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EARLY SITE PERMIT
DEMONSTRATION PROGRAM
Regulatory Criteria Evaluation Report

SNL/NM-- 699795-3 App. E

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Appendix E

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Prepared by the *Joint Contractors* (Southern Electric International, Commonwealth Research Corporation, and Public Service Corporation of New Jersey), in cooperation with the Electric Power Research Institute, the Nuclear Management and Resources Council, the U.S. Department of Energy and Sandia National Laboratories.

N P O C

BUILDING
NEW
NUCLEAR
POWER
PLANTS

POPULATION DENSITY REPORT

MASTER

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**EVALUATION OF POPULATION DISTRIBUTION
RELATIVE TO MEETING THE QUANTITATIVE
HEALTH OBJECTIVES OF THE NRC SAFETY GOAL
POLICY FOR OFFSITE RISK ASSOCIATED WITH
NUCLEAR POWER PLANTS**

EARLY SITE PERMIT DEMONSTRATION PROGRAM

MARCH 1993

March 1993

Executive Summary

In this study the proposed U.S. Nuclear Regulatory Commission (NRC) population density and exclusion zone size criteria were evaluated for their potential effectiveness to meet the quantitative public health and safety objectives associated with the NRC Safety Goal Policy. In addition, the individual and societal risks from postulated accidents (based on NUREG-1150 methods) were determined and compared to the quantitative prompt and latent health objectives of the safety goal policy.

The proposed revisions to 10 CFR 100 regarding population density and exclusion zone size criteria were evaluated using the MELCOR Accident Consequence Code System (MACCS). Surry Nuclear Plant radiological and meteorological input data were taken from the NUREG-1150 analysis. Surry was selected for the following reasons: it is one of the five plants evaluated by the NRC in NUREG-1150; it is one of two plants for which NUREG-1150 considered analysis of external events; and its source term data are readily available.

This analysis demonstrates that a nuclear power plant of current design located in an area of controlled population distribution (500 people per square mile) and minimum exclusion zone size (0.33 miles), as specified by the NRC in the proposed 10 CFR 100 revision, results in a calculated societal radiation dose risk from reactor accidents of approximately 13 person-rem per reactor-year. The calculated individual radiological risks from postulated accidents demonstrate that the prompt and latent Quantitative Health Objectives (QHOs) of the NRC Safety Goal Policy are met by a wide margin.

NUREG-1150 used site-related factors including plant design features (source term), meteorology, population density and distribution, and estimates of the effectiveness of emergency plans. Therefore, by using the NUREG-1150 study methodology, which fully considers population density and distribution, it is possible to compare offsite health effects to the QHOs of the NRC Safety Goal Policy.

Further, the population density criterion proposed for codification will not provide the flexibility needed to select consistently the most favorable sites. Alternative population configurations with larger overall populations may result in lower societal radiation risk yet fail the NRC proposed criterion. This may result in eliminating acceptable (possibly the more favorable) sites from further consideration while less favorable sites survive the site selection screening process. The proposed criterion could unnecessarily constrain future activities at existing sites (if they were to be evaluated under the proposed rule), sites that have previously been approved but have never had operating reactors, and new sites that would likely be considered for future reactors. For example, in the NRC Staff briefing to the Commission, the staff stated that eight licensed sites do not currently meet the five hundred persons per square mile population density criterion, and twenty-five licensed sites do not meet the 0.4 mile exclusion area distance criterion. All of these sites have been demonstrated to adequately protect the public health and safety and easily meet the NRC Safety Goal Policy.

If codified, then the restrictive numeric criteria would establish a de facto safety goal in regulation that is more stringent than the NRC Safety Goal Policy and is applicable to future plants only. The next generation of nuclear plants will have comprehensively considered 30 years of accumulated experience including improved understanding of postulated severe accident phenomena, probability, and consequences. Therefore, these plants will reflect in their design, construction, and operation risk characteristics as low or lower than those for existing plants.

March 1993

The proposed criteria would be significantly more restrictive than what is required to meet the QHOs of the Safety Goal Policy and would therefore be unnecessarily restrictive. Since the risk to the public would be so much less than the 0.1 % increase from all other factors established by the QHOs of the Safety Goal Policy, there is insufficient health risk basis for establishing such a restrictive criteria.

March 1993

EVALUATION OF THE PROPOSED NRC POPULATION DENSITY CRITERION ON SEVERE ACCIDENT RISK

I. Summary

10 CFR Part 100, *Reactor Site Criteria*, sets forth the criteria for an exclusion area, a low population zone, and a population center distance around nuclear plants. The U.S. Nuclear Regulatory Commission (NRC) has established population density guidelines in Regulatory Guide 4.7 to be used as an indicator as to whether a facility meets the Federal requirements. The NRC has issued for public comment the proposed change to 10 CFR Part 100 which, if promulgated, would codify these population density and exclusion zone size guidelines. This study was undertaken to evaluate (1) the proposed population and exclusion zone size criteria for their effectiveness in meeting the public health and safety objectives associated with the prompt and latent quantitative health objectives (QHOs) of the NRC Safety Goal Policy and (2) the individual and societal health and safety risks for several population distributions using the methodology used by the NRC in NUREG-1150, *Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants*.

The results of this study indicate that population density and exclusion zone size criteria proposed in 10 CFR Part 100 do not influence conformance to the QHOs of the safety goal policy. Further, the results of this study have shown that a reactor of current design located in an area of controlled population distribution and minimum exclusion zone size as specified by the NRC in the proposed 10 CFR Part 100 revision results in a calculated societal risk of approximately 13 person-rem per reactor-year. The study also demonstrates that such a facility meets the individual QHOs by a wide margin.

NUREG-1150 used site-related factors including plant design features (source term), meteorology, population density and distribution, and estimates of the effectiveness of emergency plans. Therefore, by using the NUREG-1150 study methodology, which fully considers population density and distribution, it is possible to compare offsite health effects to the QHOs of the NRC Safety Goal Policy. However, because population density guidelines would only limit the number of potentially affected individuals in the vicinity of the plant, it does not address the benefits of other influences pertinent to protection of public health and safety.

Thus, the population density criterion proposed for codification will not provide the flexibility needed to select consistently the most favorable sites. Alternative population configurations with larger overall populations may result in lower societal radiation risk yet fail the NRC proposed criterion. This may result in eliminating acceptable (possibly the more favorable) sites from further consideration while less favorable sites survive the site selection screening process. The proposed criterion could unnecessarily constrain future activities at existing sites (if they were to be evaluated under the proposed rule), sites that have previously been approved but have never had operating reactors, and new sites that would likely be considered for future reactors. For example, in the NRC Staff briefing to the Commission, the staff stated that eight licensed sites do not currently meet the five hundred persons per square mile population density criterion, and twenty-five licensed sites do not meet the 0.4 mile exclusion area distance criterion. All of these sites have been demonstrated to adequately protect the public health and safety and easily meet the NRC Safety Goal Policy.

If codified, then the restrictive numeric criteria would establish a de facto safety goal in regulation that is more stringent than the NRC Safety Goal Policy and is applicable to future plants only. The next generation of nuclear plants will have comprehensively considered 30 years of accumulated experience including improved understanding of postulated severe accident phenomena, probability, and consequences. Therefore, these plants will reflect in their design, construction, and operation; risk characteristics as low or lower than those for existing plants.

The proposed criteria would be significantly more restrictive than what is required to meet the QHOs of the Safety Goal Policy and would therefore be unnecessarily restrictive. Since the risk to the public would be so much less than the 0.1 % increase from all other factors established by the QHOs of the Safety Goal Policy, there is insufficient health risk basis for establishing such a restrictive criteria.

II. Objective

To evaluate population distributions adjacent to a nuclear power plant and exclusion zone size relative to offsite risks from postulated accidents and to compare these risks to the QHOs of the NRC Safety Goal Policy using the NUREG-1150 methodology.

III. Background

Existing NRC policy establishes the safety goals for individual risks in the vicinity of nuclear power plants. These goals define what the NRC considers as adequate safety and risk beyond which the risk is considered to be remote and speculative for new rulemaking. These safety goals state that the risk of prompt or latent (cancer) fatalities to an average individual in the vicinity of a nuclear power plant should not exceed 0.1% of the risk from other causes. For the prompt risk safety goal, the vicinity is defined as within one mile of the reactor, whereas for the latent risk safety goal the vicinity is defined as within a ten mile radius of the reactor. The NRC has defined QHOs corresponding to the above goals as equivalent to 5×10^{-7} and 2×10^{-6} fatalities per reactor-year for prompt and latent cancer fatalities, respectively.¹

10 CFR Part 100 specifies the following:

- An exclusion area
- A low population zone (LPZ)
- A population center distance criteria

The exclusion area is to be of sufficient size such that any individual in this area (but outside the plant protected area) would not receive, upon two hours of exposure following a postulated release, a total radiation dose to the whole body in excess of 25 rem or a total radiation dose to the thyroid in excess of 300 rem from iodine exposure.

The LPZ is defined such that any individual in the LPZ who is exposed to the entire passage of the radioactive cloud resulting from a postulated release would not receive a dosage in excess of the above limits.

The population center distance criteria states that no population center (defined as a densely populated area with a total population in excess of 25,000 people) shall be closer than 1 1/3 times the distance from the reactor to the outer boundary of the LPZ.

The intent of the LPZ is to ensure that the population is sufficiently limited and, furthermore, is distributed in such a fashion that mitigating action (such as an evacuation) would likely be successful. The exclusion area was established based on the premise that successful mitigation might not be achievable given excessive radiation. Finally, the population center distance criterion addresses societal risk (albeit indirectly) in that it has been deemed desirable to keep large populations relatively far from the reactor, thereby minimizing the potential for large scale consequences.

¹U.S. Nuclear Regulatory Commission, SECY-89-102, *Implementation of the Safety Goal Policy*, March 30, 1989, and Staff Requirements Memorandum to SECY-89-102, June 15, 1989.

The distances and areas specified above cannot be precisely determined due to variability of release patterns, meteorology, and location of the most exposed individual and are specific to each operating nuclear plant site. The NRC established population density guidelines in Regulatory Guide 4.7 issued in November, 1975. The NRC determined that an exclusion area radius of 0.4 miles would be acceptable for all reactors when the Regulatory Guide was published. In addition, the NRC determined that an acceptable surrounding population density should not exceed 500 people per square mile averaged over any radial distance out to 30 miles. Furthermore, the projected population for the operating lifetime of the reactor (40 years) should not exceed 1000 people per square mile. The probable intent of the above population density requirements was to ensure a sufficiently low population density to allow for effective mitigation measures, while assuring that population centers would not encroach on the reactor.²

The NRC is now proposing to codify the above guidelines. The stated objective for this change is to, "Relocate the requirements that apply to plant design into 10 CFR part 50 thereby effectively decoupling siting from plant design."³ Another projected benefit is that by establishing quantitative population rules, the siting requirements become clearly defined. This, in turn, is expected to reduce the time and costs associated with obtaining site approval.

IV. Methodology

The population density and exclusion zone size criteria in the proposed revisions to 10 CFR Part 100 were evaluated using the MELCOR Accident Consequence Code System (MACCS), developed by Sandia National Laboratories for the NRC. Surry Nuclear Plant radiological and meteorological input data were taken from the NUREG-1150 analysis. Surry was selected for the following reasons: it is one of the five plants evaluated by the NRC in NUREG-1150; it is one of two plants for which NUREG-1150 considered analysis of external events; and its source term data are readily available. Additional discussion of the data and assumptions utilized in the MACCS analysis is provided in Appendix A.

V. Results

This section presents the results of the MACCS analysis of the four release scenarios discussed in Appendix A applied to a uniform population density of 500 people per square mile and an exclusion zone size of 0.33 miles. While MACCS can be used to calculate and display a variety of quantitative information, individual risk probabilities (identical to NRC QHO units) and the population dose (assumed to be a measure of cumulative societal risk) were selected for comparison.

For the NUREG 1150 analysis, the Surry source terms were divided into 17 groups and further divided into three subgroups each for a total of 51 subgroups. For purposes of this analysis, these groups were condensed into four categories, and representative source terms were chosen for each category. Appendix A provides further detail on the assumptions used for this analysis.

Figures 1 and 2 present comparisons of the quantitative prompt and latent individual risks from this analysis with their respective QHOs. As can be seen in Figure 1, the prompt risk is well

²The draft document for Regulatory Guide 4.7 issued the previous year made no mention of these population density guidelines. From review of available public documents, it was determined that (1) if these guidelines, which were later added to Regulatory Guide 4.7, had been added without allowing for public discussion and (2) the justification for the selection of the 500/1000 population density criterion (not given in Regulatory Guide 4.7) could not be found.

³U. S. Nuclear Regulatory Commission, October 20, 1992 (57 FR 47803).

below the QHO, reaching its peak value of 8.4% of the QHO at the boundary of the exclusion area (assumed for this study to be 0.33 miles). The prompt risk is shown to drop rapidly, and the residual radiation risk is insignificant beyond five miles. The individual latent risk, shown in Figure 2, meets the QHO by even a wider margin with a maximum individual risk of about 1% of the QHO. As with the prompt risk, the individual latent risk initially declines rapidly with distance with the risk reduced to about 10% of its maximum value at ten miles out. The risk then continues to decrease with radial distance.

Using a weighted average of all four modeled scenarios, Figure 3 depicts a plot of the resulting dose as it varies with radial distance. This dose is comprised of two components: dose from immediate impact (e.g., cloudshine, groundshine, inhalation, and deposition on skin), and dose from long term impact (e.g., consumption of contaminated milk, water, or food, as well as, dose from decontamination related activities). As with the individual risk, the dose declines rapidly with distance and is reduced to less than 10% of its maximum value at ten miles.

Figure 4 shows the immediate, long term, and combined population dose (individual dose weighted with population) as a function of distance. The combined population dose is shown to decline only slightly with distance, indicating that the decrease in individual dose is balanced by the increase in population. Note, however, that the immediate population dose decreases with distance, sharply at first, then declining more gradually. In contrast, the long term cumulative population dose increases with distance for approximately the first 15 miles then becomes stable with distance.

The total societal risk to a population (i.e., the cumulative risk to all people in the 30-mile radius) using the algorithms in MACCS is determined by integrating the area under the combined population curve in Figure 4. This assumes that latent risk is proportional to cumulative dose regardless of how the dose is distributed to individuals (e.g., one person-rem divided among 1000 people is equivalent to one person-rem to a single person). Using the NUREG-1150 data for the Surry Plant and assuming a uniform population density of 500 people per square mile and exclusion zone size of 0.33 miles, the total dose is calculated to be about 13 person-rem per reactor-year. This, of course, is the value of an average exposure taken over millions of reactor-years as the population would only be exposed in the extremely unlikely chance of a core damage event. This value represents a valid quantitative measure of the societal risk.

Figure 5 presents the results of an analysis performed using two assumed population distributions. The first assumes a uniform population density of 500 people per square mile. The second assumes that most of the population is located between the 20 and 30 mile radial distances. This analysis shows a scenario in which the NRC population density criteria is exceeded, yet the total societal risk is significantly lower than a scenario which conforms to the population density criteria.

Figures 6 through 8 present the results of an analysis performed assuming a reduced exclusion zone of 0.2 miles from the reactor with and without evacuation. This analysis shows that, assuming evacuation, the individual risk for the 0.2 mile exclusion zone is slightly higher than the individual risk for the 0.33 mile exclusion zone.

The individual risk to the nearby population is greater if there is no evacuation, as noted in Figures 6 and 7, and reflected in the population dose profile (Figure 8). The evacuation was assumed to be carried out to ten miles only (the same assumption as NUREG-1150); therefore, the risk to the greater than ten mile population is identical for both evacuation assumptions.

VI. Conclusions

As demonstrated by the results presented above, it can be concluded that a reactor of current design located in an area of controlled population distribution and minimum exclusion zone size, as specified by the NRC in the proposed 10 CFR Part 100 revision, results in a calculated annual societal risk which is well within the QHO of the NRC Safety Goal Policy. Considering that such a facility meets the individual QHOs by a wide margin, the proposed NRC population density and exclusion zone size criteria are unnecessarily restrictive and therefore, should not be promulgated. In addition, codification of these criteria is questionable because new future advanced nuclear plant designs have added features to lower the risk when compared to the current generation of plants.

Further, the population density criterion proposed for codification will not provide the flexibility needed to select consistently the most favorable sites. Alternative population configurations with larger overall populations may result in lower societal radiation risk yet fail the NRC proposed criterion. This may result in eliminating acceptable (possibly the more favorable) sites from further consideration while less favorable sites survive the site selection screening process. The proposed criterion could unnecessarily constrain future activities at existing sites (if they were to be evaluated under the proposed rule), sites that have previously been approved but have never had operating reactors, and new sites that would likely be considered for future reactors. For example, in the NRC Staff briefing to the Commission, the staff stated that eight licensed sites do not currently meet the five hundred persons per square mile population density criterion, and twenty-five licensed sites do not meet the 0.4 mile exclusion area distance criterion. All of these sites have been demonstrated to adequately protect the public health and safety and easily meet the NRC Safety Goal Policy.

If codified, then the restrictive numeric criteria would establish a de facto safety goal in regulation that is more stringent than the NRC Safety Goal Policy and is applicable to future plants only. The next generation of nuclear plants will have comprehensively considered 30 years of accumulated experience including improved understanding of postulated severe accident phenomena, probability, and consequences. Therefore, these plants will reflect in their design, construction, and operation; risk characteristics as low or lower than those for existing plants. The proposed criteria would be significantly more restrictive than what is required to meet the QHOs of the Safety Goal Policy and would therefore be unnecessarily restrictive. Since the risk to the public would be so much less than the 0.1 % increase from all other factors established by the QHOs of the Safety Goal Policy, there is insufficient health risk basis for establishing such a restrictive criteria.

FIGURE 1 PROMPT INDIVIDUAL RISK

FIGURE 2 LATENT INDIVIDUAL RISK

FIGURE 3 INDIVIDUAL DOSE PROFILE

The modeled core damage accident is an amalgamation of a range of release scenarios.

FIGURE 4 POPULATION DOSE PROFILE

Cumulative population Dose Risk (within 30 miles) equivalent to approximately 13 man-rem per reactor year.

FIGURE 5 POPULATION DOSE PROFILE FOR VARYING POP. PROFILE

- - - - Stratified Pop: 50 people/sq mile 0.3 - 10 miles;
 250 people/sq mile 10 - 20 miles; 750
 people/sq mile 20 - 30 miles

——— Uniform Pop: 500 people/sq mile 0.3 - 30 miles.

Total exposure:

Uniform Population: 13 man-rem/reactor-year

Stratified Population: 8.5 man-rem/reactor-year

FIGURE 6 PROMPT INDIVIDUAL RISK

FIGURE 7 LATENT INDIVIDUAL RISK

FIGURE 8 POPULATION DOSE PROFILE

Total dose with evacuation = 13.3 man-rem/year (for both 0.2 and 0.33 mile exclusion zones).
 Total dose without evacuation = 14.5 man-rem/year (0.2 mile exclusion zone).

APPENDIX A

Data and Assumptions for MACCS Analysis

In the NUREG-1150 analysis of the Surry Nuclear Plant, the thousands of source terms were partitioned into 17 groups. Each of these groups were subdivided into three subgroups:

Subgroup 1: Evacuation starts at least 30 minutes before the release begins.

Subgroup 2: Evacuation starts between 30 minutes before and 1 hour after the release begins.

Subgroup 3: Evacuation starts more than 1 hour after the release begins.

While this gives a potential for 51 subgroups, some of these subgroups have zero or nearly zero conditional probability. Nevertheless, this leaves 34 non-trivial subgroups for MACCS input. Rather than evaluating all of these groups, they were combined to a more manageable number of events while retaining the integrity of the analysis. Figure A-1 depicts the partitioning of the 17 groups by early and latent health effects as evaluated in the NUREG-1150 study. These groups were then further organized into four categories as shown. NUREG-1150 lists the expected annual frequencies of each of the 17 source terms, the conditional probability of each of their subgroups, and the predicted consequences of each of their occurrence. Table A-1 presents this data condensed into the four categories used for this study, along with the source terms selected to represent each of these categories. With the exception of source term 17-1, each of the selected postulated releases closely matches the expected impact of their respective categories. Source term 17-1 was conservatively selected to represent its category as it was the only source term in this group with a non-negligible impact.

Once the representative release scenarios were selected, it was necessary to calibrate their frequencies to match the calculated Surry impact given in NUREG-1150. The four representative release scenarios were input to MACCS with parameters set identically to those used in the NUREG-1150 study. The NUREG-1150 analysis predicted a mean probability per reactor-year of 2.0×10^{-6} and 5.23×10^{-3} for prompt and latent fatalities, respectively. As shown in Table 1, the use of the four input scenarios results in a good correlation to the NUREG-1150 calculated prompt fatality rate but over-predicts the latent fatality rate. This was accounted for through a reduction factor on the annual frequency of scenario 17-1. Reducing the frequency of this scenario to 1.4×10^{-5} per year results in a latent fatality rate matching that given in NUREG-1150 and still retains the balance of the source term groupings.

FIGURE A-1

Distribution of Source Terms with Nonzero Early Fatality

Distribution of Source Terms with Zero Early Fatality

Source Groups:

- I
- II
- III
- IV

Representative Source Term:

- 11-3
- 7-3
- 3-3
- 17-1

FIGURE A-2
Selection Of Category Scenarios

Category	NUREG-1150 Groups	Representative Source Term	Combined Frequency	Expected Annual Risk	
				Prompt	Latent
I	10,11,5	11-3	3.24E-07	1.31E-06	9.46E-04
II	6,7,8,12,13,14	7-3	9.34E-07	4.70E-07	1.31E-03
III	1,2,3,4,9	3-3	1.72E-06	2.70E-07	1.37E-03
IV	15,16,17	17-1	3.74E-05	0	4.23E-03

Total Predicted Annual Risk:	2.05E-06	7.85E-03
NUREG-1150 Results:	2.00E-06	5.23E-03

TABLE A-1

OPERATING UNIT SOURCE TERMS EVALUATED WITH UNIFORM POPULATION (500 PEOPLE/SQ MILE)

OVERALL WEIGHTED ANALYSIS

Population Distribution		Pop Centroid (mi)	Individual Fatality Rate (Fatality per Rx-Yr)		% of Safety Goal		Annular Population Dose (Rem per Rx-Yr)			Pop. Dose/Mile (Rem per Rx-Yr)			Individual Dose (Rem per Rx-Yr)	
Inside radius (km)	Outside radius (km)		Early	Latent	Early	Latent	Early	Chronic	Combined	Early	Chronic	Combined	Early	Chronic Combined
0.5	1.2	0.59	4.24E-08	2.28E-08	8.48	1.14	0.16	0.03	0.19	0.38	0.06	0.44	5.77	6.74
1.2	1.6	0.89	2.25E-08		4.51		0.10	0.02	0.12	0.40	0.07	0.47	3.69	4.37
1.6	2.1	1.18	1.11E-08	1.91E-08	2.22	0.95								
2.1	3.2	1.73	4.86E-09	1.43E-08	0.97	0.72	0.22	0.08	0.30	0.33	0.12	0.45	1.61	2.19
3.2	4	2.28	3.13E-09	1.27E-08	0.63	0.64	0.16	0.09	0.25	0.33	0.18	0.50	0.82	1.25
4.8	5.6	3.25	5.14E-10	9.58E-09	0.10	0.48	0.43	0.35	0.79	0.22	0.18	0.40	0.29	0.54
5.6	8.1	4.43	3.82E-11	6.43E-09	0.01	0.32								
8.1	11.3	6.24	5.09E-13	3.91E-09	0.00	0.20	0.49	0.95	1.44	0.17	0.32	0.49	0.12	0.34
11.3	16.1	8.85	1.75E-14	2.40E-09	0.00	0.12	0.47	1.40	1.87	0.12	0.36	0.48	0.05	0.21
16.1	20.9	11.77												
25.8	32.2	18.34				0.04								
32.2	40.2	22.90		8.89E-10			0.04	0.16	0.20	0.09	0.32	0.40	0.03	0.13
40.2	41	24.85				0.03	0.05	0.19	0.24	0.07	0.31	0.39	0.02	0.10
48	49	29.69		6.11E-10										0.10

Note: Table blanks indicate that no data was generated for that distance interval and effect

TABLE A-2

**OPERATING UNIT SOURCE TERMS EVALUATED WITH UNIFORM POPULATION
(500 PEOPLE/SQ MILE) AND STRATIFIED POPULATION**

Population Distribution			Combined Source Terms					
Inside radius (km)	Outside radius (km)	Pop Centroid (mi)	Avg. Miles	Stratified Pop	Strat. Area	Unified Pop	Unit. Area	
0.5	1.2	0.59	0.59	0.05	0.01	0.01	0.21	
1.2	1.6	0.89	0.89	0.05	0.01	0.01	0.47	
1.6	2.1	1.18	1.18	0.05				
2.1	3.2	1.73	1.73	0.06	0.04	0.04	0.45	0.38
3.2	4	2.28	2.28	0.06	0.03			
4	4.8	2.77	2.77	0.06				
4.8	5.6	3.25	3.25	0.05	0.05	0.05	0.50	0.73
5.6	8.1	4.43	4.43	0.05	0.06	0.06		
8.1	11.3	6.24	6.24	0.05	0.08	0.08	0.40	1.35
11.3	16.1	8.85	8.85	0.05	0.12	0.12		
16.1	20.9	11.77	11.77	0.25	0.43	0.43	0.49	2.48
20.9	25.8	14.74	14.74	0.24	0.72	0.72		
25.8	32.2	18.34	18.34	0.24	0.86	0.86	0.48	3.19
32.2	40.2	22.90	22.90	0.65	2.02	2.02		
40.2	41	24.85	24.85	0.60	1.21	1.21	0.40	2.87
41	42	25.42	25.42	0.62	0.35	0.35		
42	43	26.03	26.03	0.60	0.37	0.37		
44	45	27.25	27.25	0.59	0.72	0.72		
46	47	28.47	28.47	0.58	0.71	0.71		
48	49	29.69	29.69	0.57	0.71	0.71	0.39	1.92

Total Dose (man-rem per year) **8.54** **13.13**

Note: Table blanks indicate that no data was generated for that distance and effect

TABLE A-3

0.33 MILE EXCLUSION ZONE WITH EVACUATION

OVERALL WEIGHTED ANALYSIS

Population Distribution		Pop Centroid (mi)	Individual Fatality Rate (Fatality per Rx-Yr)		% of Safety Goal		Annular Pop. Dose (Rem per Rx-Yr)		Pop. Dose/Mile (Rem per Rx-Yr)		Individual Dose (Rem per Rx-Yr)	
Inside radius (km)	Outside radius (km)		Early	Latent	Early	Latent	Combined	Area	Combined	Area	Combined	Area
0.5	1.2	0.59	4.24E-08	2.28E-08	8.48	1.14	0.19	0.44	0.26	0.26	6.74	
1.2	1.6	0.89	2.25E-08		4.51	0.00	0.12	0.47	0.14	0.14	4.37	
1.6	2.1	1.18	1.11E-08	1.91E-08	2.22	0.95						
2.1	3.2	1.73	4.86E-09	1.43E-08	0.97	0.72	0.30	0.45	0.38	0.38	2.19	
3.2	4	2.28	3.13E-09	1.27E-08	0.63	0.64						
4.8	5.6	3.25	5.14E-10	9.58E-09	0.10	0.48	0.25	0.50	0.73	0.73	1.25	
5.6	8.1	4.43	3.82E-11	6.43E-09	0.01	0.32						
8.1	11.3	6.24	5.09E-13	3.91E-09	0.00	0.20	0.79	0.40	1.35	1.35	0.54	
11.3	16.1	8.85	1.75E-14	2.40E-09	0.00	0.12						
16.1	20.9	11.77				0.00	1.44	0.49	2.47	2.47	0.34	
25.8	32.2	18.34					1.87	0.48	3.19	3.19	0.21	
32.2	40.2	22.90				0.04	0.20	0.40	2.87	2.87	0.13	
40.2	41	24.85					0.24	0.39	1.92	1.92	0.10	
48	49	29.69				0.03						

Note: Table blanks indicate that no data was generated for that distance interval and effect

TABLE A-4

0.2 MILE EXCLUSION ZONE WITH EVACUATION

OVERALL WEIGHTED ANALYSIS

Population Distribution		Pop Centroid (mi)	Individual Fatality Rate (Fatality per Rx-Yr)		% of Safety Goal		Annular Population (Rem per Rx-Yr)		Pop. Dose/Mile (Rem per Rx-Yr)		Individual Dose (Rem per Rx-Yr) Combined
Inside radius (km)	Outside radius (km)		Early	Latent	Early	Latent	Combined	Latent	Combined	Area	
0.3	1.2	0.55	4.87E-08	2.33E-08	9.74	1.17	0.24	0.24	0.44	0.24	7.54
1.2	1.6	0.89	2.26E-08		4.51		0.12	0.12	0.47	0.16	4.37
1.6	2.1	1.18	1.11E-08	1.91E-08	2.22	0.95					
2.1	3.2	1.73	4.86E-09	1.43E-08	0.97	0.72	0.30	0.30	0.45	0.38	2.19
3.2	4	2.28	3.13E-09	1.28E-08	0.63	0.64					
4.8	5.6	3.25	5.14E-10	9.58E-09	0.10	0.48	0.25	0.25	0.50	0.73	1.25
5.6	8.1	4.43	3.82E-11	6.43E-09	0.01	0.32	0.79	0.79	0.40	1.35	0.54
8.1	11.3	6.24	5.09E-13	3.91E-09	0.00	0.20					
11.3	16.1	8.85	1.75E-14	2.40E-09	0.00	0.12					
16.1	20.9	11.77					1.44	1.44	0.49	2.48	0.34
25.8	32.2	18.34					1.87	1.87	0.48	3.19	0.21
32.2	40.2	22.90		9.49E-10		0.05					
40.2	41	24.85					0.20	0.20	0.40	2.87	0.13
48	49	29.69		6.40E-10		0.03	0.24	0.24	0.39	1.91	0.10

Note: Table blanks indicate that no data was generated for that distance interval and effect

TABLE A-5

0.2 MILE EXCLUSION ZONE WITHOUT EVACUATION

OVERALL WEIGHTED ANALYSIS

Population Distribution			Individual Fatality Rate		% of Safety Goal		Annular Population D	Pop. Dose/Mile	Individual Dose	
Inside radius	Outside radius	Pop Centroid	(Fatality per Rx-Yr)				(Rem per Rx-Yr)	(Rem per Rx-Yr)	(Rem per Rx-Yr)	
(km)	(km)	(mi)	Early	Latent	Early	Latent	Combined	Combined	Area	Combined
0.3	0.5	0.26	1.67E-07	1.17E-07	33.35	5.85	0.12	0.98	0.26	31.58
0.5	1.2	0.59					0.19	0.76	0.68	7.00
1.2	1.6	0.89	3.93E-08		7.86					
1.6	2.1	1.18	2.51E-08	3.31E-08	5.02	1.66				
2.1	3.2	1.73	1.23E-08		2.46		0.43	0.64	0.58	3.14
3.2	4	2.28	6.04E-09	1.62E-08	1.21	0.81				
4.8	5.6	3.25	1.45E-09	1.01E-08	0.29	0.50	0.28	0.57	0.93	1.42
5.6	8.1	4.43	4.45E-10	6.59E-09	0.09	0.33			0.34	
8.1	11.3	6.24	1.86E-11	3.86E-09	0.00	0.19	0.87	0.45		0.60
11.3	16.1	8.85		2.84E-09		0.14				
16.1	20.9	11.77					1.44	0.49	0.72	0.34
25.8	32.2	18.34					1.87	0.48	3.18	0.21
32.2	40.2	22.90		9.49E-10		0.05			1.09	
40.2	41	24.85					0.20	0.40		0.13
48	49	29.69		6.40E-10		0.03	0.24	0.39	1.32	0.10

Note: Table blanks indicate that no data was generated for that distance interval and effect

EVALUATION OF OFFSITE RISK FROM NORMAL NUCLEAR POWER PLANT OPERATIONS

EARLY SITE PERMIT DEMONSTRATION PROGRAM

MARCH 1993

March 1993

EVALUATION OF OFFSITE RISK FROM NORMAL NUCLEAR REACTOR OPERATIONS

I. Summary

In this evaluation, the performance of several of the most recently licensed light-water reactors was examined with regard to the conformance with the U.S. Nuclear Regulatory Commission (NRC) safety goal for risk to the general public in the vicinity of these plants. The NRC safety goal was translated into an annual dose level of approximately 6.7 mrem/yr, assuming a 60 year exposure to plant operation. Licensee generated data for offsite dose from routine operation as documented in Semi-Annual Effluent Reports was obtained for a sample of recently licensed light-water reactors for the period 1987 - 1991. This data was compiled, and average offsite dose impacts from liquid and atmospheric releases calculated for all evaluated plants. From this data it was concluded that routine operation of these plants is well within the offsite risk limits defined in the NRC safety goal and, by inference, that routine operation of advanced light water reactors (ALWRs) will also be well within the safety goal risk limits. An evaluation of the offsite risks from postulated accidents at nuclear power plants is contained in the companion report, *Evaluation Of The Effects Of Population Distribution On The Ability To Meet NRC Safety Goals For Offsite Risk*.

II. Objective

To determine and compare with the NRC safety goal, the risk to the population in the area near existing nuclear power plants so as to bound the expected risk from an ALWR from routine plant operation.

III. Background

The NRC policy statement, *Safety Goals for the Operations of Nuclear Power Plants*,¹ defines the safety goal from plant operations as "...one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes." An annual dose level, for constant exposure over 60 years of plant operation, equivalent to this goal is calculated as approximately 6.7 mrem/yr (see Appendix B).

In the policy statement, the Commission defines the population considered subject to significant risk as the population within 10 miles of the plant site. For reporting purposes, NRC reactor licensees calculate population doses out to 50 miles but do not report incremental doses to only the 10 mile population. Thus, the dose to the hypothetical maximum exposed individual (MEI) at each of the evaluated plants was used as a conservative (by at least one order of magnitude) surrogate for the average dose to individuals within 10 miles. Doses to this individual serve as a bounding condition for determinations of latent cancer risks.

IV. Methodology

The methodology used in this evaluation is to obtain, chart, and trend for each of the evaluated plants the maximum offsite dose impact from airborne and liquid releases from a sample of recently licensed Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). This data demonstrates that the impact from routine operation of these plants has resulted in offsite doses much less than the dose corresponding to the NRC safety goal (6.7 mrem/yr).

¹U. S. Nuclear Regulatory Commission, November 30, 1988.

The method for selection of the subject plants was to include the last twelve licensed PWRs and ten BWRs as documented in *Commercial Nuclear Power Plants*, Edition No. 19.² The only exception was Comanche Peak-1 which began commercial operation on August 13, 1990, and for which there are insufficient years of operation to provide substantial data. The selection of twelve PWRs was made to provide a data set containing ten PWRs having a licensed power of over 3000 MWt, without excluding two newer plants having lower licensed power levels. For multi-unit sites the oldest unit was chosen to provide a more complete data set. The exceptions to this rule were: (1) Beaver Valley, where unit 1 is significantly (approx. 11 years) older than unit 2; (2) Millstone, where units 1 and 2 are significantly older than unit 3 and of different design; and (3) Nine Mile Point, where unit 1 is significantly (approx. 19 years) older than unit 2. The selected plants are listed in Appendix C along with their licensed power and the date of assumption of commercial operations.

The analysis used the previous five years (1987 - 1991) Semi-Annual Effluent Reports for the subject plants. These reports are prepared in accordance with NRC Regulatory Guide 1.21 and present doses to the MEI to demonstrate compliance with plant Radiological Effluent Technical Specifications. The offsite dose values were extracted from these reports, analyzed, charted, and trended. The following values were included in the evaluation: (1) total body dose from liquid effluents; (2) total body dose from airborne emissions of iodines, particulates, and tritium; and (3) gamma "airdose" from airborne emissions of noble gases.

A listing of the offsite dose commitments for the period of 1987-1991 for each selected plant is provided in Appendix D. There is currently no NRC-mandated reporting format for offsite dose values; each utility reports these numbers differently. With these reporting disparities in mind, the following assumptions were made in extracting these values from the Semi-Annual Effluent Reports:

- For the following sites, the offsite dose impact was calculated on a site-wide, rather than a per unit basis. For these sites, the following techniques were used to derive an approximation of the dose impact of the selected unit.
 - Palo Verde: All dose values were presented on a site-wide (three unit) basis. Since all units are identical, the dose impacts were divided by three to determine an approximate Palo Verde-1 contribution. The exception to this was 1987, where the total was divided by two because Palo Verde-3 did not achieve initial criticality until October 1987.
 - Byron: All dose values for 1987 and 1988 were presented on a site-wide (two unit) basis. Since both units are identical, the dose impacts for these years were divided by two to determine an approximate Byron -1 contribution.
 - Nine Mile Point: All dose values were presented for both Nine Mile Point units and the James A. Fitzpatrick plant combined. For these sites, the total dose was divided by three, which should yield a conservative value because Nine Mile Point 2 is significantly newer than the other units and because of new effluent control technologies, its dose impact is probably less than one third of the total value.
 - Susquehanna: All dose values were presented on a site-wide (two unit) basis. Since the units are identical, the dose impacts were divided by two to determine an approximate Susquehanna-1 contribution.

²HALLIBURTON NUS Environmental Corporation, December 1991.

- LaSalle: All gaseous dose values were presented on a site-wide (two unit) basis, while liquid dose impacts are presented on a per unit basis. Since the units are identical, the dose impacts of gaseous releases were divided by two to determine an approximate LaSalle-1 contribution.
- For plants that only reported the critical organ dose for airborne releases of iodines, particulates, and tritium and not the total body dose, the critical organ dose was used as a surrogate value for total body dose. This should lead to a conservative result as the critical organ dose should be greater than or equal to the total body dose.
- When extracting the total body dose from airborne releases for plants that did not distinguish between noble gas releases and releases of iodines, particulates, and tritium, the listed values were considered to be from releases of iodines, particulates, and tritium. This should lead to a conservative result because the listed value will be greater than or equal to the dose from releases of iodines, particulates, and tritium.

If a given parameter was not reported in the Semi-Annual Effluent Report, the parameter is listed in Appendix D as N/R. If a given parameter was not applicable to the selected plant (i.e., the time period was prior to initial criticality or the effluent pathway does not exist), the parameter is listed in Appendix D as N/A.

V. Results

For the data presented in Appendix D, the mean and standard deviation of each dose parameter, were calculated (for each reactor type) for each year (see Appendix E) and a summary of the results is presented below.

PWRs

Year	Average Total Body Dose, Liquid Releases (mrem)	Average Total Body Dose, Particulate, Iodine, and Tritium Releases (mrem)	Gamma Airdose (mrad)
1987	(1.25 ± 1.54) E-01	(1.58 ± 2.23) E-01	(4.57 ± 4.65) E-01
1988	(1.30 ± 1.85) E-01	(6.18 ± 6.85) E-02	(2.29 ± 3.01) E-01
1989	(1.39 ± 1.78) E-01	(1.15 ± 2.33) E-01	(6.82 ± 10.9) E-02
1990	(2.29 ± 3.13) E-01	(7.58 ± 15.5) E-02	(1.85 ± 3.59) E-01
1991	(2.49 ± 3.58) E-01	(8.56 ± 11.8)E-02	(9.45 ± 14.1) E-02
Average	(1.74 ± 1.12) E-01	(9.92 ± 7.66) E-02	(2.07 ± 1.37) E-01

BWRs

Year	Average Total Body Dose, Liquid Releases (mrem)	Average Total Body Dose Particulate, Iodine, and Tritium Releases (mrem)	Gamma Airdose (mrad)
1987	(1.47 ± 2.48) E-01	(2.14 ± 4.48) E-01	(8.07 ± 12.2) E-02
1988	(7.89 ± 18.1) E-02	(7.96 ± 20.4) E-01	(4.46 ± 10.3) E-01
1989	(1.91 ± 3.53) E-01	(4.55 ± 6.66) E-02	(1.54 ± 2.32) E-01
1990	(9.57 ± 21.1) E-02	(6.06 ± 14.2) E-01	(7.57 ± 16.2) E-02
1991	(8.24 ± 18.0) E-02	(3.33 ± 6.66) E-01	(4.57 ± 5.94) E-02
Average	(1.19 ± 1.09) E-01	(3.99 ± 5.23) E-01	(1.60 ± 2.15) E-01

Chronological offsite dose impacts are graphically represented in Figure II-1 through Figure II-6. For each year, the average dose value is plotted along with an error bar representing two standard deviations above and below the average. This representation approximates a 95 percent confidence interval for each result. The 95 percent confidence interval (± 2 standard deviations) for the mean value over the entire time period (final row in the above tables) is presented below.

Type	Average Total Body Dose, Liquid Releases (mrem)	Average Total Body Dose Particulate, Iodine, and Tritium Releases (mrem)	Gamma Airdose (mrad)
PWR	(1.74 ± 2.24) E-01	(9.92 ± 15.32) E-02	(2.07 ± 2.74) E-01
BWR	(1.19 ± 2.18) E-01	(3.99 ± 10.46) E-01	(1.60 ± 4.30) E-01

Even if the three dose values were added (which is not a realistic assumption because these doses are calculated for different critical receptor locations, and it is usually not possible to receive the maximum dose from all pathways), the results would be 0.48 mrem/yr for PWRs and 0.68 mrem/yr for BWRs. This cumulative value, at a 95 percent confidence level, is well below the NRC safety goal of 6.7 mrem/yr.

VI. Conclusions

As demonstrated by the results presented above, it can be concluded with confidence that routine operation of current light-water reactors is well within the NRC safety goal risk limits to the population residing in the vicinity of these plants. This conclusion was determined by analyzing the routine operating history of recently licensed commercial light-water reactors. From this result, it can be further concluded that operation of ALWRs, because of the use of advanced effluent control and treatment technologies which are expected to be incorporated in the design and development of ALWRs, will also satisfy the NRC safety goal.

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Figure 1

Figure 2

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Figure 3

Figure 4

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Figure 5

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Figure 6

APPENDIX A

Calculation Of Annual Dose Level Equivalent To The NRC Safety Goal

International Commission on Radiological Protection Publication 60, *1990 Recommendations of the International Commission on Radiological Protection*, presents revised risk coefficients for fatal cancer from exposure to ionizing radiation. Table S-3 lists the risk coefficient for the whole population as $5 \times 10^{-2} \text{ Sv}^{-1}$. This is equivalent to $5 \times 10^{-4} \text{ rem}^{-1}$. This is assumed to be a cumulative lifetime exposure.

Assuming a normal fatal cancer incidence of 20% (BIER V), the safety goal will correspond to 0.1 % (10^{-3}) of this:

$$\frac{(0.2)(10^{-3})}{5 \times 10^{-4} \text{ rem}^{-1}} = 0.4 \text{ rem} = 400 \text{ mrem (lifetime)}$$

Assuming a nuclear plant life of 60 years, this equals $(400 \text{ mrem})/(60 \text{ yr}) = 6.7 \text{ mrem/yr}$.

APPENDIX B

Reactors Chosen for Evaluation of Normal Operational Dose

<u>Plant</u>	<u>Type</u>	<u>Licensed Power (MWt)</u>	<u>Commercial Operation</u>
Beaver Valley 2	PWR	2660	11/17/87
Braidwood 1	PWR	3411	07/29/88
Byron 1	PWR	3411	09/16/85
Callaway	PWR	3565	12/19/84
Catawba 1	PWR	3411	06/29/85
Harris 1	PWR	2775	05/02/87
Millstone 3	PWR	3411	04/23/86
Palo Verde 1	PWR	3800	01/28/86
South Texas Project 1	PWR	3800	08/25/88
Vogtle 1	PWR	3411	06/01/87
Waterford 3	PWR	3390	09/24/85
Wolf Creek	PWR	3411	09/03/85
Clinton	BWR	2894	11/24/87
Fermi 2	BWR	3292	01/23/88
Hope Creek	BWR	3293	12/20/86
LaSalle 1	BWR	3323	01/01/84
Limerick 1	BWR	3293	02/01/86
Nine Mile Point 2	BWR	3323	03/11/88
Perry 1	BWR	3579	11/18/87
River Bend	BWR	2894	06/16/86
Susquehanna 1	BWR	3293	06/08/83
WNP 2	BWR	3323	12/13/84

March 1993

APPENDIX C

**Calculated Offsite Annual Dose
To Maximum Exposed Individuals
From Normal Operations
(1987-1991)**

March 1993

PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Over Valley 2 <u>TYPE</u> PWR <u>LICENSED POWER</u> 2660 <u>COMMERCIAL OPERATION</u> 11/17/87	1987	N/A	N/A	N/A
	1988	6.27E-02	1.71E-02	7.62E-03
	1989	2.04E-02	7.36E-02	9.46E-03
	1990	3.67E-02	9.27E-02	1.39E-01
	1991	1.69E-02	2.25E-01	1.32E-02

Braidwood 1 <u>TYPE</u> PWR <u>LICENSED POWER</u> 3411 <u>COMMERCIAL OPERATION</u> 12/29/88	1987	N/R	N/R	N/R
	1988	1.59E-02	3.00E-03	0.00E+00
	1989	5.63E-02	3.00E-03	3.30E-02
	1990	N/R (data not available)	3.18E-03	1.18E-02
	1991	2.92E-03	1.96E-02	7.43E-02

Byron 1 <u>TYPE</u> PWR <u>LICENSED POWER</u> 3411 <u>COMMERCIAL OPERATION</u> 09/16/85	1987	7.65E-02	6.15E-02	4.00E-03
	1988	9.57E-02	8.25E-02	4.00E-03
	1989	2.38E-02	2.71E-04	1.06E-03
	1990	1.15E-02	7.15E-04	1.56E-03
	1991	N/R (data not available)	N/R (data not available)	N/R (data not available)

March 1993

PLANT

Blawie

TYPE

PWR

LICENSED POWER

3565

COMMERCIAL OPERATION

12/19/84

YEAR**TOTAL BODY
LIQUID DOSE
(MREM)****TOTAL BODY
PARTICULATE, I,
AND H-3 DOSE
(MREM)****GAMMA AIRDOSE
(MRAD)**

1987	1.7E-01	6.4E-03	4.0E-02
1988	1.19E-01	4.07E-03	7.98E-03
1989	3.38E-03	1.09E-03	1.10E-02
1990	5.72E-03	1.19E-03	1.27E-02
1991	4.35E-03	1.24E-03	4.21E-03

Catawba 1

TYPE

PWR

LICENSED POWER

3411

COMMERCIAL OPERATION

06/29/85

1987	4.56E-01	3.33E-01 (thyroid)	7.54E-01
1988	5.46E-01	1.42E-01 (thyroid)	6.33E-01
1989	2.67E-01	1.90E-01 (thyroid)	3.80E-01
1990	5.17E-01	5.51E-01 (thyroid)	8.86E-01
1991	3.49E-01	3.26E-01 (thyroid)	4.27E-01

Harris 1

TYPE

PWR

LICENSED POWER

2775

COMMERCIAL OPERATION

05/02/87

1987	4.98E-03	2.16E-02	3.45E-02
1988	4.66E-03	9.49E-02	1.48E-01
1989	2.71E-01	3.79E-02	5.93E-02
1990	9.59E-01	2.04E-02	3.11E-02
1991	4.45E-02	1.56E-02	2.49E-02

March 1993

PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Millstone 3 <u>TYPE</u> PWR <u>LICENSED POWER</u> 3411 <u>COMMERCIAL OPERATION</u> 04/23/86	1987	1.90E-02	1.74E-02	N/R
	1988	1.62E-02	5.03E-03	N/R
	1989	5.83E-02	1.07E-02	8.00E-03
	1990	2.57E-02	7.05E-02	1.12E-02
	1991	4.37E-02	9.96E-03	8.78E-03

Palo Verde 1
TYPE
 PWR
LICENSED POWER
 3800
COMMERCIAL OPERATION
 01/28/86

1987	N/A	2.61E-01 (thyroid)	3.25E-01
1988	N/A	1.17E-01 (thyroid)	1.74E-01
1989	N/A	5.13E-02 (thyroid)	7.87E-02
1990	N/A	5.67E-02 (total body)	8.33E-02
1991	N/A	5.77E-02 (total body)	1.00E-01

South Texas Project 1
TYPE
 PWR
LICENSED POWER
 3800
COMMERCIAL OPERATION
 08/25/88

1987	N/A	N/A	N/A
1988	1.40E-02	1.64E-01 (thyroid)	4.53E-01
1989	4.06E-02	8.33E-01 (thyroid)	6.15E-02
1990	7.04E-02	3.10E-02 (thyroid)	9.45E-03
1991	1.94E-01	3.33E-02 (thyroid)	4.17E-03

March 1993

PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Little 1 TYPE PWR LICENSED POWER 3411 COMMERCIAL OPERATION 06/01/87	1987	9.78E-03	9.83E-03	1.02E+00
	1988	1.74E-02	2.78E-03	7.13E-01
	1989	2.93E-02	3.71E-03	1.35E-03
	1990	1.78E-02	8.65E-04	1.06E-03
	1991	4.18E-02	1.96E-04	1.65E-03

Waterford 3 TYPE PWR LICENSED POWER 3390 COMMERCIAL OPERATION 02/24/85	1987	4.89E-02	6.61E-01	1.02E+00
	1988	4.37E-01	1.79E-01	7.55E-01
	1989	5.86E-01	8.90E-02	1.07E-01
	1990	3.39E-01	1.30E-01	1.01E-00
	1991	7.27E-01	2.40E-01	3.06E-01

Wolf Creek TYPE PWR LICENSED POWER 3411 COMMERCIAL OPERATION 09/03/85	1987	2.11E-01	5.37E-02	N/R
	1988	9.77E-02	1.04E-01	2.33E-02
	1989	1.77E-01	8.75E-02	N/R
	1990	3.10E-01	1.44E-02	2.78E-02
	1991	1.04E+00	1.31E-02	7.54E-02

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PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Anton TYPE BWR <u>LICENSED POWER</u> 2894 <u>COMMERCIAL OPERATION</u> 11/24/87	1987	N/R	N/R	N/R
	1988	1.01E-04	1.22E-02	1.23E-04
	1989	1.46E-03	2.04E-03	6.33E-04
	1990	2.96E-03	1.11E-04	6.88E-04
	1991	9.57E-03	1.59E-04	2.40E-05

Fermi 2 TYPE BWR <u>LICENSED POWER</u> 3292 <u>COMMERCIAL OPERATION</u> 01/23/88	1987	1.48E-03	0.00E+00	0.00E+00
	1988	4.15E-03	2.15E-04	1.79E-04
	1989	3.33E-02	6.26E-04	1.36E-01
	1990	5.13E-03	2.04E-04	4.58E-02
	1991	1.63E-02	3.31E-04	N/R

Hope Creek TYPE BWR <u>LICENSED POWER</u> 3293 <u>COMMERCIAL OPERATION</u> 12/20/86	1987	4.80E-01	1.31E+00 (thyroid)	N/R
	1988	5.39E-01	1.22E-02 (thyroid)	N/R
	1989	8.47E-01	9.36E-06 (thyroid)	N/R
	1990	6.87E-01	1.27E-02	N/R
	1991	1.36E-01	5.77E-05	N/R

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PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Salle 1 TYPE BWR LICENSED POWER 3323 COMMERCIAL OPERATION 01/01/84	1987	N/R	N/R	N/R
	1988	N/R	N/R	N/R
	1989	6.17E-05	7.45E-03	1.29E-02
	1990	1.81E-05	1.07E-04	2.04E-04
	1991	0.00E+00	1.15E-03	4.97E-03

Limerick 1 TYPE BWR LICENSED POWER 3293 COMMERCIAL OPERATION 03/01/86	1987	6.06E-01	2.06E-01 (bone)	1.14E-03
	1988	0.00E+00	1.73E-01 (thyroid)	5.70E-03
	1989	1.01E-01	7.07E-02 (skin)	5.70E-03
	1990	6.77E-02	4.50E-03 (thyroid)	1.31E-03
	1991	2.18E-02	9.68E-07 (GI)	2.33E-03

Nine Mile Point 2 TYPE BWR LICENSED POWER 3323 COMMERCIAL OPERATION 03/11/88	1987	0.00E+00	8.00E-03	N/R
	1988	3.33E-04	4.33E-03	N/R
	1989	3.47E-03	1.20E-02	N/R
	1990	3.73E-03	5.00E-03	N/R
	1991	1.52E-04	8.57E-03	N/R

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PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Try 1 TYPE BWR LICENSED POWER 3579 COMMERCIAL OPERATION 11/18/87	1987	2.06E-04	2.60E-02 (thyroid)	1.17E-01
	1988	1.65E-03	6.19E+00 (thyroid)	2.55E+00
	1989	5.18E-02	1.79E-01 (thyroid)	6.13E-01
	1990	9.64E-02	1.25E+00 (thyroid)	3.65E-02
	1991	5.63E-02	1.84E+00 (thyroid)	7.87E-02

River Bend TYPE BWR LICENSED POWER 2894 COMMERCIAL OPERATION 06/16/86	1987	7.00E-02	4.80E-03	N/R
	1988	1.62E-01	3.30E-03	N/R
	1989	8.70E-01	5.00E-03	N/R
	1990	9.00E-02	4.50E+00 (thyroid)	N/R
	1991	5.80E-01	1.30E+00 (thyroid)	N/R

Susquehanna 1 TYPE BWR LICENSED POWER 3293 COMMERCIAL OPERATION 06/08/83	1987	4.02E-04	4.88E-02	5.60E-03
	1988	4.00E-04	1.84E-02	2.14E-03
	1989	2.91E-04	2.80E-02	2.82E-03
	1990	2.19E-04	7.50E-02	5.25E-03
	1991	3.91E-04	4.77E-02 (lung)	3.80E-02

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PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
NP 2 TYPE BWR LICENSED POWER 3323 COMMERCIAL OPERATION 12/13/84	1987	1.7E-02	1.1E-01	2.8E-01
	1988	2.1E-03	7.5E-01	1.2E-01
	1989	1.8E-03	1.5E-01	3.2E-01
	1990	3.5E-03	2.1E-01	4.4E-01
	1991	5.2E-03	1.3E-01	1.5E-01

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APPENDIX D

Statistical Analysis of Operating Data

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Plant	Type	Licensed Power (MW)	Commercial Operation	Total Body Liquid Dose (mrem)				
				1987	1988	1989	1990	1991
Beaver Valley 2	PWR	2660	11/17/87	N/A	6.27E-02	2.04E-02	3.67E-02	1.69E-02
Braidwood 1	PWR	3411	7/29/88	N/A	1.59E-02	5.63E-02	N/R	2.92E-02
Byron 1	PWR	3411	9/16/85	7.65E-02	9.57E-02	2.38E-02	1.15E-02	N/R
Callaway	PWR	3565	12/19/84	1.70E-01	1.19E-01	3.38E-03	5.72E-03	4.35E-03
Catawba 1	PWR	3411	6/29/85	4.56E-01	5.46E-01	2.67E-01	5.17E-01	3.49E-01
Harris 1	PWR	2775	5/2/87	4.98E-03	4.66E-03	2.71E-01	9.59E-01	4.45E-02
Millstone 3	PWR	3411	4/23/86	1.90E-02	1.62E-02	5.83E-02	2.57E-02	4.37E-02
Palo Verde 1	PWR	3800	1/28/86	N/A	N/A	N/A	N/A	N/A
South Texas Project 1	PWR	3800	8/25/88	N/A	1.40E-02	4.06E-02	7.04E-02	1.94E-01
Vogtle 1	PWR	3411	6/1/87	9.78E-03	1.74E-02	2.93E-02	1.78E-02	4.18E-02
Waterford 3	PWR	3390	9/24/85	4.89E-02	4.37E-01	5.86E-01	3.39E-01	7.27E-01
Wolf Creek	PWR	3411	9/3/85	2.11E-01	9.77E-02	1.77E-01	3.10E-01	1.04E+00
Average				1.25E-01	1.30E-01	1.39E-01	2.29E-01	2.49E-01
Standard Deviation				1.54E-01	1.85E-01	1.78E-01	3.13E-01	3.58E-01
Clinton	BWR	2894	11/24/87	N/A	1.01E-04	1.46E-03	2.96E-03	9.57E-03
Ferni 2	BWR	3292	1/23/88	1.48E-03	4.15E-03	3.33E-02	5.13E-03	1.63E-02
Hope Creek	BWR	3293	12/20/86	4.80E-01	5.39E-01	8.47E-01	6.87E-01	1.36E-01
LaSalle 1	BWR	3323	1/1/84	N/R	N/R	6.17E-05	1.81E-05	0.00E+00
Limerick 1	BWR	3293	2/1/86	6.06E-01	0.00E+00	1.01E-01	6.77E-02	2.18E-02
Nine Mile Point 2	BWR	3323	3/11/88	0.00E+00	3.33E-04	3.47E-03	3.73E-03	1.52E-04
Perry 1	BWR	3579	11/18/87	2.06E-04	1.65E-03	5.18E-02	9.64E-02	5.63E-02
River Bend	BWR	2894	6/16/86	7.00E-02	1.62E-01	8.70E-01	9.00E-02	5.80E-01
Susquehanna 1	BWR	3293	6/8/83	4.02E-04	4.00E-04	2.91E-04	2.19E-04	3.91E-04
WNP 2	BWR	3323	12/13/84	1.70E-02	2.10E-03	1.80E-03	3.50E-03	3.20E-03
Average				1.47E-01	7.89E-02	1.91E-01	9.57E-02	8.24E-02
Standard Deviation				2.48E-01	1.81E-01	3.53E-01	2.11E-01	1.80E-01

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Plant	Type	Licensed Power (MWt)	Commercial Operation	Total Body Particulate, Iodine, And Tritium Dose (mrem)				
				1987	1988	1989	1990	1991
Beaver Valley 2	PWR	2660	11/17/87	N/A	1.71E-02	7.36E-02	9.27E-02	2.25E-01
Braidwood 1	PWR	3411	7/29/88	N/A	3.00E-03	3.00E-03	3.18E-03	1.96E-02
Byron 1	PWR	3411	9/16/85	6.15E-02	8.25E-02	2.71E-04	7.15E-04	N/R
Callaway	PWR	3565	12/19/84	6.40E-03	4.07E-03	1.09E-03	1.19E-03	1.24E-03
Catawba 1	PWR	3411	6/29/85	3.33E-01	1/42E-01	1.90E-01	5.51E-01	3.26E-01
Harris 1	PWR	2775	5/2/87	2.16E-02	9.49E-02	3.79E-02	2.04E-02	1.56E-02
Millstone 3	PWR	3411	4/23/86	1.74E-02	5.03E-03	1.07E-02	7.05E-03	9.96E-03
Palo Verde 1	PWR	3800	1/28/86	2.61E-01	1.17E-01	5.13E-02	5.67E-02	5.77E-02
South Texas Project 1	PWR	3800	8/25/88	N/A	1.64E-01	8.33E-01	3.10E-02	3.33E-02
Vogtle 1	PWR	3411	6/1/87	9.83E-03	2.78E-03	3.71E-03	8.65E-04	1.96E-04
Waterford 3	PWR	3390	9/24/85	6.61E-01	1.79E-01	8.90E-02	1.30E-01	2.40E-01
Wolf Creek	PWR	3411	9/3/85	5.37E-02	1.04E-02	8.75E-02	1.44E-02	1.31E-02
Average				1.58E-01	6.18E-02	1.15E-01	7.58E-02	8.56E-02
Standard Deviation				2.23E-01	6.85E-02	2.33E-01	1.55E-01	1.18E-01
Clinton	BWR	2894	11/24/87	N/A	1.22E-02	2.04E-03	1.11E-04	1.59E-04
Fermi 2	BWR	3292	1/23/88	0.00E+00	2.15E-04	6.26E-04	2.04E-04	3.31E-04
Hope Creek	BWR	3293	12/20/86	1.31E+00	1.22E-02	9.36E-06	1.27E-02	5.77E-05
LaSalle 1	BWR	3323	1/1/84	N/R	N/R	7.45E-03	1.07E-04	1.15E-03
Limerick 1	BWR	3293	2/1/86	2.06E-01	1.73E-01	7.07E-02	4.50E-03	9.68E-07
Nine Mile Point 2	BWR	3323	3/11/88	8.00E-03	4.33E-03	1.20E-02	5.00E-03	8.57E-03
Perry 1	BWR	3579	11/18/87	2.60E-02	6.19E+00	1.79E-01	1.25E+00	1.84E+00
River Bend	BWR	2894	6/16/86	4.80E-03	3.30E-03	5.00E-03	4.50E+00	1.30E+00
Susquehanna 1	BWR	3293	6/8/83	4.88E-02	1.84E-02	2.80E-02	7.50E-02	4.77E-02
WNP 2	BWR	3323	12/13/84	1.10E-01	7.50E-01	1.50E-01	2.10E-01	1.30E-01
Average				2.14E-01	7.96E-01	4.55E-02	6.06E-01	3.33E-01
Standard Deviation				4.48E-01	2.04E+00	6.66E-02	1.42E+00	6.66E-01

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Plant	Type	Licensed Power (MWt)	Commercial Operation	Gamma Airdose (mrad)				
				1987	1988	1989	1990	1991
Beaver Valley 2	PWR	2660	11/17/87	N/A	7.62E-03	9.46E-03	1.39E-01	1.32E-02
Braidwood 1	PWR	3411	7/29/88	N/A	0	3.30E-02	1.18E-02	7.43E-02
Byron 1	PWR	3411	9/16/85	4.00E-03	4.00E-03	1.06E-03	1.56E-03	N/R
Callaway	PWR	3565	12/19/84	4.00E-02	7.98E-03	1.10E-02	1.27E-02	4.21E-03
Catawba 1	PWR	3411	6/29/85	7.54E-01	6/33E-01	3.80E-01	8.86E-01	4.27E-01
Harris 1	PWR	2775	5/2/87	3.45E-02	1.48E-01	5.93E-02	3.11E-02	2.49E-02
Millstone 3	PWR	3411	4/23/86	N/R	N/R	8.00E-03	1.12E-02	8.78E-03
Palo Verde 1	PWR	3800	1/28/86	3.25E-01	1.74E-01	7.87E-02	8.33E-02	1.00E-01
South Texas Project 1	PWR	3800	8/25/88	N/A	4.53E-01	6.15E-02	9.45E-03	4.17E-03
Vogtle 1	PWR	3411	6/1/87	1.02E+00	7.13E-01	1.35E-03	1.06E-03	1.65E-03
Waterford 3	PWR	3390	9/24/85	1.02E+00	7.55E-01	1.07E-01	1.01E+00	3.06E-01
Wolf Creek	PWR	3411	9/3/85		2.33E-02		2.78E-02	7.54E-02
Average				4.57E-01	2.29E-01	6.82E-02	1.85E-01	9.45E-02
Standard Deviation				4.65E-01	3.01E-01	1.09E-01	3.59E-01	1.41E-01
Clinton	BWR	2894	11/24/87	N/A	1.23E-04	6.33E-04	6.88E-04	2.40E-05
Fermi 2	BWR	3292	1/23/88	0.00E+00	1.79E-04	1.36E-01	4.58E-02	N/R
Hope Creek	BWR	3293	12/20/86	N/R	N/R	N/R	N/R	N/R
LaSalle 1	BWR	3323	1/1/84	N/R	N/R	1.29E-02	2.04E-04	4.97E-03
Limerick 1	BWR	3293	2/1/86	1.14E-03	5.70E-03	5.70E-03	1.31E-03	2.33E-03
Nine Mile Point 2	BWR	3323	3/11/88	N/R	N/R	N/R	N/R	N/R
Perry 1	BWR	3579	11/18/87	1.17E-01	2.55E+00	6.13E-01	3.65E-02	7.87E-02
River Bend	BWR	2894	6/16/86	N/R	N/R	N/R	N/R	N/R
Susquehanna 1	BWR	3293	6/8/83	5.60E-03	2.14E-03	2.82E-03	5.25E-03	3.80E-02
WNP 2	BWR	3323	12/13/84	2.80E-01	1.20E-01	3.10E-01	4.40E-01	1.50E-01
Average				8.07E-02	4.46E-01	1.54E-01	7.57E-02	4.57E-02
Standard Deviation				1.22E-01	1.03E+00	2.32E-01	1.62E-01	5.94E-02

POPULATION DENSITY REPORT

**EVALUATION OF POPULATION DISTRIBUTION
RELATIVE TO MEETING THE QUANTITATIVE
HEALTH OBJECTIVES OF THE NRC SAFETY GOAL
POLICY FOR OFFSITE RISK ASSOCIATED WITH
NUCLEAR POWER PLANTS**

EARLY SITE PERMIT DEMONSTRATION PROGRAM

MARCH 1993

March 1993

Executive Summary

In this study the proposed U.S. Nuclear Regulatory Commission (NRC) population density and exclusion zone size criteria were evaluated for their potential effectiveness to meet the quantitative public health and safety objectives associated with the NRC Safety Goal Policy. In addition, the individual and societal risks from postulated accidents (based on NUREG-1150 methods) were determined and compared to the quantitative prompt and latent health objectives of the safety goal policy.

The proposed revisions to 10 CFR 100 regarding population density and exclusion zone size criteria were evaluated using the MELCOR Accident Consequence Code System (MACCS). Surry Nuclear Plant radiological and meteorological input data were taken from the NUREG-1150 analysis. Surry was selected for the following reasons: it is one of the five plants evaluated by the NRC in NUREG-1150; it is one of two plants for which NUREG-1150 considered analysis of external events; and its source term data are readily available.

This analysis demonstrates that a nuclear power plant of current design located in an area of controlled population distribution (500 people per square mile) and minimum exclusion zone size (0.33 miles), as specified by the NRC in the proposed 10 CFR 100 revision, results in a calculated societal radiation dose risk from reactor accidents of approximately 13 person-rem per reactor-year. The calculated individual radiological risks from postulated accidents demonstrate that the prompt and latent Quantitative Health Objectives (QHOs) of the NRC Safety Goal Policy are met by a wide margin.

NUREG-1150 used site-related factors including plant design features (source term), meteorology, population density and distribution, and estimates of the effectiveness of emergency plans. Therefore, by using the NUREG-1150 study methodology, which fully considers population density and distribution, it is possible to compare offsite health effects to the QHOs of the NRC Safety Goal Policy.

Further, the population density criterion proposed for codification will not provide the flexibility needed to select consistently the most favorable sites. Alternative population configurations with larger overall populations may result in lower societal radiation risk yet fail the NRC proposed criterion. This may result in eliminating acceptable (possibly the more favorable) sites from further consideration while less favorable sites survive the site selection screening process. The proposed criterion could unnecessarily constrain future activities at existing sites (if they were to be evaluated under the proposed rule), sites that have previously been approved but have never had operating reactors, and new sites that would likely be considered for future reactors. For example, in the NRC Staff briefing to the Commission, the staff stated that eight licensed sites do not currently meet the five hundred persons per square mile population density criterion, and twenty-five licensed sites do not meet the 0.4 mile exclusion area distance criterion. All of these sites have been demonstrated to adequately protect the public health and safety and easily meet the NRC Safety Goal Policy.

If codified, then the restrictive numeric criteria would establish a de facto safety goal in regulation that is more stringent than the NRC Safety Goal Policy and is applicable to future plants only. The next generation of nuclear plants will have comprehensively considered 30 years of accumulated experience including improved understanding of postulated severe accident phenomena, probability, and consequences. Therefore, these plants will reflect in their design, construction, and operation risk characteristics as low or lower than those for existing plants.

The proposed criteria would be significantly more restrictive than what is required to meet the QHOs of the Safety Goal Policy and would therefore be unnecessarily restrictive. Since the risk to the public would be so much less than the 0.1 % increase from all other factors established by the QHOs of the Safety Goal Policy, there is insufficient health risk basis for establishing such a restrictive criteria.

March 1993

EVALUATION OF THE PROPOSED NRC POPULATION DENSITY CRITERION ON SEVERE ACCIDENT RISK

I. Summary

10 CFR Part 100, *Reactor Site Criteria*, sets forth the criteria for an exclusion area, a low population zone, and a population center distance around nuclear plants. The U.S. Nuclear Regulatory Commission (NRC) has established population density guidelines in Regulatory Guide 4.7 to be used as an indicator as to whether a facility meets the Federal requirements. The NRC has issued for public comment the proposed change to 10 CFR Part 100 which, if promulgated, would codify these population density and exclusion zone size guidelines. This study was undertaken to evaluate (1) the proposed population and exclusion zone size criteria for their effectiveness in meeting the public health and safety objectives associated with the prompt and latent quantitative health objectives (QHOs) of the NRC Safety Goal Policy and (2) the individual and societal health and safety risks for several population distributions using the methodology used by the NRC in NUREG-1150, *Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants*.

The results of this study indicate that population density and exclusion zone size criteria proposed in 10 CFR Part 100 do not influence conformance to the QHOs of the safety goal policy. Further, the results of this study have shown that a reactor of current design located in an area of controlled population distribution and minimum exclusion zone size as specified by the NRC in the proposed 10 CFR Part 100 revision results in a calculated societal risk of approximately 13 person-rem per reactor-year. The study also demonstrates that such a facility meets the individual QHOs by a wide margin.

NUREG-1150 used site-related factors including plant design features (source term), meteorology, population density and distribution, and estimates of the effectiveness of emergency plans. Therefore, by using the NUREG-1150 study methodology, which fully considers population density and distribution, it is possible to compare offsite health effects to the QHOs of the NRC Safety Goal Policy. However, because population density guidelines would only limit the number of potentially affected individuals in the vicinity of the plant, it does not address the benefits of other influences pertinent to protection of public health and safety.

Thus, the population density criterion proposed for codification will not provide the flexibility needed to select consistently the most favorable sites. Alternative population configurations with larger overall populations may result in lower societal radiation risk yet fail the NRC proposed criterion. This may result in eliminating acceptable (possibly the more favorable) sites from further consideration while less favorable sites survive the site selection screening process. The proposed criterion could unnecessarily constrain future activities at existing sites (if they were to be evaluated under the proposed rule), sites that have previously been approved but have never had operating reactors, and new sites that would likely be considered for future reactors. For example, in the NRC Staff briefing to the Commission, the staff stated that eight licensed sites do not currently meet the five hundred persons per square mile population density criterion, and twenty-five licensed sites do not meet the 0.4 mile exclusion area distance criterion. All of these sites have been demonstrated to adequately protect the public health and safety and easily meet the NRC Safety Goal Policy.

If codified, then the restrictive numeric criteria would establish a de facto safety goal in regulation that is more stringent than the NRC Safety Goal Policy and is applicable to future plants only. The next generation of nuclear plants will have comprehensively considered 30 years of accumulated experience including improved understanding of postulated severe accident phenomena, probability, and consequences. Therefore, these plants will reflect in their design, construction, and operation; risk characteristics as low or lower than those for existing plants.

The proposed criteria would be significantly more restrictive than what is required to meet the QHOs of the Safety Goal Policy and would therefore be unnecessarily restrictive. Since the risk to the public would be so much less than the 0.1 % increase from all other factors established by the QHOs of the Safety Goal Policy, there is insufficient health risk basis for establishing such a restrictive criteria.

II. Objective

To evaluate population distributions adjacent to a nuclear power plant and exclusion zone size relative to offsite risks from postulated accidents and to compare these risks to the QHOs of the NRC Safety Goal Policy using the NUREG-1150 methodology.

III. Background

Existing NRC policy establishes the safety goals for individual risks in the vicinity of nuclear power plants. These goals define what the NRC considers as adequate safety and risk beyond which the risk is considered to be remote and speculative for new rulemaking. These safety goals state that the risk of prompt or latent (cancer) fatalities to an average individual in the vicinity of a nuclear power plant should not exceed 0.1% of the risk from other causes. For the prompt risk safety goal, the vicinity is defined as within one mile of the reactor, whereas for the latent risk safety goal the vicinity is defined as within a ten mile radius of the reactor. The NRC has defined QHOs corresponding to the above goals as equivalent to 5×10^{-7} and 2×10^{-6} fatalities per reactor-year for prompt and latent cancer fatalities, respectively.¹

10 CFR Part 100 specifies the following:

- An exclusion area
- A low population zone (LPZ)
- A population center distance criteria

The exclusion area is to be of sufficient size such that any individual in this area (but outside the plant protected area) would not receive, upon two hours of exposure following a postulated release, a total radiation dose to the whole body in excess of 25 rem or a total radiation dose to the thyroid in excess of 300 rem from iodine exposure.

The LPZ is defined such that any individual in the LPZ who is exposed to the entire passage of the radioactive cloud resulting from a postulated release would not receive a dosage in excess of the above limits.

The population center distance criteria states that no population center (defined as a densely populated area with a total population in excess of 25,000 people) shall be closer than 1 1/3 times the distance from the reactor to the outer boundary of the LPZ.

The intent of the LPZ is to ensure that the population is sufficiently limited and, furthermore, is distributed in such a fashion that mitigating action (such as an evacuation) would likely be successful. The exclusion area was established based on the premise that successful mitigation might not be achievable given excessive radiation. Finally, the population center distance criterion addresses societal risk (albeit indirectly) in that it has been deemed desirable to keep large populations relatively far from the reactor, thereby minimizing the potential for large scale consequences.

¹U.S. Nuclear Regulatory Commission, SECY-89-102, *Implementation of the Safety Goal Policy*, March 30, 1989, and Staff Requirements Memorandum to SECY-89-102, June 15, 1989.

The distances and areas specified above cannot be precisely determined due to variability of release patterns, meteorology, and location of the most exposed individual and are specific to each operating nuclear plant site. The NRC established population density guidelines in Regulatory Guide 4.7 issued in November, 1975. The NRC determined that an exclusion area radius of 0.4 miles would be acceptable for all reactors when the Regulatory Guide was published. In addition, the NRC determined that an acceptable surrounding population density should not exceed 500 people per square mile averaged over any radial distance out to 30 miles. Furthermore, the projected population for the operating lifetime of the reactor (40 years) should not exceed 1000 people per square mile. The probable intent of the above population density requirements was to ensure a sufficiently low population density to allow for effective mitigation measures, while assuring that population centers would not encroach on the reactor.²

The NRC is now proposing to codify the above guidelines. The stated objective for this change is to, "Relocate the requirements that apply to plant design into 10 CFR part 50 thereby effectively decoupling siting from plant design."³ Another projected benefit is that by establishing quantitative population rules, the siting requirements become clearly defined. This, in turn, is expected to reduce the time and costs associated with obtaining site approval.

IV. Methodology

The population density and exclusion zone size criteria in the proposed revisions to 10 CFR Part 100 were evaluated using the MELCOR Accident Consequence Code System (MACCS), developed by Sandia National Laboratories for the NRC. Surry Nuclear Plant radiological and meteorological input data were taken from the NUREG-1150 analysis. Surry was selected for the following reasons: it is one of the five plants evaluated by the NRC in NUREG-1150; it is one of two plants for which NUREG-1150 considered analysis of external events; and its source term data are readily available. Additional discussion of the data and assumptions utilized in the MACCS analysis is provided in Appendix A.

V. Results

This section presents the results of the MACCS analysis of the four release scenarios discussed in Appendix A applied to a uniform population density of 500 people per square mile and an exclusion zone size of 0.33 miles. While MACCS can be used to calculate and display a variety of quantitative information, individual risk probabilities (identical to NRC QHO units) and the population dose (assumed to be a measure of cumulative societal risk) were selected for comparison.

For the NUREG 1150 analysis, the Surry source terms were divided into 17 groups and further divided into three subgroups each for a total of 51 subgroups. For purposes of this analysis, these groups were condensed into four categories, and representative source terms were chosen for each category. Appendix A provides further detail on the assumptions used for this analysis.

Figures 1 and 2 present comparisons of the quantitative prompt and latent individual risks from this analysis with their respective QHOs. As can be seen in Figure 1, the prompt risk is well

²The draft document for Regulatory Guide 4.7 issued the previous year made no mention of these population density guidelines. From review of available public documents, it was determined that (1) if these guidelines, which were later added to Regulatory Guide 4.7, had been added without allowing for public discussion and (2) the justification for the selection of the 500/1000 population density criterion (not given in Regulatory Guide 4.7) could not be found.

³U. S. Nuclear Regulatory Commission, October 20, 1992 (57 FR 47803).

below the QHO, reaching its peak value of 8.4% of the QHO at the boundary of the exclusion area (assumed for this study to be 0.33 miles). The prompt risk is shown to drop rapidly, and the residual radiation risk is insignificant beyond five miles. The individual latent risk, shown in Figure 2, meets the QHO by even a wider margin with a maximum individual risk of about 1% of the QHO. As with the prompt risk, the individual latent risk initially declines rapidly with distance with the risk reduced to about 10% of its maximum value at ten miles out. The risk then continues to decrease with radial distance.

Using a weighted average of all four modeled scenarios, Figure 3 depicts a plot of the resulting dose as it varies with radial distance. This dose is comprised of two components: dose from immediate impact (e.g., cloudshine, groundshine, inhalation, and deposition on skin), and dose from long term impact (e.g., consumption of contaminated milk, water, or food, as well as, dose from decontamination related activities). As with the individual risk, the dose declines rapidly with distance and is reduced to less than 10% of its maximum value at ten miles.

Figure 4 shows the immediate, long term, and combined population dose (individual dose weighted with population) as a function of distance. The combined population dose is shown to decline only slightly with distance, indicating that the decrease in individual dose is balanced by the increase in population. Note, however, that the immediate population dose decreases with distance, sharply at first, then declining more gradually. In contrast, the long term cumulative population dose increases with distance for approximately the first 15 miles then becomes stable with distance.

The total societal risk to a population (i.e., the cumulative risk to all people in the 30-mile radius) using the algorithms in MACCS is determined by integrating the area under the combined population curve in Figure 4. This assumes that latent risk is proportional to cumulative dose regardless of how the dose is distributed to individuals (e.g., one person-rem divided among 1000 people is equivalent to one person-rem to a single person). Using the NUREG-1150 data for the Surry Plant and assuming a uniform population density of 500 people per square mile and exclusion zone size of 0.33 miles, the total dose is calculated to be about 13 person-rem per reactor-year. This, of course, is the value of an average exposure taken over millions of reactor-years as the population would only be exposed in the extremely unlikely chance of a core damage event. This value represents a valid quantitative measure of the societal risk.

Figure 5 presents the results of an analysis performed using two assumed population distributions. The first assumes a uniform population density of 500 people per square mile. The second assumes that most of the population is located between the 20 and 30 mile radial distances. This analysis shows a scenario in which the NRC population density criteria is exceeded, yet the total societal risk is significantly lower than a scenario which conforms to the population density criteria.

Figures 6 through 8 present the results of an analysis performed assuming a reduced exclusion zone of 0.2 miles from the reactor with and without evacuation. This analysis shows that, assuming evacuation, the individual risk for the 0.2 mile exclusion zone is slightly higher than the individual risk for the 0.33 mile exclusion zone.

The individual risk to the nearby population is greater if there is no evacuation, as noted in Figures 6 and 7, and reflected in the population dose profile (Figure 8). The evacuation was assumed to be carried out to ten miles only (the same assumption as NUREG-1150); therefore, the risk to the greater than ten mile population is identical for both evacuation assumptions.

VI. Conclusions

As demonstrated by the results presented above, it can be concluded that a reactor of current design located in an area of controlled population distribution and minimum exclusion zone size, as specified by the NRC in the proposed 10 CFR Part 100 revision, results in a calculated annual societal risk which is well within the QHO of the NRC Safety Goal Policy. Considering that such a facility meets the individual QHOs by a wide margin, the proposed NRC population density and exclusion zone size criteria are unnecessarily restrictive and therefore, should not be promulgated. In addition, codification of these criteria is questionable because new future advanced nuclear plant designs have added features to lower the risk when compared to the current generation of plants.

Further, the population density criterion proposed for codification will not provide the flexibility needed to select consistently the most favorable sites. Alternative population configurations with larger overall populations may result in lower societal radiation risk yet fail the NRC proposed criterion. This may result in eliminating acceptable (possibly the more favorable) sites from further consideration while less favorable sites survive the site selection screening process. The proposed criterion could unnecessarily constrain future activities at existing sites (if they were to be evaluated under the proposed rule), sites that have previously been approved but have never had operating reactors, and new sites that would likely be considered for future reactors. For example, in the NRC Staff briefing to the Commission, the staff stated that eight licensed sites do not currently meet the five hundred persons per square mile population density criterion, and twenty-five licensed sites do not meet the 0.4 mile exclusion area distance criterion. All of these sites have been demonstrated to adequately protect the public health and safety and easily meet the NRC Safety Goal Policy.

If codified, then the restrictive numeric criteria would establish a de facto safety goal in regulation that is more stringent than the NRC Safety Goal Policy and is applicable to future plants only. The next generation of nuclear plants will have comprehensively considered 30 years of accumulated experience including improved understanding of postulated severe accident phenomena, probability, and consequences. Therefore, these plants will reflect in their design, construction, and operation; risk characteristics as low or lower than those for existing plants. The proposed criteria would be significantly more restrictive than what is required to meet the QHOs of the Safety Goal Policy and would therefore be unnecessarily restrictive. Since the risk to the public would be so much less than the 0.1 % increase from all other factors established by the QHOs of the Safety Goal Policy, there is insufficient health risk basis for establishing such a restrictive criteria.

FIGURE 1 PROMPT INDIVIDUAL RISK

FIGURE 2 LATENT INDIVIDUAL RISK

FIGURE 3 INDIVIDUAL DOSE PROFILE

The modeled core damage accident is an amalgamation of a range of release scenarios.

FIGURE 4 POPULATION DOSE PROFILE

Cumulative population Dose Risk (within 30 miles) equivalent to approximately 13 man-rem per reactor year.

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FIGURE 5 POPULATION DOSE PROFILE FOR VARYING POP. PROFILE

--- Stratified Pop: 50 people/sq mile 0.3 - 10 miles; 250 people/sq mile 10 - 20 miles; 750 people/sq mile 20 - 30 miles; — Uniform Pop: 500 people/sq mile 0.3 - 30 miles.

Total exposure:

Uniform Population: 13 man-rem/reactor-year

Stratified Population: 8.5 man-rem/reactor-year

FIGURE 6 PROMPT INDIVIDUAL RISK

0.33 mile exclusion w/ evac.
 0.2 mile exclusion w/ evac.
 0.2 mile exclusion w/o evac.
 Safety Goal

FIGURE 7 LATENT INDIVIDUAL RISK

FIGURE 8 POPULATION DOSE PROFILE

Total dose with evacuation = 13.3 man-rem/year (for both 0.2 and 0.33 mile exclusion zones).

Total dose without evacuation = 14.5 man-rem/year (0.2 mile exclusion zone).

APPENDIX A

Data and Assumptions for MACCS Analysis

In the NUREG-1150 analysis of the Surry Nuclear Plant, the thousands of source terms were partitioned into 17 groups. Each of these groups were subdivided into three subgroups:

Subgroup 1: Evacuation starts at least 30 minutes before the release begins.

Subgroup 2: Evacuation starts between 30 minutes before and 1 hour after the release begins.

Subgroup 3: Evacuation starts more than 1 hour after the release begins.

While this gives a potential for 51 subgroups, some of these subgroups have zero or nearly zero conditional probability. Nevertheless, this leaves 34 non-trivial subgroups for MACCS input. Rather than evaluating all of these groups, they were combined to a more manageable number of events while retaining the integrity of the analysis. Figure A-1 depicts the partitioning of the 17 groups by early and latent health effects as evaluated in the NUREG-1150 study. These groups were then further organized into four categories as shown. NUREG-1150 lists the expected annual frequencies of each of the 17 source terms, the conditional probability of each of their subgroups, and the predicted consequences of each of their occurrence. Table A-1 presents this data condensed into the four categories used for this study, along with the source terms selected to represent each of these categories. With the exception of source term 17-1, each of the selected postulated releases closely matches the expected impact of their respective categories. Source term 17-1 was conservatively selected to represent its category as it was the only source term in this group with a non-negligible impact.

Once the representative release scenarios were selected, it was necessary to calibrate their frequencies to match the calculated Surry impact given in NUREG-1150. The four representative release scenarios were input to MACCS with parameters set identically to those used in the NUREG-1150 study. The NUREG-1150 analysis predicted a mean probability per reactor-year of 2.0×10^{-6} and 5.23×10^{-3} for prompt and latent fatalities, respectively. As shown in Table 1, the use of the four input scenarios results in a good correlation to the NUREG-1150 calculated prompt fatality rate but over-predicts the latent fatality rate. This was accounted for through a reduction factor on the annual frequency of scenario 17-1. Reducing the frequency of this scenario to 1.4×10^{-5} per year results in a latent fatality rate matching that given in NUREG-1150 and still retains the balance of the source term groupings.

FIGURE A-1

Distribution of Source Terms with Nonzero Early Fatality

Distribution of Source Terms with Zero Early Fatality

Source Groups:

Representative Source Term:

I		11-3
II		7-3
III		3-3
IV		17-1

FIGURE A-2
Selection Of Category Scenarios

Category	NUREG-1150 Groups	Representative Source Term	Combined Frequency	Expected Annual Risk	
				Prompt	Latent
I	10,11,5	11-3	3.24E-07	1.31E-06	9.46E-04
II	6,7,8,12,13,14	7-3	9.34E-07	4.70E-07	1.31E-03
III	1,2,3,4,9	3-3	1.72E-06	2.70E-07	1.37E-03
IV	15,16,17	17-1	3.74E-05	0	4.23E-03

Total Predicted Annual Risk:	2.05E-06	7.85E-03
NUREG-1150 Results:	2.00E-06	5.23E-03

TABLE A-1

OPERATING UNIT SOURCE TERMS EVALUATED WITH UNIFORM POPULATION (500 PEOPLE/SQ MILE)

OVERALL WEIGHTED ANALYSIS

Population Distribution		Pop Centroid (mi)	Individual Fatality Rate (Fatality per Rx-Yr)		% of Safety Goal		Annual Population Dose (Rem per Rx-Yr)			Pop. Dose/Mile (Rem per Rx-Yr)			Individual Dose (Rem per Rx-Yr)		
Inside radius (km)	Outside radius (km)		Early	Latent	Early	Latent	Early	Chronic	Combined	Early	Chronic	Combined	Early	Chronic	Combined
0.5	1.2	0.59	4.24E-08	2.28E-08	8.48	1.14	0.16	0.03	0.19	0.38	0.06	0.44	5.77	0.97	6.74
1.2	1.6	0.89	2.25E-08		4.51		0.10	0.02	0.12	0.40	0.07	0.47	3.69	0.68	4.37
1.6	2.1	1.18	1.11E-08	1.91E-08	2.22	0.95									
2.1	3.2	1.73	4.86E-09	1.43E-08	0.97	0.72	0.22	0.08	0.30	0.33	0.12	0.45	1.61	0.58	2.19
3.2	4	2.28	3.13E-09	1.27E-08	0.63	0.64	0.16	0.09	0.25	0.33	0.18	0.50	0.82	0.43	1.25
4.8	5.6	3.25	5.14E-10	9.58E-09	0.10	0.48	0.43	0.35	0.79	0.22	0.18	0.40	0.29	0.24	0.54
5.6	8.1	4.43	3.82E-11	6.43E-09	0.01	0.32	0.20								
8.1	11.3	6.24	5.09E-13	3.91E-09	0.00	0.12									
11.3	16.1	8.85	1.75E-14	2.40E-09	0.00										
16.1	20.9	11.77													
25.8	32.2	18.34													
32.2	40.2	22.90													
40.2	41	24.85													
48	49	29.69													
				8.89E-10		0.04	0.04	0.16	0.20	0.09	0.32	0.40	0.03	0.10	0.13
				6.11E-10		0.03	0.05	0.19	0.24	0.07	0.31	0.39	0.02	0.08	0.10

Note: Table blanks indicate that no data was generated for that distance interval and effect

TABLE A-2

**OPERATING UNIT SOURCE TERMS EVALUATED WITH UNIFORM POPULATION
(500 PEOPLE/SQ MILE) AND STRATIFIED POPULATION**

Population Distribution			Combined Source Terms					
Inside radius (km)	Outside radius (km)	Pop Centroid (mi)	Avg. Miles	Stratified Pop	Size, Area	Unified Pop	Unit, Area	
0.5	1.2	0.59	0.59	0.05	0.01	0.01	0.21	
1.2	1.6	0.89	0.89	0.05	0.01	0.01	0.47	
1.6	2.1	1.18	1.18	0.05				
2.1	3.2	1.73	1.73	0.06	0.04	0.04	0.45	0.38
3.2	4	2.28	2.28	0.06	0.03			
4	4.8	2.77	2.77	0.06				
4.8	5.6	3.25	3.25	0.05	0.05	0.05	0.50	0.73
5.6	8.1	4.43	4.43	0.05	0.06			
8.1	11.3	6.24	6.24	0.05	0.08	0.08	0.40	1.35
11.3	16.1	8.85	8.85	0.05	0.12	0.12		
16.1	20.9	11.77	11.77	0.25	0.43	0.43	0.49	2.48
20.9	25.8	14.74	14.74	0.24	0.72	0.72		
25.8	32.2	18.34	18.34	0.24	0.86	0.86	0.48	3.19
32.2	40.2	22.90	22.90	0.65	2.02	2.02		
40.2	41	24.85	24.85	0.60	1.21	1.21	0.40	2.87
41	42	25.42	25.42	0.62	0.35	0.35		
42	43	26.03	26.03	0.60	0.37	0.37		
44	45	27.25	27.25	0.59	0.72	0.72		
46	47	28.47	28.47	0.58	0.71	0.71		
48	49	29.69	29.69	0.57	0.71	0.71	0.39	1.92

Total Dose (man rem per year) **8.54** **13.13**

Note: Table blanks indicate that no data was generated for that distance and effect

TABLE A-3

0.33 MILE EXCLUSION ZONE WITH EVACUATION

OVERALL WEIGHTED ANALYSIS

Population Distribution			Individual Fatality Rate		% of Safety Goal		Annual Pop. Dose	Pop. Dose/Mile		Individual Dose
Inside radius (km)	Outside radius (km)	Pop. Centroid (mi)	Fatality per Rx-Yr Early	Fatality per Rx-Yr Latent	Early	Latent	(Rem per Rx-Yr) Combined	(Rem per Rx-Yr) Combined	Area	(Rem per Rx-Yr) Combined
0.5	1.2	0.59	4.24E-08	2.28E-08	8.48	1.14	0.19	0.44	0.26	6.74
1.2	1.6	0.89	2.25E-08		4.51	0.00	0.12	0.47	0.14	4.37
1.6	2.1	1.18	1.11E-08	1.91E-08	2.22	0.95				
2.1	3.2	1.73	4.86E-09	1.43E-08	0.97	0.72	0.30	0.45	0.38	2.19
3.2	4	2.28	3.13E-09	1.27E-08	0.63	0.64				
4.8	5.6	3.25	5.14E-10	9.58E-09	0.10	0.48	0.25	0.50	0.73	1.25
5.6	8.1	4.43	3.82E-11	6.43E-09	0.01	0.32				
8.1	11.3	6.24	5.09E-13	3.91E-09	0.00	0.20	0.79	0.40	1.35	0.54
11.3	16.1	8.85	1.75E-14	2.40E-09	0.00	0.12				
16.1	20.9	11.77				0.00	1.44	0.49	2.47	0.34
25.8	32.2	18.34					1.87	0.48	3.19	0.21
32.2	40.2	22.90		8.89E-10		0.04				
40.2	41	24.85					0.20	0.40	2.87	0.13
48	49	29.69		6.11E-10		0.03	0.24	0.39	1.92	0.10

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Note: Table blanks indicate that no data was generated for that distance interval and effect

TABLE A-4

0.2 MILE EXCLUSION ZONE WITH EVACUATION

OVERALL WEIGHTED ANALYSIS

Population Distribution			Individual Fatality Rate		% of Safety Goal		Annular Population	Pop. Dose/Mile		Individual Dose
Inside radius	Outside radius	Pop Centroid	(Fatality per Rx-Yr)		(Rem per Rx-Yr)		(Rem per Rx-Yr)	(Rem per Rx-Yr)		(Rem per Rx-Yr)
(km)	(km)	(mi)	Early	Latent	Early	Latent	Combined	Combined	Area	Combined
0.3	1.2	0.55	4.87E-08	2.33E-08	9.74	1.17	0.24	0.44	0.24	7.54
1.2	1.6	0.89	2.26E-08		4.51		0.12	0.47	0.16	4.37
1.6	2.1	1.18	1.11E-08	1.91E-08	2.22	0.95				
2.1	3.2	1.73	4.86E-09	1.43E-08	0.97	0.72	0.30	0.45	0.38	2.19
3.2	4	2.28	3.13E-09	1.28E-08	0.63	0.64				
4.8	5.6	3.25	5.14E-10	9.58E-09	0.10	0.48	0.25	0.50	0.73	1.25
5.6	8.1	4.43	3.82E-11	6.43E-09	0.01	0.32				
8.1	11.3	6.24	5.09E-13	3.91E-09	0.00	0.20	0.79	0.40	1.35	0.54
11.3	16.1	8.85	1.75E-14	2.40E-09	0.00	0.12				
16.1	20.9	11.77					1.44	0.49	2.48	0.34
25.8	32.2	18.34					1.87	0.48	3.19	0.21
32.2	40.2	22.90		9.49E-10		0.05				
40.2	41	24.85					0.20	0.40	2.87	0.13
48	49	29.69		6.40E-10		0.03	0.24	0.39	1.91	0.10

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Note: Table blanks indicate that no data was generated for that distance interval and effect

TABLE A-5

0.2 MILE EXCLUSION ZONE WITHOUT EVACUATION

OVERALL WEIGHTED ANALYSIS

Population Distribution			Individual Fatality Rate		% of Safety Goal		Annular Population	Pop. Dose/Mile	Individual Dose	
Inside radius	Outside radius	Pop Control	(Fatality per Rx-Yr)				(Rem per Rx-Yr)	(Rem per Rx-Yr)	(Rem per Rx-Yr)	
(km)	(km)	(mi)	Early	Latent	Early	Latent	Combined	Combined	Area	Combined
0.3	0.5	0.26	1.67E-07	1.17E-07	33.35	5.85	0.12	0.98	0.26	31.58
0.5	1.2	0.59					0.19	0.76	0.68	7.00
1.2	1.6	0.89	3.93E-08		7.86					
1.6	2.1	1.18	2.51E-08	3.31E-08	5.02	1.66				
2.1	3.2	1.73	1.23E-08		2.46		0.43	0.64	0.58	3.14
3.2	4	2.28	6.04E-09	1.62E-08	1.21	0.81				
4.8	5.6	3.25	1.45E-09	1.01E-08	0.29	0.50	0.28	0.57	0.93	1.42
5.6	8.1	4.43	4.45E-10	6.59E-09	0.09	0.33			0.34	
8.1	11.3	6.24	1.86E-11	3.86E-09	0.00	0.19	0.87	0.45		0.60
11.3	16.1	8.85		2.84E-09		0.14				
16.1	20.9	11.77					1.44	0.49	0.72	0.34
25.8	32.2	18.34					1.87	0.48	3.18	0.21
32.2	40.2	22.90		9.49E-10		0.05			1.09	
40.2	41	24.85					0.20	0.40		0.13
48	49	29.69		6.40E-10		0.03	0.24	0.39	1.32	0.10

Note: Table blanks indicate that no data was generated for that distance interval and effect

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March 1993

**EVALUATION OF OFFSITE RISK FROM NORMAL
NUCLEAR POWER PLANT OPERATIONS**

EARLY SITE PERMIT DEMONSTRATION PROGRAM

MARCH 1993

March 1993

EVALUATION OF OFFSITE RISK FROM NORMAL NUCLEAR REACTOR OPERATIONS

I. Summary

In this evaluation, the performance of several of the most recently licensed light-water reactors was examined with regard to the conformance with the U.S. Nuclear Regulatory Commission (NRC) safety goal for risk to the general public in the vicinity of these plants. The NRC safety goal was translated into an annual dose level of approximately 6.7 mrem/yr, assuming a 60 year exposure to plant operation. Licensee generated data for offsite dose from routine operation as documented in Semi-Annual Effluent Reports was obtained for a sample of recently licensed light-water reactors for the period 1987 - 1991. This data was compiled, and average offsite dose impacts from liquid and atmospheric releases calculated for all evaluated plants. From this data it was concluded that routine operation of these plants is well within the offsite risk limits defined in the NRC safety goal and, by inference, that routine operation of advanced light water reactors (ALWRs) will also be well within the safety goal risk limits. An evaluation of the offsite risks from postulated accidents at nuclear power plants is contained in the companion report, *Evaluation Of The Effects Of Population Distribution On The Ability To Meet NRC Safety Goals For Offsite Risk*.

II. Objective

To determine and compare with the NRC safety goal, the risk to the population in the area near existing nuclear power plants so as to bound the expected risk from an ALWR from routine plant operation.

III. Background

The NRC policy statement, *Safety Goals for the Operations of Nuclear Power Plants*,¹ defines the safety goal from plant operations as "...one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes." An annual dose level, for constant exposure over 60 years of plant operation, equivalent to this goal is calculated as approximately 6.7 mrem/yr (see Appendix B).

In the policy statement, the Commission defines the population considered subject to significant risk as the population within 10 miles of the plant site. For reporting purposes, NRC reactor licensees calculate population doses out to 50 miles but do not report incremental doses to only the 10 mile population. Thus, the dose to the hypothetical maximum exposed individual (MEI) at each of the evaluated plants was used as a conservative (by at least one order of magnitude) surrogate for the average dose to individuals within 10 miles. Doses to this individual serve as a bounding condition for determinations of latent cancer risks.

IV. Methodology

The methodology used in this evaluation is to obtain, chart, and trend for each of the evaluated plants the maximum offsite dose impact from airborne and liquid releases from a sample of recently licensed Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). This data demonstrates that the impact from routine operation of these plants has resulted in offsite doses much less than the dose corresponding to the NRC safety goal (6.7 mrem/yr).

¹U. S. Nuclear Regulatory Commission, November 30, 1988.

The method for selection of the subject plants was to include the last twelve licensed PWRs and ten BWRs as documented in *Commercial Nuclear Power Plants*, Edition No. 19.² The only exception was Comanche Peak-1 which began commercial operation on August 13, 1990, and for which there are insufficient years of operation to provide substantial data. The selection of twelve PWRs was made to provide a data set containing ten PWRs having a licensed power of over 3000 MWt, without excluding two newer plants having lower licensed power levels. For multi-unit sites the oldest unit was chosen to provide a more complete data set. The exceptions to this rule were: (1) Beaver Valley, where unit 1 is significantly (approx. 11 years) older than unit 2; (2) Millstone, where units 1 and 2 are significantly older than unit 3 and of different design; and (3) Nine Mile Point, where unit 1 is significantly (approx. 19 years) older than unit 2. The selected plants are listed in Appendix C along with their licensed power and the date of assumption of commercial operations.

The analysis used the previous five years (1987 - 1991) Semi-Annual Effluent Reports for the subject plants. These reports are prepared in accordance with NRC Regulatory Guide 1.21 and present doses to the MEI to demonstrate compliance with plant Radiological Effluent Technical Specifications. The offsite dose values were extracted from these reports, analyzed, charted, and trended. The following values were included in the evaluation: (1) total body dose from liquid effluents; (2) total body dose from airborne emissions of iodines, particulates, and tritium; and (3) gamma "airdose" from airborne emissions of noble gases.

A listing of the offsite dose commitments for the period of 1987-1991 for each selected plant is provided in Appendix D. There is currently no NRC-mandated reporting format for offsite dose values; each utility reports these numbers differently. With these reporting disparities in mind, the following assumptions were made in extracting these values from the Semi-Annual Effluent Reports:

- For the following sites, the offsite dose impact was calculated on a site-wide, rather than a per unit basis. For these sites, the following techniques were used to derive an approximation of the dose impact of the selected unit.
 - Palo Verde: All dose values were presented on a site-wide (three unit) basis. Since all units are identical, the dose impacts were divided by three to determine an approximate Palo Verde-1 contribution. The exception to this was 1987, where the total was divided by two because Palo Verde-3 did not achieve initial criticality until October 1987.
 - Byron: All dose values for 1987 and 1988 were presented on a site-wide (two unit) basis. Since both units are identical, the dose impacts for these years were divided by two to determine an approximate Byron - 1 contribution.
 - Nine Mile Point: All dose values were presented for both Nine Mile Point units and the James A. Fitzpatrick plant combined. For these sites, the total dose was divided by three, which should yield a conservative value because Nine Mile Point 2 is significantly newer than the other units and because of new effluent control technologies, its dose impact is probably less than one third of the total value.
 - Susquehanna: All dose values were presented on a site-wide (two unit) basis. Since the units are identical, the dose impacts were divided by two to determine an approximate Susquehanna-1 contribution.

²HALLIBURTON NUS Environmental Corporation, December 1991.

- LaSalle: All gaseous dose values were presented on a site-wide (two unit) basis, while liquid dose impacts are presented on a per unit basis. Since the units are identical, the dose impacts of gaseous releases were divided by two to determine an approximate LaSalle-1 contribution.
- For plants that only reported the critical organ dose for airborne releases of iodines, particulates, and tritium and not the total body dose, the critical organ dose was used as a surrogate value for total body dose. This should lead to a conservative result as the critical organ dose should be greater than or equal to the total body dose.
- When extracting the total body dose from airborne releases for plants that did not distinguish between noble gas releases and releases of iodines, particulates, and tritium, the listed values were considered to be from releases of iodines, particulates, and tritium. This should lead to a conservative result because the listed value will be greater than or equal to the dose from releases of iodines, particulates, and tritium.

If a given parameter was not reported in the Semi-Annual Effluent Report, the parameter is listed in Appendix D as N/R. If a given parameter was not applicable to the selected plant (i.e., the time period was prior to initial criticality or the effluent pathway does not exist), the parameter is listed in Appendix D as N/A.

V. Results

For the data presented in Appendix D, the mean and standard deviation of each dose parameter, were calculated (for each reactor type) for each year (see Appendix E) and a summary of the results is presented below.

PWRs

Year	Average Total Body Dose, Liquid Releases (mrem)	Average Total Body Dose, Particulate, Iodine, and Tritium Releases (mrem)	Gamma Airdose (mrad)
1987	(1.25 ± 1.54) E-01	(1.58 ± 2.23) E-01	(4.57 ± 4.65) E-01
1988	(1.30 ± 1.85) E-01	(6.18 ± 6.85) E-02	(2.29 ± 3.01) E-01
1989	(1.39 ± 1.78) E-01	(1.15 ± 2.33) E-01	(6.82 ± 10.9) E-02
1990	(2.29 ± 3.13) E-01	(7.58 ± 15.5) E-02	(1.85 ± 3.59) E-01
1991	(2.49 ± 3.58) E-01	(8.56 ± 11.8) E-02	(9.45 ± 14.1) E-02
Average	(1.74 ± 1.12) E-01	(9.92 ± 7.66) E-02	(2.07 ± 1.37) E-01

BWRs

Year	Average Total Body Dose, Liquid Releases (mrem)	Average Total Body Dose Particulate, Iodine, and Tritium Releases (mrem)	Gamma Airdose (mrad)
1987	(1.47 ± 2.48) E-01	(2.14 ± 4.48) E-01	(8.07 ± 12.2) E-02
1988	(7.89 ± 18.1) E-02	(7.96 ± 20.4) E-01	(4.46 ± 10.3) E-01
1989	(1.91 ± 3.53) E-01	(4.55 ± 6.66) E-02	(1.54 ± 2.32) E-01
1990	(9.57 ± 21.1) E-02	(6.06 ± 14.2) E-01	(7.57 ± 16.2) E-02
1991	(8.24 ± 18.0) E-02	(3.33 ± 6.66) E-01	(4.57 ± 5.94) E-02
Average	(1.19 ± 1.09) E-01	(3.99 ± 5.23) E-01	(1.60 ± 2.15) E-01

Chronological offsite dose impacts are graphically represented in Figure II-1 through Figure II-6. For each year, the average dose value is plotted along with an error bar representing two standard deviations above and below the average. This representation approximates a 95 percent confidence interval for each result. The 95 percent confidence interval (± 2 standard deviations) for the mean value over the entire time period (final row in the above tables) is presented below.

Type	Average Total Body Dose, Liquid Releases (mrem)	Average Total Body Dose Particulate, Iodine, and Tritium Releases (mrem)	Gamma Airdose (mrad)
PWR	(1.74 ± 2.24) E-01	(9.92 ± 15.32) E-02	(2.07 ± 2.74) E-01
BWR	(1.19 ± 2.18) E-01	(3.99 ± 10.46) E-01	(1.60 ± 4.30) E-01

Even if the three dose values were added (which is not a realistic assumption because these doses are calculated for different critical receptor locations, and it is usually not possible to receive the maximum dose from all pathways), the results would be 0.48 mrem/yr for PWRs and 0.68 mrem/yr for BWRs. This cumulative value, at a 95 percent confidence level, is well below the NRC safety goal of 6.7 mrem/yr.

VI. Conclusions

As demonstrated by the results presented above, it can be concluded with confidence that routine operation of current light-water reactors is well within the NRC safety goal risk limits to the population residing in the vicinity of these plants. This conclusion was determined by analyzing the routine operating history of recently licensed commercial light-water reactors. From this result, it can be further concluded that operation of ALWRs, because of the use of advanced effluent control and treatment technologies which are expected to be incorporated in the design and development of ALWRs, will also satisfy the NRC safety goal.

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Figure 1

March 1993

Figure 2

March 1993

Figure 3

March 1993

Figure 4

March 1993

Figure 5

Figure 6

APPENDIX A

Calculation Of Annual Dose Level Equivalent To The NRC Safety Goal

International Commission on Radiological Protection Publication 60, *1990 Recommendations of the International Commission on Radiological Protection*, presents revised risk coefficients for fatal cancer from exposure to ionizing radiation. Table S-3 lists the risk coefficient for the whole population as $5 \times 10^{-2} \text{ Sv}^{-1}$. This is equivalent to $5 \times 10^{-4} \text{ rem}^{-1}$. This is assumed to be a cumulative lifetime exposure.

Assuming a normal fatal cancer incidence of 20% (BIER V), the safety goal will correspond to 0.1 % (10^{-3}) of this:

$$\frac{(0.2)(10^{-3})}{5 \times 10^{-4} \text{ rem}^{-1}} = 0.4 \text{ rem} = 400 \text{ mrem (lifetime)}$$

Assuming a nuclear plant life of 60 years, this equals $(400 \text{ mrem})/(60 \text{ yr}) = 6.7 \text{ mrem/yr}$.

APPENDIX B

Reactors Chosen for Evaluation of Normal Operational Dose

<u>Plant</u>	<u>Type</u>	<u>Licensed Power (MWt)</u>	<u>Commercial Operation</u>
Beaver Valley 2	PWR	2660	11/17/87
Braidwood 1	PWR	3411	07/29/88
Byron 1	PWR	3411	09/16/85
Callaway	PWR	3565	12/19/84
Catawba 1	PWR	3411	06/29/85
Harris 1	PWR	2775	05/02/87
Millstone 3	PWR	3411	04/23/86
Palo Verde 1	PWR	3800	01/28/86
South Texas Project 1	PWR	3800	08/25/88
Vogtle 1	PWR	3411	06/01/87
Waterford 3	PWR	3390	09/24/85
Wolf Creek	PWR	3411	09/03/85
Clinton	BWR	2894	11/24/87
Fermi 2	BWR	3292	01/23/88
Hope Creek	BWR	3293	12/20/86
LaSalle 1	BWR	3323	01/01/84
Limerick 1	BWR	3293	02/01/86
Nine Mile Point 2	BWR	3323	03/11/88
Perry 1	BWR	3579	11/18/87
River Bend	BWR	2894	06/16/86
Susquehanna 1	BWR	3293	06/08/83
WNP 2	BWR	3323	12/13/84

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APPENDIX C

**Calculated Offsite Annual Dose
To Maximum Exposed Individuals
From Normal Operations
(1987-1991)**

March 1993

PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Duver Valley 2 <u>TYPE</u> PWR <u>LICENSED POWER</u> 2660 <u>COMMERCIAL OPERATION</u> 11/17/87	1987	N/A	N/A	N/A
	1988	6.27E-02	1.71E-02	7.62E-03
	1989	2.04E-02	7.36E-02	9.46E-03
	1990	3.67E-02	9.27E-02	1.39E-01
	1991	1.69E-02	2.25E-01	1.32E-02

Braidwood 1 <u>TYPE</u> PWR <u>LICENSED POWER</u> 3411 <u>COMMERCIAL OPERATION</u> 05/29/88	1987	N/R	N/R	N/R
	1988	1.59E-02	3.00E-03	0.00E+00
	1989	5.63E-02	3.00E-03	3.30E-02
	1990	N/R (data not available)	3.18E-03	1.18E-02
	1991	2.92E-03	1.96E-02	7.43E-02

Byron 1 <u>TYPE</u> PWR <u>LICENSED POWER</u> 3411 <u>COMMERCIAL OPERATION</u> 09/16/85	1987	7.65E-02	6.15E-02	4.00E-03
	1988	9.57E-02	8.25E-02	4.00E-03
	1989	2.38E-02	2.71E-04	1.06E-03
	1990	1.15E-02	7.15E-04	1.56E-03
	1991	N/R (data not available)	N/R (data not available)	N/R (data not available)

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PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Claway TYPE PWR LICENSED POWER 3565 COMMERCIAL OPERATION 12/19/84	1987	1.7E-01	6.4E-03	4.0E-02
	1988	1.19E-01	4.07E-03	7.98E-03
	1989	3.38E-03	1.09E-03	1.10E-02
	1990	5.72E-03	1.19E-03	1.27E-02
	1991	4.35E-03	1.24E-03	4.21E-03

Catawba 1 TYPE PWR LICENSED POWER 3411 COMMERCIAL OPERATION 06/29/85	1987	4.56E-01	3.33E-01 (thyroid)	7.54E-01
	1988	5.46E-01	1.42E-01 (thyroid)	6.33E-01
	1989	2.67E-01	1.90E-01 (thyroid)	3.80E-01
	1990	5.17E-01	5.51E-01 (thyroid)	8.86E-01
	1991	3.49E-01	3.26E-01 (thyroid)	4.27E-01

Harris 1 TYPE PWR LICENSED POWER 2775 COMMERCIAL OPERATION 05/02/87	1987	4.98E-03	2.16E-02	3.45E-02
	1988	4.66E-03	9.49E-02	1.48E-01
	1989	2.71E-01	3.79E-02	5.93E-02
	1990	9.59E-01	2.04E-02	3.11E-02
	1991	4.45E-02	1.56E-02	2.49E-02

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2 of 2

PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Unit 1 TYPE PWR LICENSED POWER 3411 COMMERCIAL OPERATION 06/01/87	1987	9.78E-03	9.83E-03	1.02E+00
	1988	1.74E-02	2.78E-03	7.13E-01
	1989	2.93E-02	3.71E-03	1.35E-03
	1990	1.78E-02	8.65E-04	1.06E-03
	1991	4.18E-02	1.96E-04	1.65E-03

Waterford 3 TYPE PWR LICENSED POWER 3390 COMMERCIAL OPERATION 02/24/85	1987	4.89E-02	6.61E-01	1.02E+00
	1988	4.37E-01	1.79E-01	7.55E-01
	1989	5.86E-01	8.90E-02	1.07E-01
	1990	3.39E-01	1.30E-01	1.01E-00
	1991	7.27E-01	2.40E-01	3.06E-01

Wolf Creek TYPE PWR LICENSED POWER 3411 COMMERCIAL OPERATION 09/03/85	1987	2.11E-01	5.37E-02	N/R
	1988	9.77E-02	1.04E-01	2.33E-02
	1989	1.77E-01	8.75E-02	N/R
	1990	3.10E-01	1.44E-02	2.78E-02
	1991	1.04E+00	1.31E-02	7.54E-02

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PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Anton <u>TYPE</u> BWR <u>LICENSED POWER</u> 2894 <u>COMMERCIAL OPERATION</u> 11/24/87	1987	N/R	N/R	N/R
	1988	1.01E-04	1.22E-02	1.23E-04
	1989	1.46E-03	2.04E-03	6.33E-04
	1990	2.96E-03	1.11E-04	6.88E-04
	1991	9.57E-03	1.59E-04	2.40E-05

Fermi 2 <u>TYPE</u> BWR <u>LICENSED POWER</u> 3292 <u>COMMERCIAL OPERATION</u> 01/23/88	1987	1.48E-03	0.00E+00	0.00E+00
	1988	4.15E-03	2.15E-04	1.79E-04
	1989	3.33E-02	6.26E-04	1.36E-01
	1990	5.13E-03	2.04E-04	4.58E-02
	1991	1.63E-02	3.31E-04	N/R

Hope Creek <u>TYPE</u> BWR <u>LICENSED POWER</u> 3293 <u>COMMERCIAL OPERATION</u> 12/20/86	1987	4.80E-01	1.31E+00 (thyroid)	N/R
	1988	5.39E-01	1.22E-02 (thyroid)	N/R
	1989	8.47E-01	9.36E-06 (thyroid)	N/R
	1990	6.87E-01	1.27E-02	N/R
	1991	1.36E-01	5.77E-05	N/R

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PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Salle 1 <u>TYPE</u> BWR <u>LICENSED POWER</u> 3323 <u>COMMERCIAL OPERATION</u> 01/01/84	1987	N/R	N/R	N/R
	1988	N/R	N/R	N/R
	1989	6.17E-05	7.45E-03	1.29E-02
	1990	1.81E-05	1.07E-04	2.04E-04
	1991	0.00E+00	1.15E-03	4.97E-03

Limerick 1 <u>TYPE</u> BWR <u>LICENSED POWER</u> 3293 <u>COMMERCIAL OPERATION</u> 08/01/86	1987	6.06E-01	2.06E-01 (bone)	1.14E-03
	1988	0.00E+00	1.73E-01 (thyroid)	5.70E-03
	1989	1.01E-01	7.07E-02 (skin)	5.70E-03
	1990	6.77E-02	4.50E-03 (thyroid)	1.31E-03
	1991	2.18E-02	9.68E-07 (GI)	2.33E-03

Nine Mile Point 2 <u>TYPE</u> BWR <u>LICENSED POWER</u> 3323 <u>COMMERCIAL OPERATION</u> 03/11/88	1987	0.00E+00	8.00E-03	N/R
	1988	3.33E-04	4.33E-03	N/R
	1989	3.47E-03	1.20E-02	N/R
	1990	3.73E-03	5.00E-03	N/R
	1991	1.52E-04	8.57E-03	N/R

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PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
Try 1 TYPE BWR LICENSED POWER 3579 COMMERCIAL OPERATION 11/18/87	1987	2.06E-04	2.60E-02 (thyroid)	1.17E-01
	1988	1.65E-03	6.19E+00 (thyroid)	2.55E+00
	1989	5.18E-02	1.79E-01 (thyroid)	6.13E-01
	1990	9.64E-02	1.25E+00 (thyroid)	3.65E-02
	1991	5.63E-02	1.84E+00 (thyroid)	7.87E-02

River Bend TYPE BWR LICENSED POWER 2894 COMMERCIAL OPERATION 06/16/86	1987	7.00E-02	4.80E-03	N/R
	1988	1.62E-01	3.30E-03	N/R
	1989	8.70E-01	5.00E-03	N/R
	1990	9.00E-02	4.50E+00 (thyroid)	N/R
	1991	5.80E-01	1.30E+00 (thyroid)	N/R

Susquehanna 1 TYPE BWR LICENSED POWER 3293 COMMERCIAL OPERATION 06/08/83	1987	4.02E-04	4.88E-02	5.60E-03
	1988	4.00E-04	1.84E-02	2.14E-03
	1989	2.91E-04	2.80E-02	2.82E-03
	1990	2.19E-04	7.50E-02	5.25E-03
	1991	3.91E-04	4.77E-02 (lung)	3.80E-02

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PLANT	YEAR	TOTAL BODY LIQUID DOSE (MREM)	TOTAL BODY PARTICULATE, I, AND H-3 DOSE (MREM)	GAMMA AIRDOSE (MRAD)
NP 2 TYPE BWR LICENSED POWER 3323 COMMERCIAL OPERATION 12/13/84	1987	1.7E-02	1.1E-01	2.8E-01
	1988	2.1E-03	7.5E-01	1.2E-01
	1989	1.8E-03	1.5E-01	3.2E-01
	1990	3.5E-03	2.1E-01	4.4E-01
	1991	5.2E-03	1.3E-01	1.5E-01

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APPENDIX D

Statistical Analysis of Operating Data

March 1993

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Plant	Type	Licensed Power (MWt)	Commercial Operation	Total Body Liquid Dose (mrem)				
				1987	1988	1989	1990	1991
Beaver Valley 2	PWR	2660	11/17/87	N/A	6.27E-02	2.04E-02	3.67E-02	1.69E-02
Braidwood 1	PWR	3411	7/29/88	N/A	1.59E-02	5.63E-02	N/R	2.92E-02
Byron 1	PWR	3411	9/16/85	7.65E-02	9.57E-02	2.38E-02	1.15E-02	N/R
Callaway	PWR	3565	12/19/84	1.70E-01	1.19E-01	3.38E-03	5.72E-03	4.35E-03
Catawba 1	PWR	3411	6/29/85	4.56E-01	5.46E-01	2.67E-01	5.17E-01	3.49E-01
Harris 1	PWR	2775	5/2/87	4.98E-03	4.66E-03	2.71E-01	9.59E-01	4.45E-02
Millstone 3	PWR	3411	4/23/86	1.90E-02	1.62E-02	5.83E-02	2.57E-02	4.37E-02
Palo Verde 1	PWR	3800	1/28/86	N/A	N/A	N/A	N/A	N/A
South Texas Project 1	PWR	3800	8/25/88	N/A	1.40E-02	4.06E-02	7.04E-02	1.94E-01
Vogtle 1	PWR	3411	6/1/87	9.78E-03	1.74E-02	2.93E-02	1.78E-02	4.18E-02
Waterford 3	PWR	3390	9/24/85	4.89E-02	4.37E-01	5.86E-01	3.39E-01	7.27E-01
Wolf Creek	PWR	3411	9/3/85	2.11E-01	9.77E-02	1.77E-01	3.10E-01	1.04E+00
Average				1.25E-01	1.30E-01	1.39E-01	2.29E-01	2.49E-01
Standard Deviation				1.54E-01	1.85E-01	1.78E-01	3.13E-01	3.58E-01
Clinton	BWR	2894	11/24/87	N/A	1.01E-04	1.46E-03	2.96E-03	9.57E-03
Fermi 2	BWR	3292	1/23/88	1.48E-03	4.15E-03	3.33E-02	5.13E-03	1.63E-02
Hope Creek	BWR	3293	12/20/86	4.80E-01	5.39E-01	8.47E-01	6.87E-01	1.36E-01
LaSalle 1	BWR	3323	1/1/84	N/R	N/R	6.17E-05	1.81E-05	0.00E+00
Limerick 1	BWR	3293	2/1/86	6.06E-01	0.00E+00	1.01E-01	6.77E-02	2.18E-02
Nine Mile Point 2	BWR	3323	3/11/88	0.00E+00	3.33E-04	3.47E-03	3.73E-03	1.52E-04
Perry 1	BWR	3579	11/18/87	2.06E-04	1.65E-03	5.18E-02	9.64E-02	5.63E-02
River Bend	BWR	2894	6/16/86	7.00E-02	1.62E-01	8.70E-01	9.00E-02	5.80E-01
Susquehanna 1	BWR	3293	6/8/83	4.02E-04	4.00E-04	2.91E-04	2.19E-04	3.91E-04
WNP 2	BWR	3323	12/13/84	1.70E-02	2.10E-03	1.80E-03	3.50E-03	3.20E-03
Average				1.47E-01	7.89E-02	1.91E-01	9.57E-02	8.24E-02
Standard Deviation				2.48E-01	1.81E-01	3.53E-01	2.11E-01	1.80E-01

Plant	Type	Licensed Power (MWt)	Commercial Operation	Total Body Particulate, Iodine, And Tritium Dose (mrem)				
				1987	1988	1989	1990	1991
Beaver Valley 2	PWR	2660	11/17/87	N/A	1.71E-02	7.36E-02	9.27E-02	2.25E-01
Braidwood 1	PWR	3411	7/29/88	N/A	3.00E-03	3.00E-03	3.18E-03	1.96E-02
Byron 1	PWR	3411	9/16/85	6.15E-02	8.25E-02	2.71E-04	7.15E-04	N/R
Callaway	PWR	3565	12/19/84	6.40E-03	4.07E-03	1.09E-03	1.19E-03	1.24E-03
Catawba 1	PWR	3411	6/29/85	3.33E-01	1/42E-01	1.90E-01	5.51E-01	3.26E-01
Harris 1	PWR	2775	5/2/87	2.16E-02	9.49E-02	3.79E-02	2.04E-02	1.56E-02
Millstone 3	PWR	3411	4/23/86	1.74E-02	5.03E-03	1.07E-02	7.05E-03	9.96E-03
Palo Verde 1	PWR	3800	1/28/86	2.61E-01	1.17E-01	5.13E-02	5.67E-02	5.77E-02
South Texas Project 1	PWR	3800	8/25/88	N/A	1.64E-01	8.33E-01	3.10E-02	3.33E-02
Vogtle 1	PWR	3411	6/1/87	9.83E-03	2.78E-03	3.71E-03	8.65E-04	1.96E-04
Waterford 3	PWR	3390	9/24/85	6.61E-01	1.79E-01	8.90E-02	1.30E-01	2.40E-01
Wolf Creek	PWR	3411	9/3/85	5.37E-02	1.04E-02	8.75E-02	1.44E-02	1.31E-02
Average				1.58E-01	6.18E-02	1.15E-01	7.58E-02	8.56E-02
Standard Deviation				2.23E-01	6.85E-02	2.33E-01	1.55E-01	1.18E-01
Clinton	BWR	2894	11/24/87	N/A	1.22E-02	2.04E-03	1.11E-04	1.59E-04
Fermi 2	BWR	3292	1/23/88	0.00E+00	2.15E-04	6.26E-04	2.04E-04	3.31E-04
Hope Creek	BWR	3293	12/20/86	1.31E+00	1.22E-02	9.36E-06	1.27E-02	5.77E-05
LaSalle 1	BWR	3323	1/1/84	N/R	N/R	7.45E-03	1.07E-04	1.15E-03
Limerick 1	BWR	3293	2/1/86	2.06E-01	1.73E-01	7.07E-02	4.50E-03	9.68E-07
Nine Mile Point 2	BWR	3323	3/11/88	8.00E-03	4.33E-03	1.20E-02	5.00E-03	8.57E-03
Perry 1	BWR	3579	11/18/87	2.60E-02	6.19E+00	1.79E-01	1.25E+00	1.84E+00
River Bend	BWR	2894	6/16/86	4.80E-03	3.30E-03	5.00E-03	4.50E+00	1.30E+00
Susquehanna 1	BWR	3293	6/8/83	4.88E-02	1.84E-02	2.80E-02	7.50E-02	4.77E-02
WNP 2	BWR	3323	12/13/84	1.10E-01	7.50E-01	1.50E-01	2.10E-01	1.30E-01
Average				2.14E-01	7.96E-01	4.55E-02	6.06E-01	3.33E-01
Standard Deviation				4.48E-01	2.04E+00	6.66E-02	1.42E+00	6.66E-01

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Plant	Type	Licensed Power (MWt)	Commercial Operation	Gamma Airdose (mrad)				
				1987	1988	1989	1990	1991
Beaver Valley 2	PWR	2660	11/17/87	N/A	7.62E-03	9.46E-03	1.39E-01	1.32E-02
Braidwood 1	PWR	3411	7/29/88	N/A	0	3.30E-02	1.18E-02	7.43E-02
Byron 1	PWR	3411	9/16/85	4.00E-03	4.00E-03	1.06E-03	1.56E-03	N/R
Callaway	PWR	3565	12/19/84	4.00E-02	7.98E-03	1.10E-02	1.27E-02	4.21E-03
Catawba 1	PWR	3411	6/29/85	7.54E-01	6/33E-01	3.80E-01	8.86E-01	4.27E-01
Harris 1	PWR	2775	5/2/87	3.45E-02	1.48E-01	5.93E-02	3.11E-02	2.49E-02
Millstone 3	PWR	3411	4/23/86	N/R	N/R	8.00E-03	1.12E-02	8.78E-03
Palo Verde 1	PWR	3800	1/28/86	3.25E-01	1.74E-01	7.87E-02	8.33E-02	1.00E-01
South Texas Project 1	PWR	3800	8/25/88	N/A	4.53E-01	6.15E-02	9.45E-03	4.17E-03
Vogtle 1	PWR	3411	6/1/87	1.02E+00	7.13E-01	1.35E-03	1.06E-03	1.65E-03
Waterford 3	PWR	3390	9/24/85	1.02E+00	7.55E-01	1.07E-01	1.01E+00	3.06E-01
Wolf Creek	PWR	3411	9/3/85		2.33E-02		2.78E-02	7.54E-02
Average				4.57E-01	2.29E-01	6.82E-02	1.85E-01	9.45E-02
Standard Deviation				4.65E-01	3.01E-01	1.09E-01	3.59E-01	1.41E-01
Clinton	BWR	2894	11/24/87	N/A	1.23E-04	6.33E-04	6.88E-04	2.40E-05
Fermi 2	BWR	3292	1/23/88	0.00E+00	1.79E-04	1.36E-01	4.58E-02	N/R
Hope Creek	BWR	3293	12/20/86	N/R	N/R	N/R	N/R	N/R
LaSalle 1	BWR	3323	1/1/84	N/R	N/R	1.29E-02	2.04E-04	4.97E-03
Limerick 1	BWR	3293	2/1/86	1.14E-03	5.70E-03	5.70E-03	1.31E-03	2.33E-03
Nine Mile Point 2	BWR	3323	3/11/88	N/R	N/R	N/R	N/R	N/R
Perry 1	BWR	3579	11/18/87	1.17E-01	2.55E+00	6.13E-01	3.65E-02	7.87E-02
River Bend	BWR	2894	6/16/86	N/R	N/R	N/R	N/R	N/R
Susquehanna 1	BWR	3293	6/8/83	5.60E-03	2.14E-03	2.82E-03	5.25E-03	3.80E-02
WNP 2	BWR	3323	12/13/84	2.80E-01	1.20E-01	3.10E-01	4.40E-01	1.50E-01
Average				8.07E-02	4.46E-01	1.54E-01	7.57E-02	4.57E-02
Standard Deviation				1.22E-01	1.03E+00	2.32E-01	1.62E-01	5.94E-02

**DATE
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END