	.14	.51	.52
MELCOR (MELCOR)	.29(.2h)	1.41(.4h)	1.43(.4h)	.09	.67	.67

Table 21. Comparison of Zion Licensing Dose with MELCOR-based LOCA Doses (rem)

Equipment Qualification Dose Analysis

In Phase III, equipment qualification doses using FSAR assumptions with the NUREG-1465 source term were again calculated. However, this time the FSAR removal rates were replaced with those of the updated models. In Phase III, our equipment qualification dose analysis was limited to Surry, because Phase II showed no plant-specific differences between Surry and Grand Gulf with respect to equipment qualification dose assessment.

For containment atmosphere and sump doses, a two-volume containment (a sprayed volume and an unsprayed volume) with a sump was used. Spray removal rates from Regulatory Guide 1.89 on equipment qualification and from the updated models in NUREG/CR-5966 were also used . For the source term, the NUREG-1465 gap and in-vessel releases of all nuclide groups was put into the containment atmosphere. Other than noble gases, the nuclides were quickly washed to the sump (90% of the airborne nuclides were washed to the sump in .07 hours). The results of the rebaselining calculations with the NUREG-1465 source term and updated assumptions are shown in Figures 8 and 9 together with the results from Phase II with the NUREG-1465 source term. The Phase III containment center gamma dose is lower, because washout of nuclides was modeled in Phase III but not in Phase II. The containment center beta dose is the same in Phase III, as is expected (washout of nuclides was modeled in Phase II). Finally, the sump dose is also about the same, because the spray washes the source term to the sump very quickly.

pH and Iodine Revaporization Analysis

The chemical form fractions of iodine in the NUREG-1465 source term are based on an assumption that the pH of the sump water is controlled to be above 7. To ensure spray effectiveness for removing iodine vapor, PWRs control pH, typically with either sodium hydroxide or sodium phosphate. However, BWRs do not control pH. It is currently assumed that plants proposing changes using the NUREG-1465 source term will control pH to above 7. However, as a part of the rebaselining study, an investigation was performed into the potential for increased offsite releases if pH is not controlled.

The TRENDS containment chemistry code was used to evaluate the magnitude of iodine revaporization for a recovered core-melt accident in which pH was not controlled. The scenario analyzed was the MELCOR Grand Gulf large-break LOCA with core injection at 3000 seconds to arrest core melting prior to lower head failure. The MELCOR thermal-hydraulics and fission product deposition were used as boundary conditions for the TRENDS calculation. The results of the TRENDS calculation performed for Grand Gulf, without pH control, indicated that revolatilization occurred, producing iodine vapor in five hours in an amount equal to that prescribed for organic iodine in NUREG-1465, which assumes sump pH is controlled.

Phase IV

Phase IV involved calculating offsite and control room DBA doses with updated models and potential plant changes using the NUREG-1465 source term. Phase IV also involved assessing the risk changes from these plant changes. To date, four plants (pilot plants) have submitted proposed changes and associated safety analyses to the NRC using the NUREG-1465 source term. The pilot plants and proposed changes are listed in Table 22. The pilot plant submittals include dose calculations for the listed plant changes with the NUREG-1465 source term. However, the pilot plant submittals do not include dose calculations for the worst two hours and TEDE as proposed by the staff for dose calculations with the NUREG-1465 source term. Therefore, in Phase IV, calculations were performed with the changes proposed by the pilot plants using the rebaselining plants as surrogates to evaluate the combined impact of the proposed plant changes, the worst two hours, and TEDE. The rebaselining plants used for each of the changes proposed by the pilot plants are also shown in Table 22.

In draft NUREG/CR-6418, Risk Importance of Containment and Related ESF System Performance Requirements, the NRC investigated the risk importance of several potential plant changes. In Phase IV, offsite and control room dose calculations were also performed for any potential plant changes listed in draft NUREG/CR-6418 that were not proposed by the pilot plants. The assessments described below were performed to investigate the dose changes as a result of potential plant changes with the NUREG-1465 source term.

Plant	Reactor	Containment	Proposed Change	Rebaselining Plant
Perry	GE 6	Mark III	eliminate MSIV leakage control system, increase allowable MSIV leak rate	Grand Gulf
Browns Ferry	GE 4	Mark I	same as Perry plus eliminate charcoal filters in SGTS and control room	Grand Gulf and Surry (charcoal filters only)
Oyster Creek	GE 2	Mark I	no change; additional control room safety assessment using NUREG-1465	Analysis performed in Phases I, II, III
Indian Point 2	West. Four-Loop	Large, Dry	eliminate charcoal and HEPA containment recirculation filters	Zion

Table 22. Pilot Plants for the NUREG-1465 Source Term

DBA Dose Analysis - Potential Plant Changes

Elimination of MSIV Leakage Control System and Increase in Allowable MSIV Leakage

For this potential plant change, rebaselining calculations were done for Grand Gulf. The starting point was the Phase III Grand Gulf calculation with the updated spray removal model. In the Phase III Grand Gulf calculation, the MSIV leaked at a rate of .22% of the drywell volume/day from 20 minutes to 2.9 hours and at .89%/day after 2.9 hours. Perry proposed eliminating the MSIV leakage control system and increasing the MSIV leak rate to 250 scfh (corresponding roughly to 2.2%/day for Grand Gulf). Therefore, two calculations were run for Grand Gulf. In one calculation, the MSIV leakage control system was eliminated. In another calculation, the MSIV leakage control system was eliminated and the MSIV leak rate was increased to 2.2%/day. In addition, the impact of modeling deposition in the main steam line, which was credited by Perry in their pilot plant submittal, was assessed. The results of these calculations are given in Table 23 together with the Phase III results. Removal of the MSIV leakage control system without credit for deposition in the main steam line resulted in doses less than the dose limits. In addition, some increase in the allowable MSIV leak rate was possible without exceeding the dose limits if credit for deposition in the main steam line is permitted.

Case	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Phase III	24.4(1.6h)	5.90(2.3h)	6.98(2.2h)	19.9	4.06	4.76	4.06	.39	.54
No LCS	226(.3h)	5.72(2.5h)	14.5(.5h)	54.4	4.22	6.82	9.34	.40	.847
No LCS, 2.2%/day	2169(.3h)	14.2(.5h)	119(.4h)	492	6.72	30.4	81.2	.57	4.49
No LCS, steam line deposition	17.1(1.5h)	5.72(2.5h)	6.43(2.0h)	5.22	4.06	4.30	1.22	.39	.446
No LCS, 2.2%/day, steam line deposition	852(.3h)	10(.5h)	50(.4h)	196	5.73	15.0	32.6	.51	2.05

Table 23. Effect of Changing MSIV Leakage

Elimination of Charcoal and HEPA Containment Recirculation Filters

To examine the impact of removing charcoal and HEPA recirculation filters from Indian Point 2 we performed analyses with the Zion plant model. Since Zion does not have those filter systems, the difference in performance was assessed by performing calculations with those features added to the Zion model (referred to as the base case in Table 24). These results were then compared to the Phase III analyses for Zion, which did not have the filter systems. Because the whole body doses are mainly from noble gases, removing recirculation filters did not significantly change the whole body doses. However, the thyroid dose is mostly from iodine. Removing recirculation filters increased the thyroid dose. In addition, most of the thyroid dose increase from removing filters is due to removal of the HEPA filter, because the NUREG-1465 source term is mostly particulate.

Case	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Base Case	77.0(0h)	1.42(.5h)	4.51(.0h)	23.6	.975	1.97
No Filters (Phase III)	145(3h)	1.94(.9h)	8.02(.5h)	47.6	1.04	3.06

Table 24. Effect of Containment Recirculation Filters

Elimination of Charcoal Filters in the SGTS and Control Room

For this potential plant change, rebaselining calculations were done for Grand Gulf and Surry. For Grand Gulf, charcoal filters were removed from the SGTS and control room. For Surry, charcoal filters were removed from the Safeguards Building (filters ECCS leakage) and the control room. The results of these calculations for Grand Gulf assuming pH control are given in Table 25 together with the Phase III results. As shown in Table 25, elimination of charcoal filters at Grand Gulf results in thyroid and TEDE dose increases. The dose increases for the EAB and LPZ are a result of elimination of charcoal filters from the SGTS (factor of three increase in thyroid dose). The dose increases for the control room are a result of the elimination of charcoal filters from the SGTS and the control room. For Grand Gulf, the control room thyroid dose limit was exceeded when the charcoal filters were removed. Similar dose increases were seen for the Surry control room. However, Surry EAB and LPZ doses did not significantly increase, because they are dominated by containment leakage which is unfiltered.

Case	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Phase III	24.4(1.6h)	5.90(2.3h)	6.98(2.2h)	19.9	4.06	4.76	4.06	.39	.54
No Charcoal Filters	77.9(1.6h)	5.94(2.3h)	8.53(2.0h)	68.8	4.09	6.29	95	.39	3.44

Table 25. Elimination of Charcoal Filters for Grand Gulf

Increased Containment Leak Rate

For this potential plant change, rebaselining calculations were done for all three rebaselining plants. For each plant the containment leak rate was arbitrarily increased by a factor of 10. The results of these calculations for Zion are given in Table 26 along with the Phase III results. For Zion, the containment leak rate was increased from .1%/day to 1%/day, and the doses increased by a factor of 10. If the EAB TEDE dose is limiting for Zion, a factor of three increase in the containment leak rate would be the highest allowable to meet the proposed TEDE dose limit. Doses for Surry and Grand Gulf increased by a factor of 10, except for Grand Gulf thyroid doses which only increase by a factor of eight due to MSIV leakage.

Case	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Phase III	145(.3h)	1.94(.9h)	8.02(.5h)	47.6	1.04	3.06
Increased Leak Rate	1398(0h)	16.2(.5h)	74.2(0h)	448	10.4	29.4

Table 26. Increased Containment Leak Rate for Zion

Delayed Spray Startup Time

For this potential plant change, rebaselining calculations were done for all three rebaselining plants. For each plant the spray startup time was delayed by an additional 30 minutes beyond the startup time used in Phase III. The results of the calculations for Zion are given in Table 27, together with the Phase III results. Surry results show similar dose increases to Zion. Grand Gulf results show almost no dose increases.

Case	EAB			LPZ		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Phase III	145(.3h)	1.94(.9h)	8.02(.5h)	47.6	1.04	3.06
Delayed Spray	162(0h)	1.66(.5h)	8.32(0h)	51.2	1.07	3.21

Table 27. Delayed Spray Startup for Zion

Increased Enclosure Building Drawdown Time

For this potential plant change, rebaselining calculations were done for Grand Gulf. The enclosure building drawdown time was increased from 2 minutes to 30 minutes. The results of these calculations are given in Table 28 together with the Phase III results. These results show the EAB thyroid dose increased by a factor of three due to bypass of the SGTS filters for 30 minutes. Also, the worst two hours thyroid dose now begins at 0 hours.

Case	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Phase III	24.4(1.6h)	5.90(2.3h)	6.98 (2.2h)	19.9	4.06	4.76	4.06	.39	.54
Increased Drawdown Time	69.8(0h)	5.83(2.4)	6.63(2.1h)	19.9	4.08	4.89	3.60	.39	.54

Table 28. Increased Enclosure Building Drawdown Time for Grand Gulf

Changing Containment from Subatmospheric to Atmospheric

For this potential plant change, rebaselining calculations were done for Surry. Instead of ending containment leakage at 1 hour, the leakage was continued at the same rate until 24 hours and then reduced to half of that value after 24 hours. The results of this calculation are given in Table 29 together with the Phase III results.

Case	EAB			LPZ			Control Room		
	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE	Thyroid	Whole Body	TEDE
Phase III	76	.46	3.55	4.41	.021	.19	19.2	.092	.76
Atmospheric Containment	444(.8h)	3.50(.8h)	22.6(.8h)	71.3	.589	3.67	678	2.62	32.0

Table 29. Changing Containment from Subatmospheric to Atmospheric for Surry

Risk Impacts of Potential Plant Changes Using the Revised Source Term

The staff has assessed the risk impacts of potential plant changes that may be proposed in conjunction with implementation of the revised source term. This assessment was conducted for the NUREG-1150 plants along with the LaSalle plant and considered the potential risk impacts from relaxation of engineered safety (ESF) systems' performance requirements. The areas evaluated included 1) containment leak rate, 2) containment spray operation, 3) filtration systems, 4) secondary containment drawdown time, and 5) subatmospheric containment operation. (The details of this study will be published in NUREG/CR-6418.) Each of these potential plant modifications is discussed below.

The approach was to determine the risk contributors from existing PRAs (NUREG-1150 and the LaSalle Integrated Risk Assessment Program) and determine the risk importance of each of the potential plant modifications. As part of this effort, the accident progression event trees (APETs) for Zion and Peach Bottom that were used for NUREG-1150 were modified to reflect the current knowledge of the early containment failure phenomena (direct containment heating, in-vessel steam explosions and liner melt-through). Once the relative risk importance was determined, the impact of relaxing the performance requirements was assessed. In some cases, conclusions could be drawn based on qualitative arguments. (This assessment evaluated the risk impact of potential changes and did not evaluate the appropriate extent of defense-in-depth, e.g., balance between prevention and mitigation, which will be determined based on the actual plant changes.) Although, the risk impacts discussed below have not been assessed with respect to the large early release frequency (LERF) acceptance guidelines established in Regulatory Guide 1.174, none are expected to be of such significance that the LERF guidelines would be violated. Rather than use the LERF measure, the discussion below uses the latent cancer fatality risk measure, which shows the impact of very small risk changes.

Containment Leak Rate

With regard to the impact of varying the containment leak rate, previous assessments in NUREG/CR-4330 and NUREG-1493 indicated that offsite consequences are not very sensitive to the containment leak rate because risk tends to be dominated by accident sequences that result in early containment failure or bypass of containment. Figure 10, taken from NUREG/CR-4330, shows an example of a modest change in the population exposure risk as a function of containment leak rate.

In draft NUREG/CR-6418, the NUREG-1150 assessments for Peach Bottom and Zion were updated to reflect the current knowledge of the early containment failure phenomena (direct containment heating and liner melt through). The resulting revised conditional probability of accident progression bins, together with the representative source terms, were used. These source terms were used with the MELCOR Accident Consequences Code System (MACCS) to estimate the offsite consequences and risk. This assessment corroborated the previous assessment findings, namely, that offsite consequence risk is not very sensitive to the containment leak rate.

Containment Spray Operation

A review of NUREG-1150 and the LaSalle risk studies indicates that the dominant contributors to risk are from sequences which do not have sprays available (e.g., station blackout). Although sprays are not important for those risk dominant sequences, they are effective at lowering the risk for a wide range of other accidents.

However, many current plant designs involve spray operation very shortly after accident initiation for the DBA LOCA. Delaying spray operation could have some benefits in sequencing diesel loads or conserving water storage inventory. A study was performed to assess the risk impacts of delaying spray operation. This study concluded that delaying spray operation would not increase plant risk and could potentially decrease risk for some sequences by preserving water supplies. Delayed spray operation would still need to be evaluated for impacts on the DBA containment pressure calculation.

Filtration Systems

The effect of the modification of filtration systems on overall plant risk is judged to be small for two principal reasons. First, systems requiring electrical power would not be available for certain significant accident contributors to risk, e.g., station blackout. Secondly, in-containment recirculation filtration systems are subject to plugging from high aerosol loadings.

Secondary Containment Drawdown Time

A review of the accident sequences in NUREG-1150 indicates that BWR Mark I containment (drywell) failure at the time of reactor vessel failure represents approximately 90% of the mean population dose. Since the ventilation system which reduces the secondary containment pressure requires electrical power and the dominant accident sequences are station blackout sequences, only some credit was given in NUREG-1150 for retention of fission products in the secondary containment and no credit for negative pressure or filtration. Secondary containment drawdown time could impact the remaining sequences representing approximately 10% of the accident sequences contributing to the mean population dose, but it is unlikely that this would significantly change the overall plant risk.

Subatmospheric Containment

The NUREG-1150 APETs were modified for Surry to assume that the plant was operating with containment at atmospheric pressure. The APETs were reevaluated and the conditional accident progression bin probabilities revised. Revised source terms were calculated and the offsite consequences were evaluated with MACCS. The revised overall plant risk was nearly identical to that in NUREG-1150. The results indicate that operating subatmospheric containments at atmospheric pressure has a very small impact on the overall plant risk. These results are due, in part, to the fact that the major contributors to risk at Surry are containment bypass sequences.