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50-364

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10 CFR 50.90

Joseph M. Farley Nuclear Plant
Response to Request for Additional Information Related to Power Uprate
Facility Operating Licenses and Technical Specifications Change Request

Ladies and Gentlemen:

By letter dated February 14, 1997, Southern Nuclear Operating Company (SNC) proposed to amend the Facility Operating Licenses and Technical Specifications for Joseph M. Farley Nuclear Plant (FNP) Unit 1 and Unit 2 to allow operation at an increased reactor core power level of 2775 megawatts thermal (Mwt). NRC letters dated July 1, 1997; August 21, 1997; and October 14, 1997 requested SNC provide additional information. SNC responded by letters dated August 5, 1997; September 22, 1997; and November 19, 1997 respectively. SNC letters dated December 17 and 31, 1997; January 23, 1998; and February 12, 1998 responded to NRC questions resulting from conference calls. During telephone conference calls on February 10 and 13, 1998, SNC responded to additional NRC Staff questions. Attachment I provides the SNC responses to these questions. Attachment II includes corrections to the power uprate BOP Licensing Report (page 62). Attachment III provides requested information associated with plateout and containment spray system iodine removal rates.

If you have any questions, please advise.

Respectfully submitted,

Dave Morey

Sworn to and subscribed before me this 26 *day of* Feb *1998*

Brenda W Singleton
Notary Public

My Commission Expires: October 3, 2001

MGE/maf: pwrup31.doc
Attachments

A001

cc: Mr. L. A. Reyes, Region II Administrator
Mr. J. I. Zimmerman, NRR Project Manager
Mr. T. M. Ross, Plant Sr. Resident Inspector

ATTACHMENT I

SNC Response to NRC Request For Additional Information
Related To Power Uprate Submittal - Joseph M. Farley Nuclear Plant, Units 1 and 2

SNC RESPONSES TO NRC QUESTIONS RESULTING FROM
NRC/SNC CONFERENCE CALLS ON FEBRUARY 10 AND 13, 1998

SNC Response to NRC Request For Additional Information
Related To Power Uprate Submittal - Joseph M. Farley Nuclear Plant, Units 1 and 2

NRC Question No. 1 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

It was indicated that the control room X/Q values were revised and were based upon one year's worth of meteorological data, April 1972 – March 1973. Provide the basis for concluding that the control room atmospheric dispersion factors should be based upon this one year's meteorological data and that this data is representative of the meteorological conditions which are anticipated to occur over the 40 year life of the facility. If this is an update of previous values due to re-assessments of the data or correction of information, provide the difference between the present and previous values and the time when the revisions were made and/or any items corrected.

SNC Response No. 1

From the evaluation documented in Safety Evaluation Report, dated May 2, 1975, and the FSAR, it was concluded that one year of meteorological data was representative of the meteorological data over the 40 year life of the facility for determining the control room atmospheric dispersion factors. Section 2.3 of the Safety Evaluation Report states the onsite joint frequency data from April 1971 through March 1973 provided an acceptable basis to make conservative and representative estimates of atmospheric dispersion characteristics for accidental and routine gaseous releases. FSAR Section 2.2.2 compares the data from April 1, 1971 through March 31, 1972 with the data from April 1, 1972 through March 31, 1973. Analysis of this data showed close similarity between the two data sets.

During the FNP control room ventilation system self-assessment, it was identified that the atmospheric dilution factors (X/Q) used in the control room dose assessment were not based upon as-built conditions. To be consistent with the FNP licensing basis as described in FSAR Section 9.4.1.6.2.3, the atmospheric dilution factors were re-assessed based on the meteorological data from April 1972 through March 1973, as well as the as-built configuration of the control room intake. The joint frequency data obtained from April 1972 through March 1973 is referenced in FSAR Table 2.3-8A.

Original Control Room Atmospheric Dilution Factors

0 - 2 hours	3.0×10^{-3} s/m^3
2 - 8 hours	1.9×10^{-3} s/m^3
8 - 24 hours	1.2×10^{-3} s/m^3
1 - 4 days	8.6×10^{-4} s/m^3
4 - 30 days	4.1×10^{-4} s/m^3

Re-assessed Control Room Atmospheric Dilution Factors

0 - 2 hours	3.28×10^{-3} s/m^3
2 - 8 hours	2.65×10^{-3} s/m^3
8 - 24 hours	2.19×10^{-3} s/m^3
1 - 4 days	1.64×10^{-3} s/m^3
4 - 30 days	1.08×10^{-3} s/m^3

The above re-assessed control room atmospheric dilution factors have been included in Revision 14 of the FSAR for Unit 1 (FSAR Table 15.4-16A).

B/dm - 2/23/98

NRC Question No. 2 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

For the [RCP] locked rotor accident, is there any fuel melting which occurs?

SNC Response No 2

The review of the RCP locked rotor analysis results confirmed that fuel melting will not occur during this postulated ANS Condition IV event.

W/[illegible] - 2/13/98 & SNC/mge - 2/13/98

NRC Question No. 3 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

Has it been confirmed by analysis that in the event of a SGTR, there will be no tube uncovery nor immediate flashing in the steam generator with the ruptured tube? It is stated on page 58 of the BOP Uprate Licensing Report that uncovery was evaluated, but there is no discussion of the results of the evaluation which was performed relative to uncovery and flashing. Were cases evaluated [where] the PORV of the faulted steam generator failed closed in one case and failed open in another case?

SNC Response No 3

Westinghouse confirmed on a generic basis, applicable to the Farley units, that the effects of partial steam generator tube uncovery on the iodine release for an SGTR is negligible. This basis is documented in WCAP-13247, "Report on the Methodology for Resolution of the Steam Generator Tube Uncovery Issue," dated March 1992, with NRC approval of the submittal and agreement with the conclusion provided in a letter dated March 10, 1992. The supporting analyses included consideration of a stuck open PORV on the ruptured steam generator. Some flashing of break flow would be expected (see SNC response to Question No. 7 below) but is not considered as part of the Farley licensing basis analysis.

A single case evaluation, which considers a stuck open PORV, was performed similar to that presented in the current FSAR, but using power uprate mass flow rates. No studies of additional single failures were performed. RCS activity was assumed to be at the Technical Specifications limit (0.5 μCi/gm, which has since been reduced) with no iodine spike, secondary side activity at the Technical Specifications limit (0.1 μCi/gm), primary to secondary leakage to the intact generators at the Technical Specifications limit (150 gpd/generator), and flow from the ruptured tube based on power uprate (150,000 lbs in 30 minutes). Leakage from the uncovered tubes is modeled as a direct 100% release to the atmosphere for 30 minutes without partitioning. The results continue to meet the FSAR results, i.e., a small fraction of the 10 CFR 100 limit.

W/ub & jm - 2/12/98 & SCS/jaw - 2/12/98

NRC Question No. 4 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

For the power uprate amendment, the control room volume is indicated as 114,000 ft^3. Previous IPC amendments and UFSAR had indicated a volume of 69,000 ft^3. What is the basis for the significant increase in volume?

SNC Response No. 4

During the FNP control room ventilation self-assessment, it was identified that the control room volume did not include other volumes with which the control room proper communicates, e.g., the volume above the ceiling tiles. As a result, subsequent control room dose calculations evaluated the consequences of the accident based on a control room volume of 114,000 ft^3. This value is included in FSAR Table 15.4-16A, Revision 14, for Unit 1.

B/dm - 2/11/98

NRC Question No. 5 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

The doses calculated for a locked rotor accident by the staff are a factor of 10 lower than those calculated by the licensee if a partition factor of 0.01 is used, but approximately the same if a partition factor of 0.1 was used. Was a partition factor of 0.1 actually utilized by the licensee for the locked rotor accident rather than a factor of 0.01?

SNC Response No. 5

A partition factor of 0.01 was used in the calculation. However, during the review to verify the partition factor, it was discovered that the source term had been inadvertently used for a core melt in lieu of the correct value for a gas gap release. (See response to Question No. 2 above.) When the correct source term is used, the doses decrease by approximately a factor of ten. As a result of this finding, page 62 of the BOP Licensing Report has been revised. The revised page is provided in Attachment II.

SCS/jaw - 2/12/98

NRC Question No. 6 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

At what time after the accident are the containment sprays initiated, and at what time after the accident is recirculation from the containment sump initiated? It is the staff's understanding that recirculation must be manually initiated by the operators. Is this the case? Is there a period of time between the initial containment spray operation and the initiation of recirculation in which no spraying is occurring?

SNC Response No. 6

Based on review of the containment analysis results, spray initiation occurs at approximately 56 seconds. This is modeled coincident with core melt at 0 seconds for radiological consequences evaluation. Based on the draw-down rate from the RWST, recirculation from the sump will start after approximately 20 minutes.

Following initiation of containment spray, the transfer of spray pump suction from the RWST to the containment sump is an evolution controlled by the plant emergency response procedures. The operator manually transfers the spray pump suction valve alignment without interrupting spray flow.

SCS/jaw - 2/12/98 & SNC/mge - 2/12/98

NRC Question No. 7 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

Provide the flashing fractions as a function of time in the faulted steam generator for the SGTR accident for two cases. In the first case, the PORV for the faulted steam generator is assumed to fail closed. In the second case, the PORV for the faulted steam generator is assumed to fail open. In both cases, the onset of the SGTR event is assumed to occur coincident with a loss of offsite power.

SNC Response No. 7

The Farley SGTR analysis does not include flashing fractions as a function of time. As requested by the NRC Staff, the attached figures provide examples of the break flow flashing fraction as a function of time for detailed SGTR transient analyses performed for plants which include flashing in the licensing basis calculation. These transients model the expected operator actions to terminate primary to secondary break flow, including isolation of the ruptured steam generator, cooldown with the intact SGs and depressurization with the pressurizer PORV. Figure 1 assumes that a PORV on an intact SG fails closes. Figure 2 assumes that the PORV on the ruptured SG fails open and requires operator action to close the associated PORV block valve. Both analyses assume a loss of offsite power. The loss of offsite power assumption for these analyses is the same as for Farley, which is discussed in Section 6.3 of WCAP-14723.

W/ub & jm- 2/12/98

NRC Question No. 8 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

In Table 2 of the Enclosure of the December 9, 1997 letter from D. N. Morey to the Staff, it was indicated that the control room pressurization filter flow was 450 cfm for the amendment request involving the spray additive tank removal, the power uprate and the ESF filters. However, the technical specification change proposed for the ESF filters amendment indicates a flow of 300 cfm. Explain the basis for the differences. Which value is the one which is utilized in the analysis by the licensee? Previous staff analyses had determined that a flow rate of 270 cfm (300 cfm −10%) was more limiting from a dose standpoint.

SNC Response No. 8

The Technical Specifications limit remains 300 cfm ± 10%. 450 cfm is the maximum calculated fan runout capability that was used in our calculations. Previous parametric studies by the licensee, and the Iodine Protection Factor (IPF) formula from "Nuclear Power Plant Control Room Ventilation System Design For Meeting General Criterion 19" by K. G. Murphy and Dr. K. M. Campe, indicate that higher intake flow results in lower IPF and higher doses.

SCS/jaw - 2/12/98

NRC Question No. 9 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

Table 2 of the December 9, 1997 letter from D. N. Morey indicates that the ECCS leakage was assumed to be [20] times the leakage presented in Table 6.3-8 of the UFSAR. However, Table 1 of the June 30, 1997 letter on the ESF filter amendment request indicates that ECCS leakage is 10 times the value in UFSAR Table 6.3-8. Which value should be used in the accident analyses? At what leakage level does the TMI III.D.1.1, Leakage Reduction Program, require maintenance actions to repair leaking systems and/or components?

SNC Response No. 9

The ECCS recirculation loop leakage value assumed for the ESF filter amendment radiological evaluations was 10 times the total shown in FSAR Table 6.3-8. This value was doubled (i.e., 20 times the Table 6.3-8 total) in accordance with SRP 15.6.5, Appendix B, for use in accident analysis dose calculations.

The Farley "Borated Water Leakage Assessment and Evaluation Program" is described in manual FNP-0-M-101. The program manual provides guidelines for identification of boric acid leaks and assessment of repair need and component wastage. Identified active leaks are documented upon discovery and subsequently trended. If the total trend from all leaks approaches the 3760 mls/hr criterion, then corrective actions are initiated by the plant staff. For any one leakage source, significant leakage is defined as approximately 10 drops per minute or greater. Significant leakage requires initiation of a Deficiency Report (DR), which is used to facilitate necessary repairs.

SCS/jaw - 2/12/98 & SNC/mge & rlm - 2/21/98

NRC Question No. 10 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

What are the differences in the control room filter unit and the recirculation filter unit and how are they aligned in the emergency mode of operation?

SNC Response No. 10

Each train of the control room HVAC system has three safety-related filters. In the emergency mode, the pressurization unit supplies filtered outside air to pressurize the control room with a flow rate as described in the response to Question No. 8 above. Also, in the emergency mode, the recirculation and filtration units provide recirculation filtration in parallel for the air inside the control room with a total flow rate of 3000 cfm ± 10%.

SCS/jaw - 2/12/98

NRC Question No. 11 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

In Table E of the Attachment 5 of the August 5, 1997 letter from the licensee on the power uprate, it is indicated that the steam released from the faulted steam generator in the event of a MSLB accident is 473,000 lbs over 30 minutes. Previous information provided in various IPC amendment requests indicated that the release from the faulted steam generator would be [96,200] lbs plus all primary to secondary leakage. For the power uprate amendment request, how does the increase in power level affect the releases from the intact and faulted steam generators and what are the values for releases to these two sources? What values were utilized in the power uprate amendment request?

SNC Response No. 11

For the various IPC submittals, the initial steam generator mass of 96,200 lbs was used for dose analyses. This mass corresponds to a pre-power uprate full power value which is appropriate for these calculations. For the power uprate, since the assumed activity levels are unchanged and the MSLB releases are based on a zero power steam generator mass, the increase in power level has insignificant impact on the releases. The power uprate steam releases for the faulted generator are the initial inventory of 168,000 lbs plus main and auxiliary feedwater flow until isolation at 30 minutes of 290,000 lbs. The more conservative steam generator mass corresponds to a hot zero power value. The flow is conservatively increased by 5% for evaluation of radiological consequences [i. e. , (168,000) + (1.05) (290,000) ≅ 473,000]. The intact generator mass releases for power uprate are 323,000 lbs (0 – 2 hr) and 695,000 lbs (2 – 8 hr), which were also conservatively increased by 5% for evaluation of radiological consequences.

SCS/jaw - 2/25/98

NRC Question No. 12 (Reference February 10, 1998 NRC/SNC Conference Call and February 11, 1998 NRC Facsimile)

What was the basis for [the building volume assumed for] dilution for the fuel handling accident?

SNC Response No. 12

The building volume considered the open area directly above the spent fuel pool area bounded by concrete walls. It does not include any adjacent areas such as the new fuel storage area, heat exchanger or pump rooms, HVAC equipment rooms, etc. For two trains of exhaust (2 x 4000 cfm assumed), the building volume is exhausted over nine times in two hours, consistent with Regulatory Guide 1.25.

SCS/jaw 2/12/98

NRC Question No. 13 (Reference February 10, 1998 NRC/SNC Conference Call)

There are differences between the iodine removal rate functions calculated by the staff and those provided by the licensee. In particular, the elemental iodine spray removal λ and deposition λ are different. Provide the bases for your values.

SNC Response No. 13

The spray removal λ was chosen based on values from NUREG/CR-0009. This λ is considered to be a conservatively small value which might occur for boric acid spray water contaminated with iodine. The deposition λ was calculated in accordance with the methodology described in NUREG-0800, Section 6.5.2, Revision 1. The total elemental removal rate assumed in our analysis (spray plus deposition) is conservatively lower than the Staff values discussed in our telephone call of February 10, 1998. As requested by the Staff, pertinent pages from the calculation are included in Attachment III.

SCS/jaw - 2/12/98

NRC Question No. 14 (Reference February 13, 1998 NRC/SNC Conference Call)

Regarding modeling assumptions f[illegible] non-LOCA events that assume LOOP and ESF actuation, what is the sensitivity of the trans[illegible]sults to the time of LOOP.

SNC Response No. 14

Previous responses have addressed which non-LOCA events are analyzed with a loss of offsite power (LOOP). For Farley, the events analyzed with a LOOP are the loss of normal feedwater, feed line break, steam line break, and locked rotor. With the exception of the locked rotor, these events are ones that assume a Engineered Safety Features (ESF) system actuation. This response will address the sensitivity of the transient results to the time of LOOP, specifically, whether or not the LOOP occurs at event initiation, at the time of reactor trip, or, as typically assumed in Westinghouse non-LOCA analyses, at approximately 2 seconds following reactor trip.

The LOOP affects the analysis assumptions primarily in two ways: (1) the time the RCPs are tripped, and (2) the ESF functions are delayed or sometimes diminished (e.g., longer SI delay due to diesel sequencing, diminished AFW capacity due to single failure considerations, etc.). The assumed time delay of RCP trip is typically the same for non-LOCA analyses. This delay time is approximately 2 seconds after reactor trip. Following RCP trip, core flow decreases as the RCPs begin to coast down. The other aspects of the LOOP, such as diesel sequencing and AFW startup delays and flow rates tend to be plant specific and are discussed in the Farley FSAR and NSSS Licensing Report. It should be noted, however, that no assumption with respect to LOOP was changed for the Farley Units 1 and 2 Uprating Program.

Steam Line Break (FSAR 15.4.2)

For the steam line break event, the LOOP occurs 3 seconds after event initiation or approximately 2 seconds (actual is 2.2 seconds) after the low pressurizer pressure safety injection setpoint is

reached. This is consistent with standard Westinghouse steam line break analysis methodology and is reported in WCAP-9226, Rev. 1.

For the steam line break core response analysis, the case with offsite power available is the limiting case. Changing the time of the LOOP to the beginning of the event will not change conclusions regarding the core response and DNB analysis, nor change the limiting case from that which has offsite power available.

Loss of Normal Feedwater (FSAR 15.2.8 and 15.2.9)

Two loss of normal feedwater cases are analyzed: (1) loss of normal feedwater with offsite power (LONF, FSAR 15.2.8) and (2) loss of normal feedwater with a loss of offsite power (LOOP, FSAR 15.2.9). The LONF case is analyzed to ensure that the AFW system can remove the core decay heat and RCP heat. The LOOP case is analyzed to ensure that the AFW system in conjunction with the primary side under a natural circulation flow regime can remove the core decay heat. The analysis applies a conservative acceptance criterion of ensuring that the pressurizer does not fill. Since both FSAR events have the same assumptions with respect to the AFW system, the LONF case is the more limiting of the two cases due to the RCP pump heat addition.

The LOOP event is not particularly sensitive to the time the loss of offsite power occurs. The transient is relatively long in duration with the limiting condition (approach to pressurizer fill) being reached at 1466 seconds. The reactor trip and AFW initiation occurs as result of the low-low steam generator water level protection. The AFW actuation delay is 60 seconds. For the case presented in the FSAR, the loss of offsite power is assumed to occur 2 seconds after reactor trip. Assuming that the loss of offsite power occurred exactly at the time the trip setpoint is reached would provide essentially the same results.

Analyzing the event with the LOOP at event initiation would result in reactor trip on low reactor coolant flow very early in the transient. The pre-trip portion of the event analyzed in this manner would be similar to and no more limiting than the complete loss of flow event presented in FSAR Section 15.3.4. The reactor trip would occur on low RCS flow. AFW initiation continues to be a result of the low-low steam generator water level protection. As to the post-trip, long term cooling effects, modeling the event with a loss of offsite power at event initiation is less limiting than the LOOP event as currently analyzed since the steam generators enter the post-trip (core decay heat removal) phase of the transient with much more steam generator water inventory.

The evaluation statements made above were verified with several LOFTRAN simulations. (LOFTRAN is an NRC approved non-LOCA Code.) The results of these simulations are graphically shown on the attached figure. Figure 3 shows a plot of pressurizer volume versus time for the FSAR case (LOOP at reactor trip plus 2 seconds), LOOP at reactor trip, and LOOP at event initiation. Note that the two cases LOOP at RT plus 2 seconds and LOOP at RT are essentially the same. Assuming the LOOP at event initiation is not limiting.

Feed Line Break (FSAR 15.4.2)

The feed line break (FLB) accident, similar to the LONF and LOOP events, is analyzed to ensure that the AFW system can remove the core decay heat for the LOOP case and the core decay heat

and RCP heat for the case with offsite power. For the case analyzed with a loss of offsite power, the LOOP is assumed to occur 2 seconds after reactor trip occurs.

For reasons similar to the LONF event, the FLB event is also not sensitive to the time of the LOOP. In all cases main feedwater is terminated at the time the feed line break occurs, and the AFW actuation delay is relative to the time the low-low steam generator water level signal is reached. The limiting conditions are reached at 900 seconds or more after the break occurs. Therefore, a LOOP at the time of the reactor trip with no delay would result in essentially the same result as the case with the LOOP delayed by 2 seconds. Also similar to the LONF w/ LOOP, should the LOOP occur at event initiation, the transient conditions would be much more favorable than those currently analyzed. This result is primarily due to the steam generators entering the post-trip (core decay heat removal) phase of the accident with much more steam generator water mass available to remove the decay heat.

W/wjs - 2/20/98

Figure 1

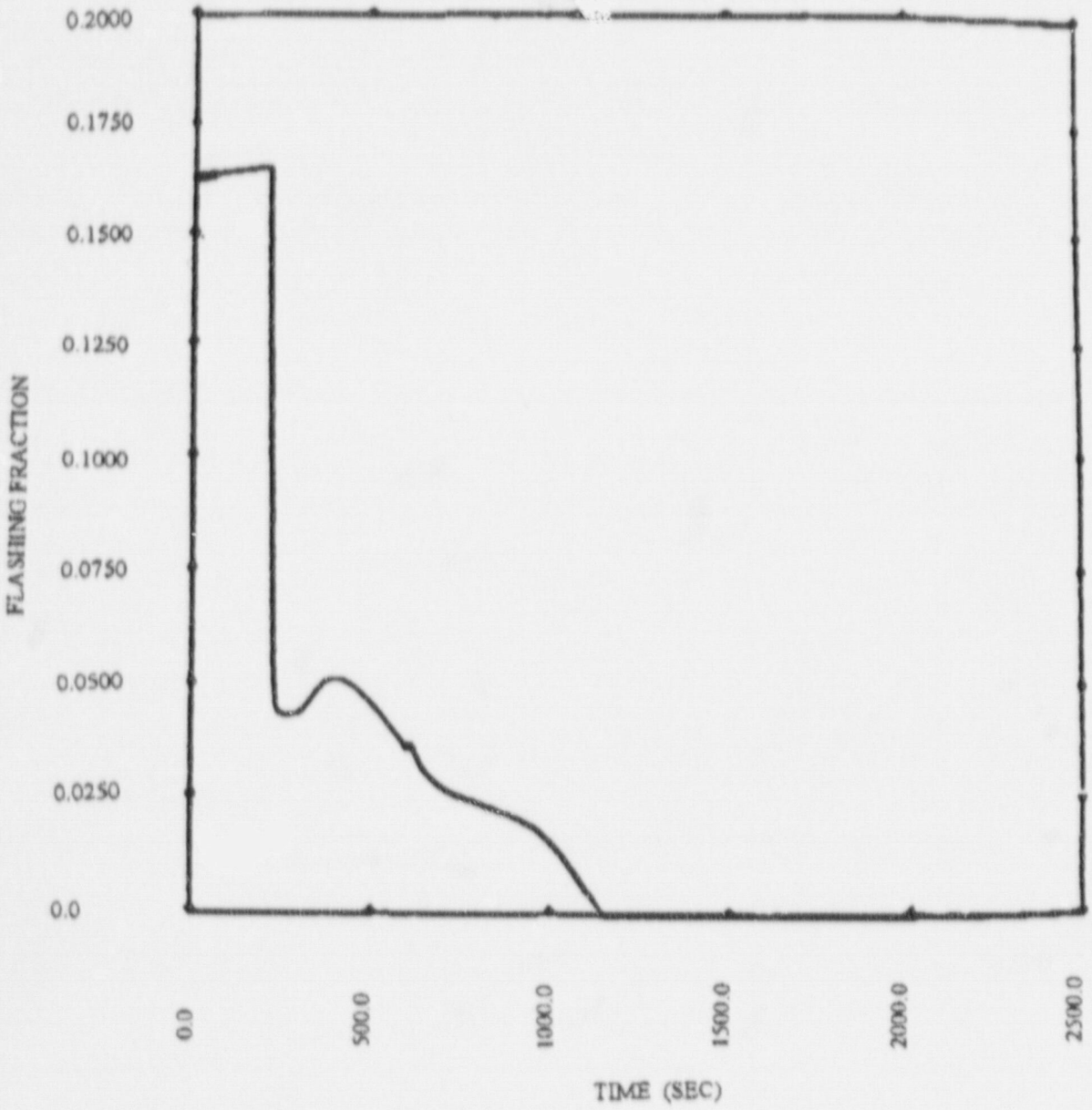

0.2000
0.1750
0.1500
0.1250
0.1000
0.0750
0.0500
0.0250
0.0
FLASHING FRACTION
0.0
500.0
1000.0
1500.0
2000.0
2500.0
TIME (SEC)

Figure 2

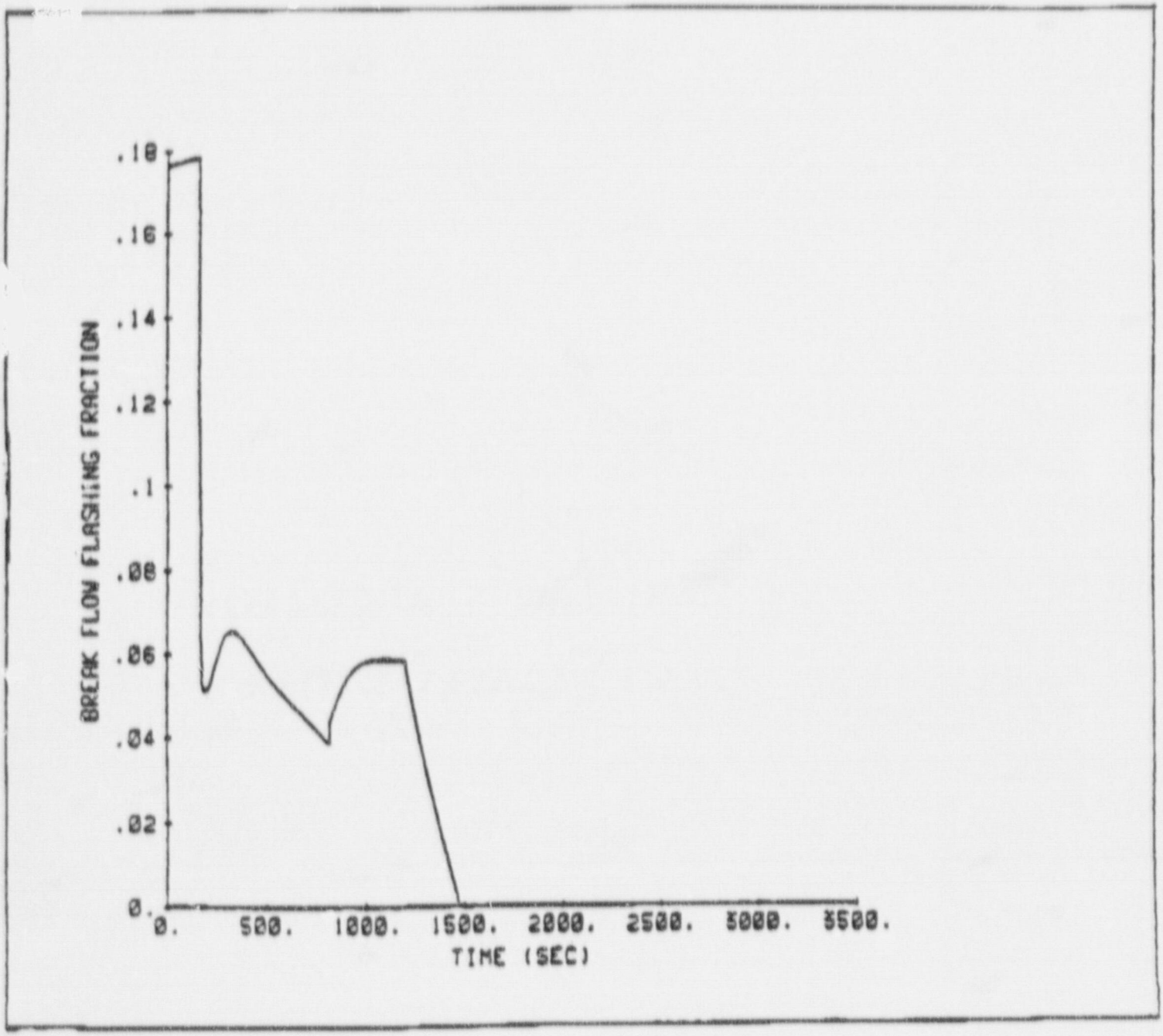
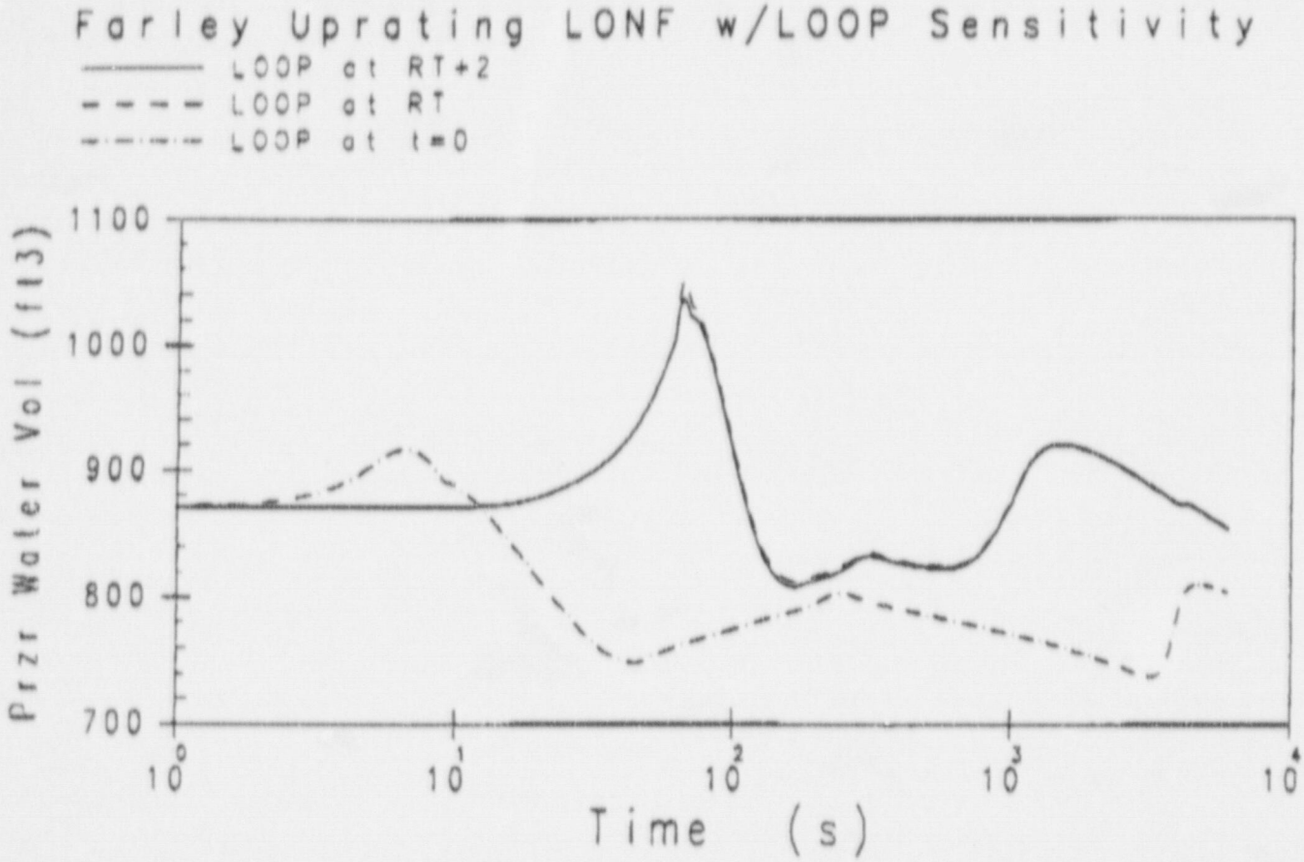

BREAK FLOW FLASHING FRACTION
.18
.16
.14
.12
.1
.08
.06
.04
.02
0.
0.
500.
1000.
1500.
2000.
2500.
3000.
3500.
TIME (SEC)

Figure 3


Farley Uprating LONF w/LOOP Sensitivity
LOOP at RT+2
LOOP at RT
LOOP at t=0
Przr Water Vol (ft3)
1100
1000
900
800
700
10^0
10^1
10^2
10^3
10^4
Time (s)

ATTACHMENT II

SNC Response to NRC Request For Additional Information
Related To Power Uprate Submittal - Joseph M. Farley Nuclear Plant, Units 1 and 2

CORRECTED PAGE NO. 62

"FARLEY NUCLEAR PLANT UNITS 1 AND 2
POWER UPRATE PROJECT BOP LICENSING REPORT"

(ATTACHMENT 6 TO SNC SUBMITTAL DATED FEBRUARY 14, 1997)

The potential impact of uncovery of the steam generator tubes during the event was also evaluated for uprated conditions. Assuming technical specification limits for RCS activity (0.5 μCi/gm) and leak rate (150 gpd per generator) and release directly to the environment (*i.e.* no mixing with the secondary side water) the offsite doses remain well within the 10 CFR 100 guidelines.

2.16.7.3.8 Evaluation of the Radiological Consequences of an RCP Locked Rotor

The radiological consequences of RCP locked rotor releases assuming 20% of the fuel clad/pellet gas gap is released to the RCS with subsequent leakage to the steam generators and secondary side steam releases were evaluated utilizing the assumptions of Standard Review Plan Section 15.3.3. These releases result in offsite doses that are a small fraction of the guidelines of 10 CFR 100, which meets the acceptance criteria.

	Thyroid Dose (Rem)	Whole Body Dose (Rem)	Beta Skin Dose (Rem)
EAB	1.62	0.20	0.18
LPZ	3.02	0.11	0.09

2.16.8 Summary of Conclusions

No changes or additions to structures, equipment, or procedures are necessary to provide adequate radiation protection for the operators or the public during normal or post-accident operations to support the uprate. The existing structures, systems, and components can safely handle the changes in post accident source terms and releases from the uprate conditions, and resulting onsite and offsite doses are less than the 10 CFR 100.11 guidelines and are within the Standard Review Plan guidelines. Therefore the radiological consequences acceptance criteria fc postulated Condition II, III, and IV events are satisfied.

ATTACHMENT III

SNC Response to NRC Request For Additional Information
Related To Power Uprate Submittal - Joseph M. Farley Nuclear Plant, Units 1 and 2

EXCERPT FROM FARLEY NUCLEAR PLANT SCS CALCULATION SM-95-8931-002

OFFSITE AND CONTROL ROOM DOSE FOR UPRATE WITH TSP FOR pH CONTROL

(Sheets 2, 3, 4, 5, & 10)

Major Equations :

The calculations were performed using the TACT5 computer program running on an NEC pentium desktop computer. The performance of TACT5 on the NEC machine was verified in reference 1. The test problems of reference 1 were successfully rerun on the computer to verify proper execution of the program. A directory listing of the TACT5 .EXE and library files is included in Attachment 1. These files were compared to the listings in reference 1 to verify the proper files are installed.

The equations used by the TACT5 computer program are described in reference 2.

Assumptions :

1. To aid retention of iodine in the sump, trisodium phosphate (TSP) will be added to the sump solution in sufficient quantity to maintain a pH of 7.5 (Ref 18d). Determination of the quantity of TSP to be added is not within the scope of this calculation. The effectiveness of the iodine retention as reflected in the decontamination factor and removal process cutoffs will be determined in accordance with the guidance of reference 4.

2. Removal of elemental iodine by the boric acid spray injection solution will be assumed to have a removal constant of 1.4/hr. Reference 9 indicates a conservative constant of 0.9/hr and reference 8 (section 6.1.11) indicates a value of 1.44/hr, with an expected value 7.8 times this value. Additional research (reference 14) indicates that injection of fresh, uncontaminated by iodine, spray solutions are effective with or without additives. A value of 1.4/hr is chosen as a conservative estimate.

3. Coatings typically have plateout retention capacity well in excess of the inventory released (reference 8, page 68). Thus all iodine plated out will remain on the plateout surface.

4. The plate-out (deposition) removal constants from will be estimated for this calculation. The deposition velocity used to derive these values (0.68cm/sec for zinc/zinc coated surfaces and 0.49 cm/sec for epoxy coated surfaces) are consistent with the data presented in reference 8, Table 5 for Dimetcote, Corbo-zinc and Amercoat 66 which are similar to the coatings used at FNP. This is assumed to reduce to approximately 5% of the estimated rate after reaching 1% of the initial concentration and to stop at 0.1% of the initial concentration (references 8 and 13).

5. No credit for removal of organic iodine is taken, nor is credit taken for removal of elemental or particulate iodine below the assumed removal cutoff of 1000. Cutoff times are determined based on I_{131} concentrations, *i.e.* ignoring decay of short lived isotopes. Removal coefficients and removal process cutoffs used per assumptions 1-4 are shown in Table 1. Since time dependent plateout is modeled (in lieu or instantaneous 50%), and organic iodine is not removed (except via leakage) the core releases to the containment are modeled to maintain the same organic source term as discussed by reference 7, i.e. 50% total release as 95.5% elemental, 2% organic, and 2.5% particulate (references 7, 13, 20).

6. The sump pH reduction (from previous NaOH addition levels) does not impact the containment pressure/temperature response; thus the sump (recirculation fluid) temperature and flashing fraction, and the ECCS leakage contribution to the total dose, are modeled as described in reference 10. ECCS leakage, taken from reference 3a, is assumed to be 10 x 4000 cc/hr. This conservatism is intended to avoid any requirement to closely monitor, or have an explicit Technical Specification on, ECCS leakage.

7. Hydrogen purge may be initiated as a backup to the redundant hydrogen recombiners. The initiation time (18 days)and flow rate required (35 cfm) are taken from reference 17.

References :

1 Nuclear Support Calculation number N-94-02, "Verification of TACT5," revision 0.

2 NUREG/CR-5106, SAIC-88/3023, "User's Guide for the TACT5 Computer Program."

3 FNP Final Safety Analysis Report
- a. Table 6.3-8
- b. Table 15.4-14
- c. Table 15.4-16
- d. Table 15.4-20
- e. Table 15B-2
- f. Figure 3.7-20
- g. Table 6.2-2
- h. Table 6.2-5
- i. Table 15.4-18

4 NUREG-0800, "U.S. Nuclear Regulatory Commission Standard Review Plan," Section 6.5.2, Revision 1.

5 10 CFR 50, Appendix A, General Design Criterion 19, "Control Room."

6 10 CFR 100.11, "Determination of exclusion area, low population zone and population center distance."

7 Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," June, 1974.

8 NUREG-CR-0009, "Technological Bases for Models of Spray Washout of Airborne Contaminants in Containment Vessels," October 1978.

9 WASH-1329, "A review of Mathematical Models for Predicting Spray Removal of Fission Products in Reactor Containment Vessels," June, 15,1974.

10 NUREG-0800, "U.S. Nuclear Regulatory Commission Standard Review Plan," Section 15.6.5, Revision 1.

11 Letter AP-21370, dated February 6, 1996, "Up-date Control Room Dose Assessment."

12 Murphy, K.G. and Campe, Dr. K.M., 13th AEC Air Cleaning Conference, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19."

13 WCAP-11611, March 1988, "Methodology for Elimination of the Containment Spray Additive."

14 Davis, R.E., et al, "Fission Product Removal Effectiveness of Chemical Additives in PWR Containment Sprays," Technical Report A-3788, 8/12/86, attached to proposed revision 2 to Standard Review Plan 6.5.2, with AIF letter of 5/11/87.

15 Letter ALA-95-756, dated 12/15/95, Analysis Input Assumption List

16 Letter ALA-99-508, dated February 1, 1966, "Final Core Inventory Source Terms."

17 Calculation 40.05, Revision 3, Post Accident Hydrogen Generation Analysis.

18 FNP Calculations
   a. Mechanical calculation 4.2
   b. Mechanical calculation 4.1
   c. SM-93-0121-001
   d. SM-95-8931-001

19 FNP Technical Specifications
   a. 3/4.6.1.2

20 NUREG-0588

21 Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water Cooled Nuclear Power Plants," Rev. 0.

The minimum pH is maintained at 7.5 as discussed in assumption 1. The partition factor between liquid and gas phases is based on reference 4, Figure 6.5.2-1. With a pH of 7.5, the partition coefficient is 440.

The elemental iodine spray removal coefficient is 1.4 hr^{-1} per assumption 2. With a partition coefficient of 440, the decontamination factor (DF) limit is based on reference 4:

$$DF = 1 + (4.92E+4\ ft3)(440)/(1.67E+6\ ft3) = 14.0$$

where the sump and containment volumes are as provided above.

Elemental iodine plateout is calculated per assumption 4, with effective plateout areas taken as the containment heat sinks (Ref 3g):

Zinc/zinc painted surfaces from heat sinks 1 and 4-15 = $2.70 \times 10^5\ ft^2$
Epoxy surfaces from heat sinks 2 and 3 = $6.47 \times 10^4\ ft^2$

Then per references 8 and 13,

$$\lambda = 118\ \Sigma\ (\text{Deposition velocity} \times \text{Area / Volume})$$

$$= 118\ \frac{(0.68 \times 2.70 \times 10^5 + 0.49 \times 6.47 \times 10^4)}{2.03 \times 10^6} = 12.5\ hr^{-1}$$

This decreases to approximately 5% of the initial value or about 0.5 hr^{-1} after reducing the original concentration by 100, and to 0 after a reduction of 1000.

The particulate spray removal coefficient is calculated as described in reference 8 (page 118):

$$\lambda = \frac{3(100\ ft)(2175\ gpm)(0.1\ cm^{-1})}{2(1.669E6\ ft^3)(7.5\ gal/ft)} \times \frac{60\ min}{hr} \times \frac{30.5\ cm}{ft} = 4.77\ hr^{-1}$$

where 0.1 cm^{-1} is a conservative washout parameter (E/d) from reference 8, section 5.3.1 (p 34), until the particulate DF = 100. After this time the value decreases by a factor of ten, until a DF of 1000 is achieved. Drop fall height is assumed to be 100 ft based on reference 3f, and spray flow of 2175 gpm is based on references 3h and 18c.

These values are input to TACT5 which is run iteratively to determine the cutoff times as described in assumptions 1-5 above. The removal rates and cutoff times are shown below: