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ARTHUR E. LUNDVALL, JR.
VICE PRESIDENT
SUPPLY

November 18, 1985

Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Attn: Mr. E. J. Butcher, Jr., Chief
Operating Reactors Branch #3
Division of Licensing

Subject: Calvert Cliffs Nuclear Power Plant
Units 1 and 2; Dockets 50-317 and 50-318
Request for Exemption

Gentlemen:

Pursuant to 10 CFR 50.12(a), Baltimore Gas and Electric Company hereby requests an exemption for Calvert Cliffs Nuclear Power Plant (CCNPP) Units Nos. 1 and 2, Facility Operating Licenses DPR-53 and -69, respectively, from a requirement of 10 CFR 50.47(c)(2) and 10 CFR 50 Appendix E, as described below.

Exemption Request

In accordance with the provisions of 10 CFR 50.12(a), we request an exemption from the general requirement of 10 CFR 50.47(c)(2) and Section I of 10 CFR 50 Appendix E, which states that the plume exposure pathway emergency planning zone (EPZ) should consist of an area of about 10 miles in radius. Specifically, we request that the plume exposure pathway EPZ surrounding CCNPP be redefined as the area within a radius of 2 miles from the center of the containment/auxiliary building complex, based on our determination that a plume exposure pathway EPZ beyond 2 miles is not necessary to achieve the underlying purpose of the rule, as discussed in Attachment 1, and that there now exist significant material circumstances which were not available for consideration when the regulation in question (i.e., the 10 mile radius) was adopted, as discussed in Attachment 2.

Our determination relative to the adequacy of the proposed 2-mile EPZ was achieved using the same regulatory philosophy and basic approach as that presented in NUREG-0396, but utilizing current source term information.

We have concluded that the requested exemption is authorized by law, will not endanger life or property or the common defense and security and is otherwise in the public interest.

The exemption is in the public interest for a number of reasons. The size proposed for our plume exposure pathway EPZ is based upon a more recent, more rigorous and more

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accurate body of knowledge than that which formed the basis for the existing EPZ size, as discussed in Attachment 2. Consequently, the more realistic EPZ size would allow BG&E and the applicable government agencies to focus and apply their emergency planning resources more efficiently within the area that is most likely to be subject to actual emergency conditions. The exemption would also eliminate the unnecessary regulatory burden of emergency planning from those local jurisdictions which would no longer be included in the plume exposure pathway EPZ. As currently configured, the EPZ encompasses portions of three counties: Calvert, Dorchester and St. Mary's. The proposed EPZ would lie completely within Calvert County. The rationale for the conclusion that life and property will not be endangered is inherent in the basis for the proposed reduced EPZ size, as discussed in Attachments 1 and 2.

Safety Committee Review

This request has been reviewed by our Plant Operations and Safety Review Committee and Off-Site Safety Review Committee, and they have concluded that, if granted, it will not result in an undue risk to the public health and safety.

Fee

In accordance with 10 CFR 170.12 and 170.21, a check is enclosed in the amount of $150.00 as payment for the fee for this application.

Very truly yours,

JA Tiernan

for A. E. Lundvall, Jr.
Vice President, Supply

STATE OF MARYLAND :
: TO WIT:
CITY OF BALTIMORE :

Joseph A. Tiernan, being duly sworn states that he is Vice President of the Baltimore Gas and Electric Company, a corporation of the State of Maryland; that he provides the foregoing response for the purposes therein set forth; that the statements made are true and correct to the best of his knowledge, information and belief; and that he was authorized to provide the response on behalf of said Corporation.

WITNESS my Hand and Notarial Seal: Minnie L. Robinson, Nov. 18, 1985

My Commission Expires: July 1, 1986

JAT/RCLO/dmk

Attachments:
1. The Rule and Its Underlying Purpose.
2. Material Circumstances Not Considered When the Regulation in Question was Adopted.

cc: D. H. Jaffe
Dr. T. E. Murley
D. A. Brune, Esq.
G. F. Trowbridge, Esq.

ATTACHMENT 1

THE RULE AND ITS UNDERLYING PURPOSE

1. Introduction

This attachment discusses our determination that a plume exposure pathway EPZ beyond 2 miles is not necessary to achieve the underlying purpose of the rule[1] as applied to the Calvert Cliffs Nuclear Power Plant (CCNPP).

10CFR50.47(c)(2) states, in part:

> "Generally, the plume exposure pathway EPZ for nuclear power plants shall consist of an area about 10 miles (16km) in radius..."

Footnote 1 to 10CFR50 Appendix E cites NUREG-0396[2] for a discussion of EPZs for power reactors. NUREG-0396 was prepared by a joint U.S. Nuclear Regulatory Commission and U.S. Environmental Protection Agency Task Force on Emergency Planning. As the planning basis document for emergency planning, it represents the primary published source of understanding of the underlying purpose of the existing rule. Specifically, NUREG-0396 (page 24) states:

> "The establishment of Emergency Planning Zones of about 10 miles for the plume exposure pathway . . . is sufficient to scope the areas in which planning for the initiation of predetermined protective action is warranted for any given nuclear plant."

The following sections summarize (a) the underlying purpose for the establishment of the plume exposure pathway EPZ and the basis for 10 miles as an appropriate EPZ radius, as discussed in NUREG-0396, and (b) the achievement of the underlying purpose of the subject rule in the particular circumstances of CCNPP.

2. The Underlying Purpose of the Plume Exposure Pathway EPZ and the Basis for 10 Miles

Insight into the underlying purpose of the rule establishing a plume exposure pathway EPZ radius (of about 10 miles) may be gained by referring to the following statement on page 5 of NUREG-0396:

> "The Task Force concluded that the objective of emergency response plans should be to provide dose savings for a spectrum of accidents that could produce offsite doses in excess of the PAGs."

Dose savings would be achieved by protective actions within the plume exposure pathway EPZ, so that the doses in question would not be received by individuals. Therefore, the underlying purpose of this plume exposure pathway EPZ is to establish an area within

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1 10CFR50.47(c)(2) and Appendix E.

2 "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants", NUREG-0396 (EPA-52011-78-016), U.S. Nuclear Regulatory Commission, December 1978.

which planning of predetermined protective actions would be expected to result in dose savings comparable to those discussed in NUREG-0396.

The following quotations from NUREG-0396 further address the conclusions and recommendations of the Task Force with regard to the basis for the establishment of a planning radius for the plume exposure pathway EPZ:

In discussing the need for planning zones, NUREG-0396 on page 8 and 11, respectively, states:

> "The most important guidance for planning officials is the distance from the nuclear facility which defines the area over which planning for pre-determined actions should be carried out. The other elements of guidance provide supporting information for planning and preparedness."

> "With regard to the area over which planning efforts should be carried out, the Task Force recommends that 'Emergency Planning Zones' (EPZs) about each nuclear facility be defined both for the short term 'plume exposure pathway' and for the longer term 'ingestion exposure pathways.'"

With regard to consideration of potential accidents which might result in the above mentioned exposures, NUREG-0396 states, on page 24 and 15, respectively:

> "A spectrum of accidents (not the source term from a single accident sequence) should be considered in developing a basis for emergency planning."

> "The Task Force agreed that emergency response plans should be useful for responding to any accident that would produce offsite doses in excess of the PAGs. This would include the more severe design basis accidents and the accident spectrum analyzed in the RSS. After reviewing the potential consequences associated with these types of accidents, it was the concensus (sic) of the Task Force that emergency plans could be based upon a generic distance out to which predetermined actions would provide dose savings for any such accidents. Beyond this generic distance it was concluded that actions could be taken on an ad hoc basis using the same considerations that went into the initial action determinations." (<u>NOTE:</u> PAGs - Protective Action Guides;[3] RSS - Reactor Safety Study[4].)

In reaching the conclusion on page 24 that a plume exposure pathway EPZ "of about 10 miles. . .is sufficient", NUREG-0396 cites technical data contained in Appendix I to that report, entitled "Rationale for the Planning Basis". Appendix I includes a discussion of various rationales for establishing a planning basis.

The consequences of Design Basis Accidents (DBAs) were calculated for a large number of nuclear power plants. Those calculations were related to the PAGs, <u>i.e.,</u> Figure I-8 of Appendix I to NUREG-0396.

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3 "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents", EPA-521/1-75-001, U.S. Environmental Protection Agency, September 1975.

4 Reactor Safety Study - "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400 (NUREG-75/1014), U.S. Nuclear Regulatory Commission, October 1975.

It should be noted that NUREG-0396 expressed explicit caution in the use of PAGs in establishing requirements for emergency response plans. On page 4, NUREG-0396 states:

> "The nature of PAGs is such that they cannot be used to assure that a given level of exposure to individuals in the population is prevented. In any particular response situation, a range of doses may be experienced, principally depending on the distance from the point of release. Some of these doses may be well in excess of the PAG levels and clearly warrant the initiation of any feasible protective actions. This does not mean, however, that doses above PAG levels can be prevented or that emergency response plans should have as their objective preventing doses above PAG levels." (emphasis added)

Figure I-11 of Appendix I to NUREG-0396 was a primary figure in the development of the rationale for recommending a planning basis incorporating a 10 mile radius for the plume exposure pathway EPZ. The data in that figure were based on the then-current, best available data from the Reactor Safety Study.

Taken together, the data presented in Appendix I of NUREG-0396 and the results of the investigation summarized therein formed the basis for the establishment of the plume exposure pathway EPZ with a radius of about 10 miles.

3. Achieving the Underlying Purpose of the Rule

As noted in NUREG-0396, the principal exposure sources in the plume exposure pathway are: (a) whole body external exposure to gamma radiation from the plume and from deposited material; and (b) inhalation exposure from the passing radioactive plume.

Analyses of these exposure sources in the plume exposure pathway have been performed for CCNPP covering a spectrum of postulated accidents, as presented in Figures 1-1 and 1-2. These analyses include the Design Basis Accidents (DBA), as reported in the Final Safety Analysis Report (FSAR) for CCNPP,[5] and the more severe "Class 9" accidents. The latter are based on updated source terms (i.e., the magnitude, type and timing of postulated releases of radionuclides to the environment), as presented in Attachment 2, in combination with the Calvert Cliffs site-specific meteorological data, as reported in the FSAR. These latter analyses were performed with the CRAC2 computer code[6]. The CRAC2 analyses assumed no evacuation and 24-hour normal activity (i.e., no special sheltering.)

The solid curve in Figure 1-1 depicts the whole body dose as a function of distance from the point of release for the DBA at CCNPP. This curve was reproduced from Figure 14.24-2 of the FSAR. It can be observed from this curve that the 5 rem PAG is not exceeded beyond 0.4 mile and the 1 rem PAG is not exceeded beyond 1.4 miles.

Whole body doses as a function of distance, as reported in Figure I-8 of NUREG-0396, are

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5 "Updated Final Safety Analysis Report - Calvert Cliffs Nuclear Power Plants - Units 1 and 2", Baltimore Gas and Electric Company, Dockets 50-317 and 50-318.

6 "CRAC2: Calculations of Nuclear Reactor Accident Consequences, Version 2, NUREG/CR-2326, Sandia National Laboratories, February 1983.

included here as the dashed and dotted curves in Figure 1-1, representing the median (50%) and highest 10% of calculations for 67 power plant sites considered in NUREG-0396.

From Figure 1-1, it can be seen that a comparable dose, e.g. 0.5 rem, is projected at a distance of 2 miles for CCNPP and at a distance of 10 miles for the highest 10% curve obtained from NUREG-0396. Thus, dose savings with a 2 mile EPZ at CCNPP would be comparable to dose savings with a 10 mile EPZ, based on the NUREG-0396 data.

Figure 1-2 presents data depicting the probability of experiencing a 200 rem whole body dose, at which significant early injuries may start to occur, as a function of distance from the point of release for the (more severe) Class 9 accidents. This figure includes an adaptation of the data presented in Figure I-11 of NUREG-0396 (i.e., the dotted curve in Figure 1-2) and two curves depicting the data for a range of conditions at CCNPP.

The dotted curve comprises the corresponding curve in Figure I-11 of NUREG-0396, as adapted for Figure 1-2 by multiplying the conditional probabilities (conditional on a core melt accident occurring) times the WASH-1400 core melt probability of 5×10^{-5} per reactor year, reported in Figure I-11.

Two CCNPP site-specific curves are included in Figure 1-2, based on the recent source terms presented in Attachment 2. Both curves include contributions from the in-containment accident sequences. The difference between the two curves is due to the different contributions from the two containment bypass "V" sequences.

As discussed in Attachment 2, "low range" and "high range" V sequence source terms have been established for CCNPP. When the high range V sequence source term is included along with the in-containment source terms, the data depicted in the "V-High" curve of Figure 1-2 result. When the low range V sequence source term is included along with the in-containment source terms, the data depicted in the "V-Low" curve result. Thus, these two curves depict the effect of the range of potential V sequences on the probability of experiencing a 200 rem dose as a function of distance. The fact that the two curves are close together is an illustration of the large contributions from the in-containment source terms obtained from NUREG-0956.[7] In the cases presented in this figure, the probability of experiencing a 200 rem dose at any distance is substantially lower for CCNPP, with the current source terms, than was the case in the NUREG-0396 analysis, with WASH-1400 source terms.

The CCNPP dose curves are observed to drop off substantially at a distance of about 2 miles, whereas the NUREG-0396 dose curve is observed to drop off substantially at about 10 miles. Thus, dose savings with a 2 mile EPZ at CCNPP, based on recent source terms, are comparable to dose savings with a 10 mile EPZ, based on WASH-1400 source terms as reported in NUREG-0396.

Taken together, the data presented in Figure 1-1 and 1-2 illustrate that the underlying purpose of the EPZ plume exposure pathway rule, as discussed above, is achieved at a radius of 2 miles at the Calvert Cliffs Nuclear Power Plant.

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7 Silberberg, M. et. al., "Reassessment of the Technical Bases for Estimating Source Terms", NUREG-0956 Draft Report for Comment, U.S. Nuclear Regulatory Commission, July 1985.

FIGURE 1-1

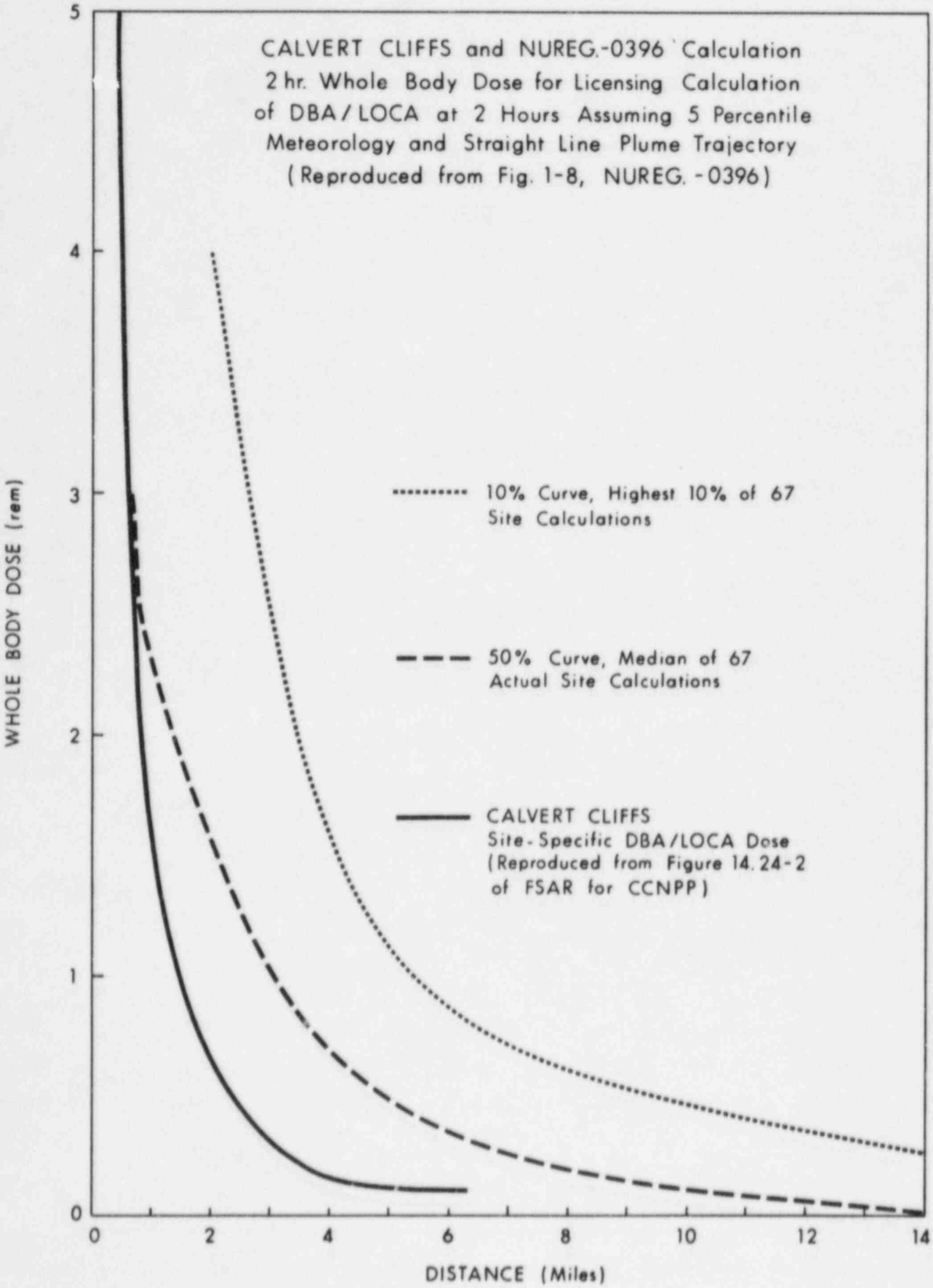
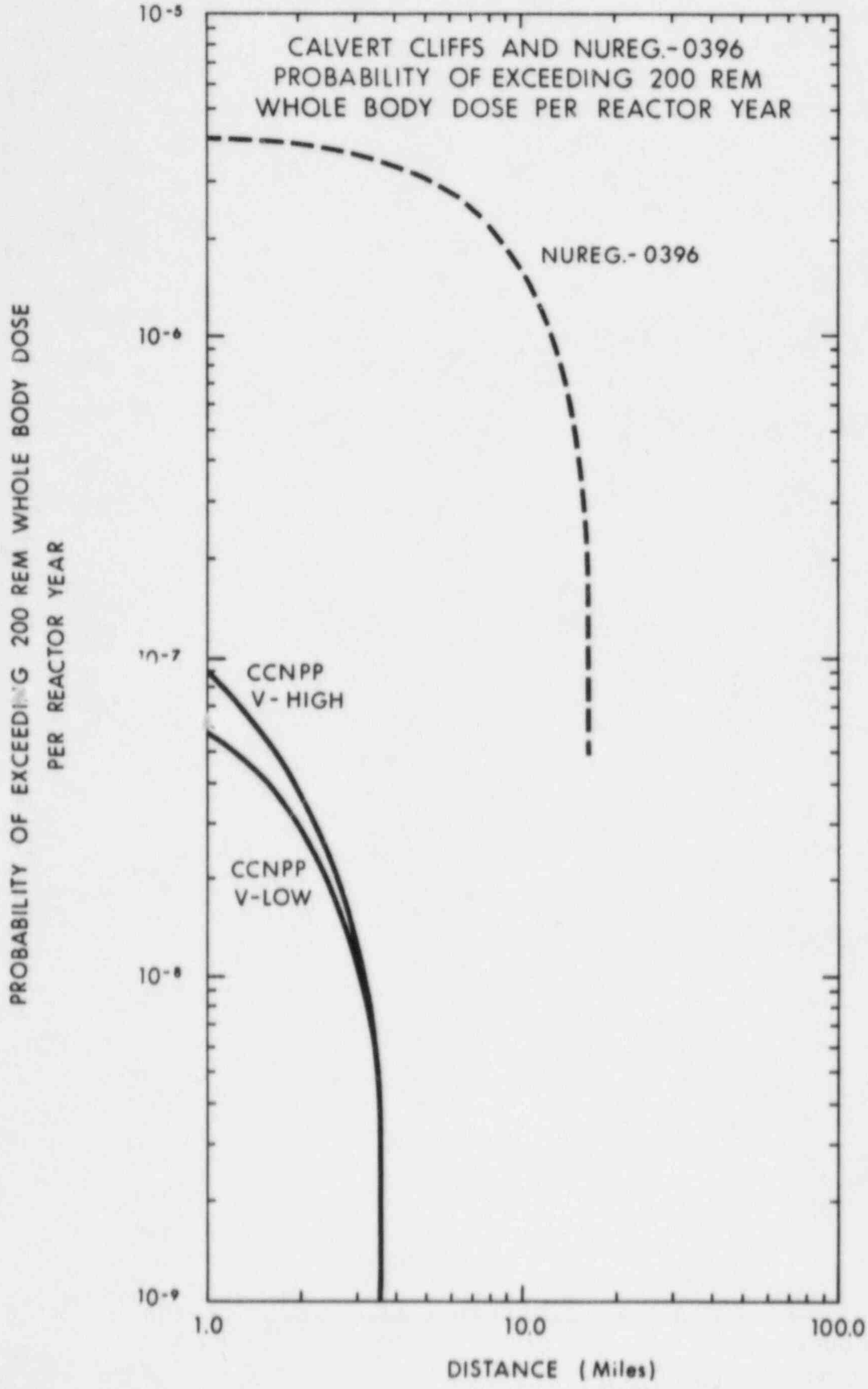

CALVERT CLIFFS and NUREG.-0396 Calculation
2 hr. Whole Body Dose for Licensing Calculation
of DBA / LOCA at 2 Hours Assuming 5 Percentile
Meteorology and Straight Line Plume Trajectory
(Reproduced from Fig. 1-8, NUREG. -0396)
10% Curve, Highest 10% of 67
Site Calculations
50% Curve, Median of 67
Actual Site Calculations
CALVERT CLIFFS
Site-Specific DBA/LOCA Dose
(Reproduced from Figure 14.24-2
of FSAR for CCNPP)
WHOLE BODY DOSE (rem)
0
1
2
3
4
5
0
2
4
6
8
10
12
14
DISTANCE (Miles)

FIGURE 1-2


CALVERT CLIFFS AND NUREG.-0396
PROBABILITY OF EXCEEDING 200 REM
WHOLE BODY DOSE PER REACTOR YEAR
NUREG.- 0396
CCNPP
V- HIGH
CCNPP
V-LOW
PROBABILITY OF EXCEEDING 200 REM WHOLE BODY DOSE
PER REACTOR YEAR
DISTANCE (Miles)
1.0
10.0
100.0

ATTACHMENT 2

MATERIAL CIRCUMSTANCES NOT CONSIDERED WHEN THE REGULATION IN QUESTION WAS ADOPTED

1.0 Introduction

The requirement of 10CFR 50.47 for a plume exposure pathway Emergency Planning Zone (EPZ) with a radius of about 10 miles[1] is discussed in NUREG-0396,[2] as noted in footnote 1 to 10CFR50 Appendix E, Section I.[3]

The technical bases for the recommendations contained in NUREG-0396 are predicated on a review of a spectrum of potential accidents ranging from the Design Basis Accidents (DBAs) to the more severe "Class 9" accidents. The analysis of the Class 9 accidents in NUREG-0396 is based on source terms (i.e., the magnitude, type, and timing of postulated releases of radionuclides to the environment) as reported in the Reactor Safety Study (WASH-1400),[4] which represented the best technical information available at the time.

There are now present other material circumstances not available for consideration when the regulation was adopted based on the WASH-1400 data. Specifically, vast improvement in the knowledge and understanding of source terms for postulated severe core damage accidents has occurred as a result of the extensive national and international reassessment of severe accident source terms. This attachment addresses the recent improvements in source term knowledge and the application of that knowledge to the Calvert Cliffs Nuclear Power Plant (CCNPP).

2.0 Recent Improvements in Source Term Knowledge

2.1 Background

WASH-1400 addressed a broad range of potential accident sequences and assigned probabilities of occurrence to each sequence. It also addressed the release of radionuclides for the various sequences and grouped the sequences into release categories. Nine release categories were developed for Pressurized Water Reactors (PWRs) and five release categories were developed for Boiling Water Reactors (BWRs). For each of these release categories, the authors of WASH-1400 established source

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1 10CFR50.47(c)(2).

2 "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Reactor Nuclear Power Plants", NUREG-0396 (EPA-52011-78-016), U.S. Nuclear Regulatory Commission, December 1978.

3 10CFR50 Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities".

4 "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400 (NUREG-75/104), U.S. Nuclear Regulatory Commission, October 1975.

terms, which included estimated releases to the environment within seven fission product groups. These seven groups can be categorized into three categories of releases - noble gases (xenon and krypton), volatile species (iodine, cesium-rubidium, and the tellurium-antimony groups), and non-volatile species (barium-strontium, ruthenium, and the lanthanum groups.) For the purposes of this exemption request, only the PWR sequences and releases are specifically discussed.

The first of these categories of releases, noble gases, is not influenced appreciably by plant features and physical phenomena, except for the timing of release. A delay in the time of release affects the noble gas source term because it allows time for natural radioactive decay to reduce the amount of radioactivity available for release to the environment. For accident sequences involving postulated early releases to the environment, the WASH-1400 investigations assumed essentially complete release of the noble gases to the environment. The recent assessment of severe accident source terms has not altered that assumption. With respect to the two other categories of releases, volatiles and non-volatiles, the new knowledge and improved understanding resulting from the extensive investigation of severe accident source terms has resulted in substantive reductions in the estimated fraction of the core inventory of these radionuclides which would be expected to be released to the environment (i.e., substantially reduced source terms when compared with WASH-1400, as discussed in subsequent sections of this attachment.)

One early finding concerning improvement in the methodology of estimating releases to the environment during severe accidents was reported in an August 4, 1980 letter[5] addressed to the then Chairman of the Nuclear Regulatory Commission, J. Ahearne, from three respected scientists, W. R. Stratton of the Los Alamos Scientific Laboratory, and A. P. Malinauskas and D. O. Campbell of the Oak Ridge National Laboratory, which addressed the releases observed during the accident at Three Mile Island Unit 2 (TMI-2) in March of 1979. In that letter, the authors stated, in part, 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." The low releases observed at TMI-2 and a call for further investigation were discussed in a December 21, 1980 letter[6] addressed to President Carter from Governor Babbitt (of Arizona) and other members of the Nuclear Safety Oversight Committee established by the President in the aftermath of TMI-2. In the period from 1980 to the present, a truly massive investment of resources has been expended by both government and private industry to improve the knowledge and understanding of the potential release of radionuclides during the postulated accidents at nuclear power plants.

2.2 Summary of Recent Investigations

The American Nuclear Society (ANS) established a Special Committee on Source Terms in June of 1982 to review and evaluate the state of knowledge of how to predict source terms deriving from severe core damage accidents, and to summarize the findings of various investigating organizations. The state of knowledge and associated findings were

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5 Stratton, W. R., Malinauskas, A. P. and Campbell, D. O., letter to NRC Chairman, J. Ahearne, August 14, 1980.

6 Babbitt, B., Deutch, J., Goldberger, M., and Lewis, H., Letter to the President from Nuclear Safety Oversight Committee (NSOC), December 1980.

compared to those of WASH-1400. The ANS Committee published its report[7] in September 1984 and summarized the results at a meeting of the Nuclear Regulatory Commission in November 1984.

The major finding of the ANS Special Committee was, in part, as follows:

> "Source items for severe core damage accidents have been overestimated by large factors both in government and industry publications. With a small number of exceptions, estimates of source terms associated with severe core damage accidents can be reduced from estimates in WASH-1400 by more than an order of magnitude to several orders of magnitude. The noble gas fission products (krypton and xenon) are exceptions. Because of their chemically inert character, they do not undergo the wide range of chemical and physical interactions which are the fundamental cause of the reduced release of most fission products. However, the very fact that they are inert also leads to low radiological consequences from their release. Specifically, the Committee finds:
>
>> For large dry PWR containments, sufficient information exists to support the calculation of source terms ranging from a small fraction of a percent to no more than a few percent of the core inventory of important fission products species."

The Committee also reported a number of specific findings relative to other plant types and other findings supporting or qualifying the major finding. Those findings addressed such areas as containment integrity, thermal hydraulics, fission product transport and deposition, and important radionuclides and chemical forms.

In recognition of the importance and complexity of the investigation of severe accident source terms, the NRC established an Accident Source Term Program Office (ASTPO) to coordinate the NRC's investigations. A major undertaking of that office was the management of an extensive study by Battelle Columbus Laboratories, the results of which were published in a multi-volume report, designated BMI-2104.[8] In further recognition of the importance and complexity of the subject, ASTPO organized and conducted extensive public peer review meetings during the course of the development of BMI-2104.

In order to provide an independent peer review by scientists not normally associated with nuclear power plant studies, ASTPO commissioned an investigation by the American Physical Society (APS) which formed a Study Group on Radionuclide Release from Severe Accidents at Nuclear Power Plants. The APS Study Group completed its report[9] in February 1985 and briefed the Nuclear Regulatory Commission in March 1985 on its findings and recommendations.

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7 "Report of the Special Committee on Source Terms", American Nuclear Society, September 1984.

8 Gieske, J. A., et. al., "Radionuclide Release Under Specific LWR Accident Conditions", BMI-2104, Battelle Columbus Laboratories, July 1984.

9 "Draft Report of the American Physical Society Study Group on Radionuclide Release from Severe Accidents at Nuclear Power Plants", American Physical Society, February 1985.

The APS Study Group's principal conclusions were as follows:

> "The study group finds that considerable progress has been made since publication of the Reactor Safety Study (NRC 1975) in developing both a scientific basis and calculational ability for predicting the source term. In a number of cases, new calculations indicate that the quantity of radionuclides that would reach the environment is significantly lower than that calculated in the Reactor Safety Study. This reduction can be attributed to three principal factors: (1) the recognition that reactor containments are stronger than assumed in the Reactor Safety Study and therefore fail, if at all, at later times; (2) inclusion in the modeling of previously neglected physical and chemical phenomena that lead to the retention of fission products; and (3) inclusion of additional sites (suppression pools, ice beds, auxiliary buildings) that trap radionuclides more efficiently than previously assumed."
>
> "The study group examined the chemical and physical phenomena considered by the technical community since the Reactor Safety Study was completed. For most sequences and most radionuclides, these phenomena reduce the source term from that calculated in the Reactor Safety Study."
>
> "However, one mechanism that might, for some sequences, increase the radionuclide releases above those calculated in the Reactor Safety Study is the release of non-volatile radionuclides in the core-concrete interaction. It is important to complete the experiments now underway to improve our knowledge of the physics and chemistry in this crucial area. Moreover, the analyses performed in the recent studies that we have surveyed have not treated all types of reactors nor all types of containments in equal detail. It is impossible to make the sweeping generalization that the calculated source term for any accident sequence involving any reactor plant would always be a small fraction of the fission product inventory at reactor shutdown. Although further studies may improve this situation, some of the reasons for this inability are enumerated . . .".

The Study Group report went on to discuss details of the above conclusions.

Subsequent to the publication of the ANS and APS reports, and reports of a number of industry organizations, such as the Industry Degraded Core Rulemaking (IDCOR) program, the Electric Power Research Institute (EPRI), the Stone and Webster Engineering Corporation, and others, the NRC published a Draft Report for comment, entitled "Reassessment of the Technical Bases for Estimating Source Terms" - NUREG-0956, in July 1985.[10]

The first conclusion provided by NUREG-0956 is:

> "The BMI-2104 suite of computer codes represents a major advance in technology <u>and can be used to replace the Reactor Safety Study methods."</u>[11] (emphasis added.)

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10 Silberberg, M. et. al., "Reassessment of the Technical Bases for Estimating Source Terms", NUREG-0956 Draft Report for Comment, U.S. Nuclear Regulatory Commission, July 1985.

11 Ibid, p. 8-1.

Further, Recommendation #1 of NUREG-0956 is:

> "The new source term analytical methods should be used to reevaluate regulatory practices that are based on the Reactor Safety Study methods. Insights from new analyses should be applied to reconsider the use of TID-14844 assumptions. Improvements are so significant that utilization of the new methods is warranted while additional confirmatory research is being completed." (emphasis added.)

NUREG-0956 goes on to state:

> "In light of all the above conclusions, it is recommended that the new analytical methods should be used to reevaluate current regulatory practices and revise them as needed."[12] (emphasis added.)

2.3 PWR Source Terms Reported in NUREG-0956

Although the primary purpose of NUREG-0956 is to document revised and improved models which describe the science and engineering of the phenomena that occur during and after a spectrum of core damage events, the results of calculations for a range of severe accidents for five reference plants (i.e., as reported in BMI-2104) were examined in the preparation of the report.

Particular emphasis is placed on a rather complete reanalysis of the Surry PWR plant in NUREG-0956. That plant was analyzed as the representative plant for PWRs in WASH-1400. Thus, the results of the reassessment of source terms for Surry, as reported in NUREG-0956, may be used in a direct comparison of the results obtained with the revised and improved models with those reported in the earlier WASH-1400 report.

Table 2-1 is a reproduction of Table 5-2 from the Main Report in the WASH-1400 series. This table lists the accident sequences considered for Surry in that investigation. Table 2-2 is a reproduction of Table 5-1 from the same WASH-1400 document. This table presents a summary of the source terms for the nine PWR (and five BWR) release categories in WASH-1400.

Analyses indicate that the low probability/high release source terms represented by release categories PWR-1, 2 and 3 constitute virtually all of the whole body exposure risk from the nine PWR release categories. (Risk is defined as the product of the probability of occurrence times the calculated consequences.) The PWR-2 release category dominates the risk, contributing approximately 75% of the whole body exposure risk. Release categories PWR-1 and 3, combined, represent approximately 25% of the exposure risk. Thus, much of the emphasis in the recent national reassessment of severe accident source terms has been concentrated on the accident sequences that are included in these release categories. This is the case in NUREG-0956, as well.

Table 2-3 is a summary of the source terms for the accident sequences analyzed for Surry, as reported in NUREG-0956.[13] The TMLB' and V sequences combined to dominate the releases in the PWR-2 release category in WASH-1400. A comparison between the

12 Ibid., p. 8-6.

13 Ibid., Table 4.13.

TABLE 2-1

REPRODUCTION OF TABLE 5-2 FROM WASH-1400

TABLE 5-2 PWR DOMINANT ACCIDENT SEQUENCES vs. RELEASE CATEGORIES

	RELEASE CATEGORIES Core Melt							No Core Melt	
	1	2	3	4	5	6	7	8	9
LARGE LOCA A	AB-α 1×10^{-11} AF-α 1×10^{-10} ACD-α 5×10^{-11} AG-α 9×10^{-11}	AB-γ 1×10^{-10} AB-δ 4×10^{-11} AHF-γ 2×10^{-11}	AD-α 2×10^{-8} AH-α 1×10^{-8} AF-δ 1×10^{-8} AG-δ 9×10^{-9}	ACD-β 1×10^{-11}	AD-β 4×10^{-9} AH-β 3×10^{-9}	AB-ϵ 1×10^{-9} AHF-ϵ 1×10^{-10} ADF-ϵ 2×10^{-10}	AD-ϵ 2×10^{-6} AH-ϵ 1×10^{-6}	A-β 2×10^{-7}	A 1×10^{-4}
A Probabilities	2×10^{-9}	1×10^{-8}	1×10^{-7}	1×10^{-8}	4×10^{-8}	3×10^{-7}	3×10^{-6}	1×10^{-5}	1×10^{-4}
SMALL LOCA S_1	S_1B-α 3×10^{-11} S_1CD-α 7×10^{-11} S_1F-α 3×10^{-10} S_1G-α 3×10^{-10}	S_1B-γ 4×10^{-10} S_1B-δ 1×10^{-10} S_1HF-γ 6×10^{-11}	S_1D-α 3×10^{-8} S_1H-α 3×10^{-8} S_1F-δ 3×10^{-8} S_1G-δ 3×10^{-8}	S_1CD-β 1×10^{-11}	S_1H-β 5×10^{-9} S_1D-β 6×10^{-9}	S_1DF-ϵ 3×10^{-10} S_1B-ϵ 2×10^{-9} S_1HF-ϵ 4×10^{-10}	S_1D-ϵ 3×10^{-6} S_1H-ϵ 3×10^{-6}	S_1-β 6×10^{-7}	S_1 3×10^{-4}
S_1 Probabilities	3×10^{-9}	2×10^{-8}	2×10^{-7}	3×10^{-8}	8×10^{-8}	6×10^{-7}	6×10^{-6}	3×10^{-5}	3×10^{-4}
SMALL LOCA S_2	S_2B-α 1×10^{-10} S_2F-α 1×10^{-9} S_2CD-α 2×10^{-10} S_2G-α 9×10^{-10} S_2C-α 2×10^{-8}	S_2B-γ 1×10^{-9} S_2HF-γ 2×10^{-10} S_2B-δ 4×10^{-10}	S_2D-α 9×10^{-8} S_2H-α 6×10^{-8} S_2F-δ 1×10^{-7} S_2C-δ 2×10^{-6} S_2G-δ 9×10^{-8}	S_2DG-β 1×10^{-12}	S_2D-β 2×10^{-8} S_2H-β 1×10^{-8}	S_2B-ϵ 8×10^{-9} S_2CD-ϵ 2×10^{-9} S_2HF-ϵ 1×10^{-9}	S_2D-ϵ 9×10^{-6} S_2H-ϵ 6×10^{-6}		
S_2 Probabilities	1×10^{-7}	3×10^{-7}	3×10^{-6}	3×10^{-7}	3×10^{-7}	2×10^{-6}	2×10^{-5}		
REACTOR VESSEL RUPTURE - R	RC-α 2×10^{-12}	RC-γ 3×10^{-11} RF-δ 1×10^{-11} RC-ϵ 1×10^{-12}	R-α 1×10^{-9}				R-ϵ 1×10^{-7}		
R Probabilities	2×10^{-11}	1×10^{-10}	1×10^{-9}	2×10^{-10}	1×10^{-9}	1×10^{-8}	1×10^{-7}		
INTERFACING SYSTEMS LOCA (CHECK VALVE) - V		V 4×10^{-6}							
V Probabilities	4×10^{-7}	4×10^{-6}	4×10^{-7}	4×10^{-8}					
TRANSIENT EVENT - T	TMLB'-α 3×10^{-8}	TMLB'-γ 7×10^{-7} TMLB'-δ 2×10^{-7}	TML-α 6×10^{-8} TKQ-α 3×10^{-8} TKMQ-α 1×10^{-8}		TML-β 3×10^{-10} TKQ-β 1×10^{-10}	TMLB'-ϵ 6×10^{-7}	TML-ϵ 6×10^{-6} TKQ-ϵ 3×10^{-6} TKMQ-ϵ 1×10^{-6}		
T Probabilities	1×10^{-7}	3×10^{-6}	4×10^{-7}	1×10^{-9}	2×10^{-7}	2×10^{-6}	1×10^{-5}		
(Σ) SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORY									
MEDIAN (50% VALUE)	9×10^{-7}	8×10^{-6}	4×10^{-6}	5×10^{-7}	7×10^{-7}	6×10^{-6}	4×10^{-5}	4×10^{-5}	4×10^{-4}
LOWER BOUND (5% VALUE)	9×10^{-8}	8×10^{-7}	6×10^{-7}	9×10^{-9}	2×10^{-7}	2×10^{-6}	1×10^{-5}	4×10^{-6}	4×10^{-5}
UPPER BOUND (95% VALUE)	9×10^{-6}	8×10^{-5}	4×10^{-5}	5×10^{-6}	4×10^{-6}	2×10^{-5}	2×10^{-4}	4×10^{-4}	4×10^{-3}

Note: The probabilities for each release category for each event tree and the Σ for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability

TABLE 2-2

REPRODUCTION OF TABLE 5-1 FROM WASH-1400

TABLE 5-1 SUMMARY OF ACCIDENTS INVOLVING CORE

RELEASE CATEGORY	PROBABILITY per Reactor-Yr	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	WARNING TIME FOR EVACUATION (Hr)	ELEVATION OF RELEASE (Meters)	CONTAINMENT ENERGY RELEASE (10^6 Btu/Hr)	FRACTION OF CORE INVENTORY RELEASED[a] Xe-Kr	Org. I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[b]	La[c]
PWR 1	9×10^{-7}	2.5	0.5	1.0	25	520[d]	0.9	6×10^{-3}	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR 2	8×10^{-6}	2.5	0.5	1.0	0	170	0.9	7×10^{-3}	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR 3	4×10^{-6}	5.0	1.5	2.0	0	6	0.8	6×10^{-3}	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR 4	5×10^{-7}	2.0	3.0	2.0	0	1	0.6	2×10^{-3}	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR 5	7×10^{-7}	2.0	4.0	1.0	0	0.3	0.3	2×10^{-3}	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR 6	6×10^{-6}	12.0	10.0	1.0	0	N/A	0.3	2×10^{-3}	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR 7	4×10^{-5}	10.0	10.0	1.0	0	N/A	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR 8	4×10^{-5}	0.5	0.5	N/A	0	N/A	2×10^{-3}	5×10^{-6}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR 9	4×10^{-4}	0.5	0.5	N/A	0	N/A	3×10^{-6}	7×10^{-9}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0
BWR 1	1×10^{-6}	2.0	2.0	1.5	25	130	1.0	7×10^{-3}	0.40	0.40	0.70	0.05	0.5	5×10^{-3}
BWR 2	6×10^{-6}	30.0	3.0	2.0	0	30	1.0	7×10^{-3}	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR 3	2×10^{-5}	30.0	3.0	2.0	25	20	1.0	7×10^{-3}	0.10	0.10	0.30	0.01	0.02	3×10^{-3}
BWR 4	2×10^{-6}	5.0	2.0	2.0	25	N/A	0.6	7×10^{-4}	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1×10^{-4}
BWR 5	1×10^{-4}	3.5	5.0	N/A	150	N/A	5×10^{-4}	2×10^{-9}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0

(a) A discussion of the isotopes used in the study is found in Appendix VI. Background on the isotope groups and release mechanisms is found in Appendix VII.

(b) Includes Mo, Rh, Tc, Co.

(c) Includes Nd, Y, Ce, Pr, La, Nb, Am, Cm, Pu, Np, Zr.

(d) A lower energy release rate than this value applies to part of the period over which the radioactivity is being released. The effect of lower energy release rates on consequences is found in Appendix VI.

TABLE 2-3

PWR SOURCE TERMS IN NUREG-0956
(Surry)

Accident Sequence	TMLB'-δ	TMLB'-β	TMLB'-ε	V Submerged	V Unsubmerged	AB-γ	AB-β	AB-ε	S_2D-γ
Probability per Reactor Year	1.7×10^{-8}	6.6×10^{-9}	3.0×10^{-7}	3×10^{-6}	1×10^{-6}	1×10^{-10}	$< 10^{-10}$	$< 10^{-10}$	5.5×10^{-8}
Release Time, Hr	2.5	2.0	12	1.0	1.0	4.5	0.5	24	2.5
Fission Product Group	Fraction of Core Inventory Released to Environment								
Xe-Kr	0.85	1.00	0.80	1.00	1.00	0.80	0.90	0.15	0.50
I-Br	0.07	0.022	0.0028	0.08	0.40	0.057	0.087	0.000048	0.005
Cs-Rb	0.058	0.013	0.00039	0.08	0.40	0.06	0.087	0.000047	0.0001
Te-Sb	0.055	0.11	0.085	0.025	0.12	0.14	0.066	0.00004	0.01
Ba-Sr	0.01	0.058	0.018	0.0022	0.011	0.097	0.076	< 0.00001	0.03
Ru	0.0013	0.0053	< 0.00001	0.00013	0.00065	0.0024	0.0029	< 0.00001	0.001
La	0.0017	0.0002	0.0001	0.00007	0.00035	0.008	0.0075	0.000036	0.0009

WASH-1400 and NUREG-0956 release fraction terms, which apply to these sequences, is presented in Table 2-4. From this comparison it can be observed that, excluding noble gases, the source terms using the revised and improved models in NUREG-0956 are generally substantially lower than those reported in WASH-1400, with the exception of the V-unsubmerged sequence.

NUREG-0956 addresses the analyses for two variations on the V sequence for Surry. the V sequence includes a postulated unisolated break in the low pressure portion of the low pressure safety injection system (LPIS) outside containment. This sequence constitutes a containment bypass sequence which permits direct release from the reactor coolant system to a contiguous structure. These analyses only apply to Surry and are not applicable to other plants, such as CCNPP.

With regard to the large break loss of coolant (LOCA) sequences, AB-γ, AB-β, and AB-ε, the probabilities of occurrence are so low, i.e., less than 1×10^{-10} per reactor year, that they were discarded from further consideration in NUREG-0956. The release fractions for the small break LOCA analysis, S_2D, are applicable to other PWRs, as is the case with the TMLB' analyses. The probability of occurrence of these sequences may be different than for Surry, based on plant-specific features.

3.0 Application of Recent Source Term Knowledge to CCNPP

In applying the recent source term knowledge to CCNPP, the approach adopted was to utilize the source terms reported in NUREG-0956 for Surry for the "in-containment" accident sequences and develop CCNPP plant-specific source terms for the containment bypass V sequence.

The first step in determining the applicability to CCNPP of source terms based on analyses of Surry is to compare the plant features. Section 3.1 summarizes a comparison of plant features for Surry, Zion and Calvert Cliffs.

In addition to the comparison of plant features, an engineering evaluation of important accident sequences at Calvert Cliffs was conducted, as summarized in Section 3.2. This evaluation concentrated on sequences which represent major contributions to calculated core melt frequency. Many of these sequences would result in low source terms owing to the operation of engineered safety features.

Severe accident source terms were included in the analysis of offsite consequences at CCNPP based on a combination of the results of the plant features comparison discussed in Section 3.1 and the engineering evaluation discussed in Section 3.2. The source terms and their associated probability of occurrence are discussed in Section 3.3.

3.1 Comparison of Plant Features

Table 2-5 lists plant features for Surry, Zion and Calvert Cliffs which are important with regard to severe accident source terms analysis. The following discussion addresses each feature or group of features.

Features 1 and 2

Both Surry and Zion are Westinghouse pressurized water reactors (PWRs) whereas Calvert Cliffs is a Combustion Engineering PWR. However, their reactors and reactor coolant systems (RCS) are sufficiently similiar from a fission product transport standpoint to expect comparable results for the RCS portions of the source term

TABLE 2-4

COMPARISON OF WASH-1400 AND NUREG-0956 PWR RELEASE FRACTIONS
(Based on Analysis of Surry)

Fraction of Core Inventory Released to the Environment

Fission Product Group	WASH-1400 Rel. Category PWR-2	NUREG-0956					
		TMLB'-δ_e	TMLB'-β	TMLB'-ε	V Submerged	V Unsubmerged	S_2D-γ
Xe-Kr	0.90	0.85	1.00	0.80	1.00	1.00	(See Note and Table 2-3)
I-Br	0.70	0.07	0.022	0.0028	0.08	0.40	
Cs-Rb	0.50	0.058	0.013	0.00039	0.08	0.40	
Te-Sb	0.30	0.055	0.11	0.085	0.025	0.12	
Ba-Sr	0.06	0.01	0.058	0.018	0.0022	0.011	
Ru	0.02	0.0013	0.0053	$<$ 0.00001	0.00013	0.00065	
La	0.004	0.0017	0.0002	0.0001	0.00007	0.00035	

(NOTE: The S_2D-γ sequence was not explicitly addressed in WASH-1400. However, the authors of NUREG-0956 have assigned a comparable WASH-1400 Release Category of PWR-5, which is about 1-2 orders of magnitude lower than PWR-2).

Table 2-5

COMPARISON OF SURRY, ZION, AND CALVERT CLIFFS PLANT FEATURES

	Feature	Surry	Zion	Calvert Cliffs
1.	Power (Mwt)	2440	3250	2700
2.	NSSS configuration/supplier	3-loop W	4-loop W	2-loop CE (1)
3.	Containment free volume (ft^3)	1.8×10^6	2.8×10^6	2.0×10^6
4.	Reinforced or post-tensioned containment	Reinforced	Post-tensioned	Post-tensioned
5.	Containment operating pressure (psia)	10-11	Atm	Atm
6.	Containment design pressure (psia)	60	62	65
7.	Containment failure pressure (psia)	135	150	140
8.	In-core instrumentation penetration location	Bottom	Bottom	Top
9.	Cavity condition (given spray failure)	Dry	Flooded	Dry
10.	Concrete type	Basaltic	Limestone	Siliceous (2)
11.	Spray without emergency power	No	Yes	No
12.	Spray recirc. independent of ECCS recirc.	Yes	No	Yes
13.	ESF containment unit coolers/filters	No	Yes	Yes
14.	No. of high pressure valves in ECCS discharge	2 (3)	4	4 (4)
15.	ECCS low pressure line break location submerged	Yes	Not investigated	See Note 5
16.	RHR (or shutdown cooling) system inside containment	Yes	No	No
17.	No. of high pressure valves in RHR/SCS letdown path	N/A	2 (6)	2 (6)
18.	RHR/SCS letdown break location submerged	N/A	No	No
19.	Contiguous structure free volume (ft^3)	10,000 (7) Engineered Safeguards Building	1.4×10^6 Auxiliary Building	2.5×10^6 Auxiliary Building

(1) Two parallel cold legs per loop.
(2) $CaCO_3$ content comparable to basaltic.
(3) Including two check valves and MOV in cold leg flow path locked open.
(4) Including one normally open weighted check valve closing on 300 gpm reverse flow.
(5) High pressure pump suction submerged, low pressure pump discharge not submerged.
(6) Interlocked to preclude opening with high RCS pressure.
(7) BMI-2104 assumed approximately 200,000 ft^3 (apparently including quench spray pump house and main steam valve house).

analyses. The reactor power levels for Surry and Calvert Cliffs are similar, allowing for close correlation of analysis features which are strongly influenced by power level.

Features 3 through 7

The containment free volume is an important consideration in aerosol behavior in the containment. The Calvert Cliffs containment free volume is sufficiently similar to that of Surry to permit the application of the results of analyses of fission product retention in the Surry containment to Calvert Cliffs. Although Surry is a reinforced containment, whereas Zion and Calvert Cliffs are post-tensioned containments, the most important feature of the containment relative to source terms considerations is the containment failure pressure. All three plants have containment failure pressure well in excess of the loads calculated to be imposed on them in the initial phases of severe core damage accidents. Although the timing of postulated late overpressure failure of the containment would vary from plant to plant, the source terms are expected to be very low for such scenarios at all three plants. The pressures included in the attached tabulation for Surry and Zion are based on NUREG-0956. The Calvert Cliffs data were developed by BG&E.

Feature 8

The Calvert Cliffs plant includes in-core instrumentation penetrations in the top head of the reactor pressure vessel, whereas the Surry and Zion plants have lower head penetrations. However, the source term analysis for Surry and Zion reported in NUREG-0956 did not include the lower vessel head penetrations as sites for early head failure. Therefore, those analyses are directly comparable to the Calvert Cliffs continuous lower head configuration.

Features 9 and 10

A dry, or nearly dry, lower reactor cavity permits an earlier and more aggressive core/concrete interaction than does a flooded cavity. The overlying water in a flooded cavity also permits the removal of fission products released from the core/concrete interaction. The Surry analyses reported in NUREG-0956 are based on a dry cavity and are thus directly applicable to Calvert Cliffs. Of somewhat less importance is the concrete type. Limestone concrete produces considerably more carbon dioxide during the core/concrete interaction than does basaltic (or siliceous) concrete, leading to a somewhat earlier containment overpressure condition. The NUREG-0956 analysis for Surry was based on basaltic concrete. From a mass and energy release perspective, it should be reasonably comparable to the siliceous concrete at Calvert Cliffs.

Features 11 through 13

Containment failure time and fission product aerosol removal are greatly affected by the availability of containment sprays. The Surry and Calvert Cliffs systems are comparable in this regard. It should be emphasized, however, that the severe accident sequences which result in the highest source terms are those in which containment sprays are assumed to be unavailable. Calvert Cliffs is similar to Zion in relation to safety-related containment unit coolers/filters, which are not included in Surry. Analyses for Surry result in higher source terms than would be the case with assumed availability of these systems. Thus, application of the Surry results to Calvert Cliffs, which includes these additional sites for removal of fission product aerosols, is conservative.

Features 14 through 19

These features are included in the attached compilation because they relate to interfacing systems loss of coolant accidents (LOCAs), which historically have been referred to as V sequences since WASH-1400. In the WASH-1400 and NUREG-0956 analyses of Surry, the V sequence analyzed included the overpressure failure of the low pressure portion of the Emergency Core Cooling System (ECCS) located outside containment. Thus, the sequence represented a containment bypass accident. At Surry, the most probable location of the low pressure piping break would be submerged by RCS water and by water from the Refueling Water Storage Tank (RWST). A review of the Calvert Cliffs plant arrangement indicates that a similarly flooded break location is possible. Due to the plant unique valve arrangement at Surry (two check valves and a locked-open motor-operated valve (MOV)) a relatively high probability was assigned to ECCS system failure. Such a failure mode is much less probable for Calvert Cliffs, and if it were to occur it would occur in the large (2.5×10^6 ft^3) auxiliary building, possibly at a flooded break location.

The probability of a failure in the low pressure Shutdown Cooling System (SCS) letdown line at Calvert Cliffs is more likely than the ECCS low pressure injection line due to the presence of only two isolation valves in the SCS line, as opposed to four in the ECCS line. (At Surry, the Shutdown Cooling System is referred to as the RHR system and is located inside containment, thus a failure in that system does not constitute a containment bypass sequence). Moreover, the ECCS low pressure injection line failure for Surry reported in NUREG-0956 is not applicable to Calvert Cliffs because of the small contiguous structure used in the analysis and the probability of the failure itself (i.e., number of valves in the line). Thus, a plant-specific analysis of the interfacing system LOCA sequence is required for Calvert Cliffs because of the substantial difference in plant features from the Surry analysis reported in NUREG-0956.

Based on the above comparison of plant features it is concluded that:

- o Calvert Cliffs is sufficiently similar to Surry to allow the application of the results of Surry source term analyses to Calvert Cliffs for accident sequences involving releases of fission products into the containment building (i.e., so called "in-containment" sequences).
- o The interfacing system LOCA sequence (V sequence) analysis in NUREG-0956 for Surry is not applicable to Calvert Cliffs, and a plant-specific analysis is required.

3.2 Engineering Evaluation of Accident Sequences at CCNPP

3.2.1 Selection of Accident Sequences

In determining which potential severe accident sequences to focus on, the following factors were considered:

- o Contribution to overall core melt frequency as presented in the Calvert Cliffs Interim Reliability Evaluation Program (IREP) Report (NUREG/CR-3511).
- o Sequences which involve:
  - a spectrum of the possible severe core damage accidents (LOCAs, transients, etc.)

- a spectrum of thermal hydraulic and fission product transport phenomena in the reactor coolant system, containment and auxiliary building

- a pre-existing breach of containment (i.e., containment isolation failure)

- containment bypass (i.e., interfacing systems LOCA outside containment).

Based on these factors, the following severe accident sequences were selected for evaluation:

- ATWS - Anticipated Transient Without SCRAM. ATWS sequences contribute 33 percent to the total core melt frequency as reported in the Calvert Cliffs IREP study. All of these sequences result in transient induced LOCAs because of either a stuck open relief valve or a rupture of the reactor coolant pressure boundary. Containment sprays are available, thus resulting in low expected source terms for this group of sequences.

- TML - Transients with complete loss of feedwater. TML sequences contribute 32 percent of the total core melt frequency in the IREP study for Calvert Cliffs. Since the shutoff head of the high pressure ECCS injection pump is less than the relief valve setpoint, it is possible that all core makeup may fail if heat cannot be removed from the steam generators. Containment sprays are available, thus resulting in low expected source terms for this group of sequences.

- S_2H/S_2D'' - Small LOCAs with loss of ECCS. These two sequences include ECCS failure in either the injection (D") or recirculation (H) phase. Together these sequences account for 12% of the total core melt frequency in the Calvert Cliffs IREP study. In both cases, containment sprays are available.

- S_2FH - Small LOCA with failure of containment spray recirculation and high pressure ECCS recirculation. This sequence was a significant contributor to the total core melt frequency in the Calvert Cliffs IREP study (9 percent) and was also analyzed in the IDCOR program analysis of PWRs. Containment sprays do not function during the time period of postulated fission product release to the containment.

- Station blackout. As analyzed in the Calvert Cliffs IREP study, this sequence is postulated to result in the loss of the power conversion system (PCS) upon loss of all ac power. After approximately 4 hours, station dc power is assumed to be lost due to battery depletion, resulting in loss of the auxiliary feedwater system (AFW). It represents one of the important core melt sequences in IREP (3 percent) and is very similar to the TMLB' PWR accident analyzed in WASH-1400, BMI-2104, IDCOR, and other studies. The difference between the traditional TMLB' and the Calvert Cliffs Station Blackout sequence is the delayed failure (after about 4 hours) of the AFW system. Phenomenologically, it is the same as the WASH-1400 TMLB'. Containment sprays are not postulated function for these sequences.

- Interfacing system LOCA sequences. As is the case with all PWRs, the Calvert Cliffs plant contains several potential containment bypass flowpaths. The WASH-1400 V sequence was not considered in the Calvert

Cliffs IREP study because of the extremely low assessed probability of the necessary valve failures to expose the low pressure portions of the ECCS to full RCS pressure. The Calvert Cliffs ECCS has 3 check valves and one normally closed motor operated valve (MOV) between the reactor vessel and the low pressure piping on both the low pressure ECCS pump discharge and the high pressure ECCS pump suction in the auxiliary building. Based on the ECCS configuration, the V sequence was dismissed in IREP on probabilistic grounds (1×10^{-11} per reactor year). The most probable pipe break location in the low pressure portion of the high pressure ECCS would be submerged by break effluent; for the low pressure ECCS it would likely not be submerged.

A Shutdown Cooling System (SCS) letdown line "V sequence" has been included in this evaluation based upon examination of other containment bypass flowpaths with fewer isolation valves than the ECCS flowpaths and an assessment of potential pipe break locations in these flowpaths which would result in an unsubmerged break condition.

3.2.2 Summary of Engineering Evaluation of Selected Sequences

Engineering evaluations were performed for selected sequences to determine which sequences would be bounded by corresponding sequences for Surry, as reported in NUREG-0956, and which sequences would require additional or plant-specific treatment. As discussed below, the CCNPP sequences were found to be bounded by the corresponding Surry sequences in all cases except the containment bypass sequences.

ATWS

In the Calvert Cliffs IREP study, several different Anticipated Transient Without SCRAM (ATWS) sequences were postulated. They differ with respect to initiating event and subsequent system failures; however, all of them include a major transient followed by a failure to SCRAM.

Although the combined contribution of ATWS sequences to the total Calvert Cliffs IREP core melt frequency is 33 percent, these sequences do not involve high source terms due to the mitigating effects of engineered safety features.

Baltimore Gas and Electric (BG&E) has previously committed to implementing those portions of the ATWS rule applicable specifically to Combustion Engineering (CE) plants; i.e., installation of an independent trip system. Additionally, as a result of the Salem ATWS event, BG&E has developed procedures for ATWS events for the Calvert Cliffs Control room operators which call for:

1) Manual SCRAM.

2) De-energization of the motor-generator sets which supply control power to the reactor protection system SCRAM circuitry (which operates on a de-energize-to-SCRAM concept).

3) Manual initiation of emergency boron injection.

BG&E has also refurbished the SCRAM breakers at Calvert Cliffs and has provided improved testing and maintenance procedures. Therefore, the basic SCRAM system failure frequency in the Calvert Cliffs IREP study has been measurably reduced.

In all postulated ATWS sequences, the containment heat removal systems, Containment Spray System (CSS) and Containment Air Recirculation and Cooling System (CARCS), remain available. Although effectiveness of the CARCS may be somewhat reduced by the presence of aerosols in the containment atmosphere, the CSS and CARCS system provide redundant and independent means of containment cooling and fission product removal. Two dedicated 100 percent capacity containment spray pumps are installed, along with a nozzle arrangement which provides complete horizontal coverage of the containment cross-section.

Both the CSS and CARCS systems are safety-related with respect to power supplies, controls, etc. and have no unusual system dependencies or operating characteristics. Since the CARCS may not be completely effective, only the CSS is assumed to be available during an ATWS accident sequence in this evaluation. From a fission product release viewpoint, ATWS should be a benign event due to continual containment heat removal and fission product scrubbing.

TML

There are five significant sequences included under this heading, all of which involve a transient-induced loss of the PCS and subsequent loss of auxiliary feedwater. Since the high pressure ECCS pumps have a relatively low shutoff head of approximately 1275 psia, it is conservatively assumed that it may not be possible to guarantee cooling of the core by "feed and bleed" with the primary relief valves cycling following a loss of secondary heat removal. Therefore, it is postulated that ECCS makeup will fail. In fact, operator actions are possible that would force the relief valves to remain open, possibly allowing the reactor pressure to decrease below the shutoff head of the high pressure ECCS pumps. However, no credit was given for them in the IREP study. For all of the TML cases, CSS is assumed to be effective in substantially reducing source terms.

S_2H/S_2D''

In both of these sequences, a small LOCA (S_2) is postulated to occur, including reactor shutdown via SCRAM and initiation of AFW. In the case of S_2H, high pressure ECCS injection is successful as well, but in the case of S_2D'', it is not. Therefore, in the case of S_2D'', loss of reactor water inventory is assumed to be immediate. In the case of S_2H, the ECCS fails when the RWST depletes and the switchover to recirculation occurs. This switchover is expected to occur between 4 to 12 hours after the start of the LOCA depending on the size of the break. In either case, due to the lack of makeup to the reactor vessel, core uncovery and fuel damage are postulated to occur. Because the CSS is assumed to be effective, both of these cases are relatively benign from the standpoint of fission product releases.

S_2FH

In this sequence a small LOCA (S_2) is postulated to occur which is followed by successful reactor shutdown via SCRAM and initiation of AFW and high pressure ECCS. This provides for decay heat removal via the steam generators and maintenance of reactor vessel water inventory. When the RWST depletes and switchover to recirculation occurs, the containment spray recirculation mode and high pressure ECCS recirculation mode are assumed to fail (F and H respectively). This switchover is expected to occur between 4 to 12 hours after the start of the LOCA depending on the size of the break. Due to the lack of makeup to the reactor vessel, core uncovery and fuel damage are postulated to occur.

Since the CSS is assumed to be inoperable due to failure of the recirculation portion of the system, the S_2FH sequence is not considered benign with respect to fission product releases. The CARCS is available to remove heat and fission products from the containment atmosphere, but because the effectiveness of the CARCS may be somewhat reduced by aerosol loading, no credit is taken for its operation.

Station Blackout

In this sequence, a loss of offsite power initiating event is postulated to occur, followed by a SCRAM and subsequent loss of onsite emergency power. The power conversion system would not be available without offsite ac power. The AFW system continues to provide makeup to the steam generators and decay heat is removed from the RCS via the steam generator relief valves. At some time subsequent to 4 hours into the sequence, station dc power is postulated to be lost due to battery depletion and the AFW system fails on loss of dc control power. Core boiloff, uncovery, and subsequent fuel damage are postulated to occur assuming that neither onsite nor offsite ac power is restored. Because ac power is assumed unavailable, no containment cooling systems (CSS, CARCS) are operable during this sequence. This accident is very similar to the definition of TMLB' found in WASH-1400 and other severe accident risk and source term studies. The difference in this case is the delayed failure of AFW by at least 4 hours, which takes credit for the availability of dc power for a given period following loss of all ac power.

Without ac power, and assuming eventual depletion of dc power, a core degradation and melt progression are postulated to occur. However, several important factors affecting accident probability and timing deserve consideration. The assumption that the station batteries will provide dc power for at least 4 hours is derived from the fact that dc loads under station blackout conditions are substantially smaller than under design basis LOCA conditions. The design basis capacity for the dc power supply is 2 hours. In addition, a fully qualified reserve battery with an additional 2 hour capacity is also available under station blackout conditions. The IREP study assumed 4 hours and actual test discharges for the Calvert Cliffs battery system have indicated that the battery will operate successfully for several hours following loss of all ac power. The length of time that dc power is available is important because it offers more time to restore offsite power and/or emergency diesel power. Additionally, the Calvert Cliffs plant has a separate NRC-approved source of offsite power which was connected during plant construction and is still used to provide hotel services to administrative areas. This power originates with the Southern Maryland Electric Co-Operative (SMECO). Although not safety-related in terms of distribution or control, it is available and is in addition to the normal offsite power supply. Use of the SMECO power supply could provide a means of recharging the station batteries. Extension of dc power operation would provide extra time to restore ac power and would extend the time of core uncovery if ac power could not be restored. Taking credit for the SMECO power supply would reduce the calculated probability of station blackout for Calvert Cliffs below that used in IREP.

V (Interfacing System LOCA Bypassing Containment)

In this sequence, a pipe or component directly connected to the reactor coolant pressure boundary (or part of that boundary) is assumed to break outside containment. Because of the isolation capability provided for all piping systems penetrating containment, these sequences require multiple valve failures to occur. System designs vary from plant to plant, necessitating attention to specific plant features which might result in an interfacing systems LOCA bypassing containment.
The following table presents the results of an engineering survey carried out by Stone and Webster Engineering Corporation and the Calvert Cliffs plant staff. Several potential V

sequence flowpaths were identified and evaluated, taking into consideration piping size and configuration, the location of the piping code class changes (high pressure to low pressure), the number of valve failures required to reach the code class change, the room elevation of the possible break, and other pertinent features. In addition, the potential for submerged break locations within flood protected compartments was evaluated.

System/ Flowpath	Pipe Size	No. of Valve Failures	Elevation of Break Within Room
HPSI	6 in.	4 (Notes 1&2)	Low
LPSI	6 in.	4 (Note 2)	High
SCS Letdown	14 in.	2	High
CVCS Letdown	2 in.	(Note 3)	High
CVCS Charging	2 in.	3 (Note 4)	Low

Note 1: HPSI pump discharge piping is code service class CC (2485 psig rating) or class DC (1600 psig rating).

Note 2: Includes one check valve which is normally open but which will close if reverse flow exceeds 300 gpm.

Note 3: CVCS letdown flow must pass through 2 air-operated valves.

Note 4: CVCS charging pump discharge lines are code service class CC (2485 psig rating). The 3 valves are all check valves.

Based upon the results of the above survey, the interfacing system LOCA considered most limiting is a break in the SCS letdown line resulting from the failure of 2 high pressure containment isolation valves. This scenario also provides the largest possible leakage path (i.e., 14 in. diameter pipe) and an unsubmerged break location.

It should be noted, however, that the two valves in the SCS letdown flowpath which must fail in order to expose the low pressure portion of the system to the RCS pressure are both flex-wedge MOVs. A survey of nuclear power plant operating histories and discussions with several large valve manufacturers have not revealed a single incident of large leakages past the disk of this type of valve, let alone catastrophic disk failure. For the purposes of this study, however, a conservative disk failure rate characteristic of check valves has been used to assign a probability to this event.

The potential also exists for a failure in the low pressure portion of the SCS at a pump seal, which would result in a submerged break in the flood protected pump compartment. Therefore, this sequence was also chosen for analysis to provide a range of V-sequence outcomes. The pump seal leak was designated as the low range V sequence, and the 14" line break was designated as the high range V sequence based on the relative sizes of the expected source terms.

Because the flowpaths involve LOCAs in the auxiliary building, the containment and its heat removal systems are directly bypassed. Attention is thus focused on fission product transport and retention in the auxiliary building. This building is large (2.5×10^6 ft^3) and consists of several elevations and a large number of compartments. It also contains a seismically supported fire protection sprinkler system. The analysis of the source terms for these sequences is summarized in Section 3.4.

3.3. Summary of Results

As a result of the above evaluations, the following sequences were included in the CCNPP source term analyses:

Sequence	Description	Origin of Source Term Used
TMLB'-δe	Station blackout with early overpressure break of the containment	NUREG-0956
TMLB'-β	Station blackout with failure to isolate the containment	NUREG-0956
S_2D-γ	Small break LOCA with containment break due to H_2 burn	NUREG-0956
V	Interfacing system LOCA bypassing containment	CCNPP Plant-Specific

The probability of occurrence of the severe accident sequences included in the CCNPP offsite dose analyses is summarized in Table 2-6. As noted in this table, the CCNPP probabilities were derived from a combination of data from the IREP study for CCNPP and containment event tree probabilities reported in NUREG-0956. The V sequence probability is based on the probability of an RHR - V sequence as reported in the Zion PRA. Table 2-7 lists the severe accident source terms utilized in the analysis of offsite doses at CCNPP.

The first three listed "in-containment" sequence source terms were taken from the NUREG-0956 analysis for Surry, based on the determination that these data are directly applicable to CCNPP. The two V sequence source terms are Calvert Cliffs plant specific. The range of the V sequences possible at CCNPP was represented by "low range" and "high range" source terms. The low range source term comprises the complete release of the noble gas core inventory (xenon and krypton), with negligible release of the volatile and non-volatile fission product groups. This source term is judged to be applicable to the low-range V sequence, involving a SCS pump seal rupture resulting in a small size leak into a flood-protected pump compartment (i.e., submerged break). The high range source term was calculated using a detailed analysis of a 14 in. diameter SCS pipe break (unsubmerged).

3.4 Description of Plant-Specific Containment Bypass Sequence Analysis

The auxiliary building at CCNPP has an extensive fire suppression system, including sprinklers on all of the levels in the potential accident discharge flowpaths to the environment. These sprinklers are individually set to discharge water when a local temperature of 212°F is reached. There are approximately 300 sprinklers on Elevation 5', which is where the 14" SCS pipe break is postulated to occur. The activation of these sprinklers markedly affects the subsequent thermal hydraulics in the auxiliary building. Initially, 63 of the sprinklers are calculated to actuate based on the temperature transient associated with the blowdown. During the course of the accident, a total of 140 sprinklers were calculated to be actuated. A minimum water supply of 600,000 gallons is available to the fire suppression system. Based on a design flow rate of 20 gpm per sprinkler, the water supply was calculated to be available until after the complete release of fission products at the break location.

TABLE 2-6

PROBABILITY OF OCCURRENCE OF SEVERE ACCIDENT SEQUENCES AT CCNPP
(Per Reactor Year)

Sequence	Mode of Release from Containment	CCNPP Probability of Event	Cont. Event Tree Probability	Probability of Release
TMLB'-δe	Early Overpressure	1.5×10^{-5}(a)	5×10^{-3}(c)	7.5×10^{-8}
TMLB'-β	Isolation Failure	1.5×10^{-5}(a)	5×10^{-3}(d)	7.5×10^{-8}
S_2D-γ	Early Overpressure	1.0×10^{-4}(b)	1×10^{-3}(c)	1.0×10^{-7}
V	Bypass Via Interfacing Syst. LOCA	1×10^{-7}	N/A	1×10^{-7}

(a) Based on combining the 4.4×10^{-6} probability of a TMB' sequence with the 1.1×10^{-5} probability of an S_2FH sequence from the IREP Study for CCNPP.

(b) Based on combining the probabilities of all core melt sequences with containment sprays operating (ATWS, transients with loss of FW and S_2D''/S_2H) for the IREP Study for CCNPP.

(c) Based on NUREG-0956.

(d) Based on EPRI/ NSAC analysis for Oconee (NSAC-60 Oconee PRA, June 1984).

TABLE 2-7

SEVERE ACCIDENT SOURCE TERMS AND PROBABILITY OF OCCURRENCE FOR CCNPP

				Range of V Sequences	
Accident Sequence	TMLB'-δe[a]	TMLB'-β[a]	S_2D-γ[a]	Low[b]	High[c]
Probability, Per Reactor Year (See TABLE 2-6)	7.5×10^{-8}	7.5×10^{-8}	1×10^{-7}	1×10^{-7}[d]	1×10^{-7}[d]
Time of Rel, hr.	2.5	2	2.5	>10[e]	4
Duration of Rel, hr.	10[f]	10[f]	3	1	3
Fission Product Group	Fraction of Core Inventory Released to Environment				
Xe-Kr	0.85	1.00	0.50	1.00	1.00
I-Br	0.07	0.022	0.005	<0.001	0.039
Cs-Rb	0.058	0.013	0.0001	<0.001	0.038
Te-Sb	0.055	0.11	0.01	<0.001	0.042
Ba-Sr	0.01	0.058	0.03	<0.0001	0.0009
Ru	0.0013	0.053	0.001	<0.0001	<0.0001
La	0.0017	0.0002	0.0009	<0.0001	<0.0001

(a) Release fractions from NUREG-0956.
(b) Low range of potential V sequences includes small leakage paths (e.g., 0.1 ft^2) at a pump seal submerged in a flood-protected pump compartment.
(c) High range of potential V sequences is based on a 14 in. diameter SCS pipe opening - unsubmerged.
(d) 1×10^{-7} is the total probability of the range of V sequences for SCS. The probability of experiencing the high range source term would be substantially lower than 1×10^{-7}.
(e) Time of release is a function of postulated size of opening in SCS.
(f) Analyzed in CRAC2 with a 2 hr. release duration.

The fission product aerosol spray removal resulting from the operation of the sprinklers was included in the analysis of the source term for the postulated unsubmerged 14" SCS pipe break. The analysis of aerosol spray removal required some adjunct calculations in conjunction with the NAUA - 4 computer program.[14]

The method of analysis of the unsubmerged SCS pipe break V sequence is summarized below:

The MAAP-2.0B computer program[15] was utilized to analyze the accident progression within the RCS and the containment building. The results of this analysis provided the event timing and the mass and energy releases at the postulated SCS pipe break location.

The thermal hydraulics portion of the analysis of the accident progression in the CCNPP auxiliary building utilized the THREED-ST computer program.[16] THREED-ST was previously utilized in the performance of extensive thermal hydraulic analysis of the Surry plant, as reported in Chapter 6 and Appendix B of the report of the ANS Special Committee on Source Terms. Additional engineering calculations were performed, as adjuncts to the THREED-ST analyses, in order to address such issues as determinat on of the effects of the size distribution of water droplets from the fire suppression sprinkler system.

Based on the results of the THREED-ST analysis of the high energy line break in the SCS in the auxiliary building, it was assumed that the roof vents would open within a few seconds of accident initiation due to the rapid pressure rise. The areas of the resultant roof-level openings to the environment are 30 ft^2 and 177 ft^2. In addition to the roof vents, a 50 ft^2 opening was assumed to occur, at approximately the same time, in the metal truck door leading into the refueling bay. These openings not only provide leakage pathways to the environment, but they also affect the thermal hydraulics aspects of the accident sequence by creating circulation paths within the auxiliary building.

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14 The Stone and Webster Engineering Corporation's version of the NAUA-4 computer program was utilized in the analysis of aerosol behavior and release from the auxiliary building. The analysis is similar to the Surry analyses reported in the ANS report.

15 "Modular Accident Analysis Program - MAAP-2.0B", Industry Degraded Core Rulemaking (IDCOR) Program, (In Publication by Atomic Industrial Forum.)

16 "A Subcompartment Transient Response Code (THREED-ST)", Stone and Webster Engineering Corporation Computer Program NU-186 (Unpublished).

The distribution of volatile fission product species in the two CCNPP V sequence source terms is as follows:

V-High Range Source Term

Fraction of Core Inventory

Species	RCS & SCS	Auxiliary Building	Environment
CsI	0.167	0.794	0.039
CsOH	0.206	0.756	0.038
Te	0.116	0.842	0.042

V-Low Range Source Term

Fraction of Core Inventory

Species	RCS & SCS	Water in Pump Comp.	Auxiliary Building	Environment
CsI	0.167	0.815	0.017	less than 0.001
CsOH	0.206	0.777	0.016	less than 0.001
Te	0.116	0.865	0.018	less than 0.001

The release of non-volatile fission product species in the V sequence is expected to be extremely low. Reactor vessel meltthrough would occur with the RCS depressurized, resulting in a low pressure release into the containment (which would be at approximately atmospheric pressure). The flowpath for the fission products released from the core/concrete interaction would then be back through the opening in the lower vessel head, through the vessel and upper vessel internals, through the RCS piping to the SCS piping, allowing substantial opportunity for retention en route to the break location with subsequent additional retention in the auxiliary building.

The in-containment sequence source terms, obtained from NUREG-0956, and the CCNPP plant-specific V sequence source terms, as reported in Table 2-7, served as inputs to the analysis of offsite doses utilizing the CRAC2 computer program[17] with CCNPP site-specific meteorological data. The results of these analyses are presented in Attachment 1.

4.0 Summary

In summary, the new material circumstances, as considered in this application for exemption, include:

(1) replacement of the Reactor Safety Study (WASH-1400) results and methods of

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17 Ritchie, L. T., et. al., "CRAC2: Calculations of Reactor Accident Consequences, Version 2', NUREG/CR-2336, Sandia National Laboratories, February 1983.

analysis with results and methods of analyses representing a major advance in the technology of analysis of severe accident source terms, as reported in NUREG-0956;

(2) development of CCNPP plant-specific source terms for the containment bypass (interfacing systems LOCA) "Event V" sequences;

(3) use of CCNPP site-specific meteorology for offsite dose calculations;

(4) recognition of the inherent strength of the containment building; and

(5) recognition of the inherent capability of plant systems and structures for the retention and removal of fission product species.