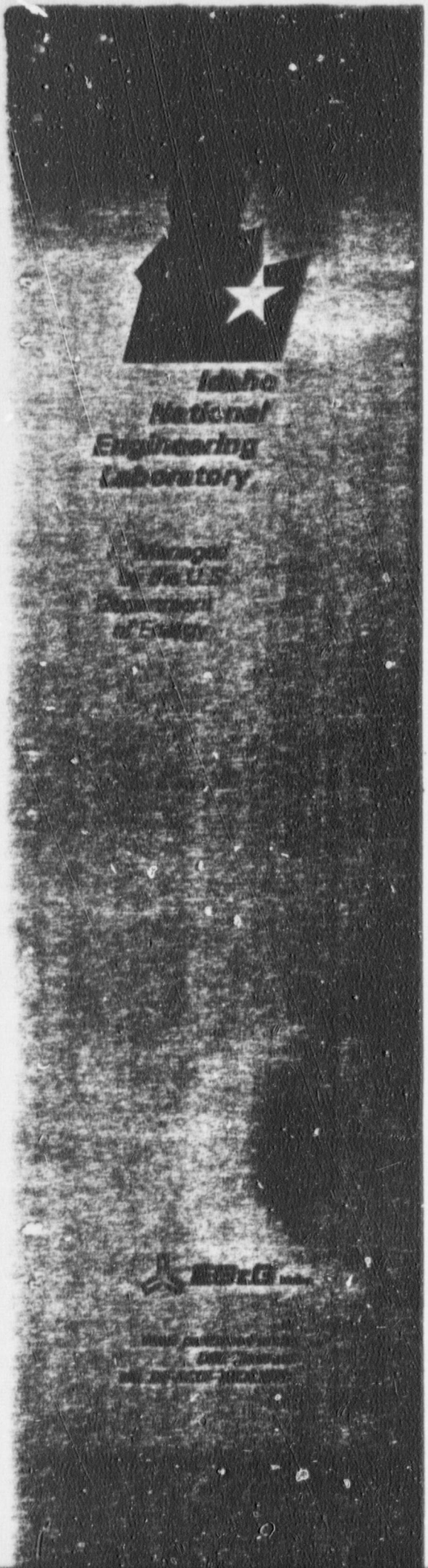

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July 1988

TECHNICAL EVALUATION REPORT

TECHNICAL EVALUATION REPORT FOR THE EVALUATION
OF ODCM UPDATED THROUGH REVISION 5
JAMES A. FITZPATRICK NUCLEAR POWER PLANT

T. S. Bohn
W. Serrano
A. L. Freeman
M. R. Winberg

Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION

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TECHNICAL EVALUATION REPORT

for the

EVALUATION OF ODCM UPDATED THROUGH REVISION 5

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

NRC Docket No. 50-333 NRC License No. DPR-59

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ABSTRACT

The Offsite Dose Calculation Manual for the James A. FitzPatrick Nuclear Power Plant (JAFNPP) contains a description of the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitor alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program. Revision 0, dated June 15, 1983, was submitted to the NRC with letter dated October 26, 1983 and was, in general, approved by the NRC on May 29, 1985. Subsequent changes identified in Revisions 1 and 2 were reported to the NRC in the Semiannual Effluent Release Report for July-December 1985. The ODCM, updated through Revision 2 was, in general, approved on an interim basis by the NRC on October 2, 1986. Changes identified in Revisions 3 and 4 were reported to the NRC in Semiannual Effluent Release Report for January-July 1986 and July-December 1986 respectively. Also, Revision 5, dated March 1987, was submitted to the NRC in letter dated April 2, 1987. The NRC transmitted the ODCM updated through Revision 5 to the Idaho National Engineering Laboratory (INEL) for review. The ODCM, complete through Revision 5, was reviewed in its entirety by EG&G Idaho at the INEL and the results of the review are presented in this report. It was determined that the ODCM updated through Revision 5 uses methods that are, in general, in agreement with the guidelines of NUREG-0133. However, it is recommended that another revision to the ODCM be submitted to address the discrepancies identified in the review.

FOREWORD

This report is submitted as partial fulfillment of the "Review of Radiological Issues" project being contracted by the Idaho National Engineering Laboratory for the U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation. The U. S. Nuclear Regulatory Commission funded the work under FIN D6034 and NRC B&R Number 20 19 05 03.

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1. INTRODUCTION

Purpose of Review

This document reports the review and evaluation updated through Revision 5, of the Offsite Dose Calculation Manual (ODCM), submitted by the New York Power Authority, the Licensee for the James A. FitzPatrick Nuclear Power Plant (JAFNPP). The ODCM is a supplementary document for implementing the Radiological Effluent Technical Specifications (RETS) in compliance with 10 CFR 50, Appendix I requirements.[1]

Plant-Specific Background

The New York Power Authority submitted to the Nuclear Regulatory Commission (NRC) on October 26, 1983,[2] ODCM Revision 0, dated June 15, 1983 for JAFNPP. The NRC reviewed the ODCM and found it to be, in general, acceptable as stated in letter dated May 29, 1985.[3] Subsequent changes identified in Revisions 1 and 2 were reported to the NRC in the Semiannual Effluent Release Report for July-December 1985.[4] The ODCM, updated through Revision 2 was, in general, approved on an interim basis by the NRC as stated in letter dated October 2, 1986.[5] Changes identified in Revisions 3 and 4 were reported to the NRC in Semiannual Effluent Release Report for January-June 1986,[6] and July-December 1986[7], respectively. Also, Revision 5, dated March 1987, was submitted to the NRC in letter dated April 2, 1987.[8] The NRC transmitted the ODCM updated through Revision 5 to the Idaho National Engineering Laboratory (INEL) for review. The ODCM, updated through Revision 5, was reviewed in its entirety by EG&G Idaho at the INEL. The results and conclusions of the review are presented in this report.

2. REVIEW CRITERIA

Review criteria for the ODCM were provided by the NRC in two documents:

NUREG-0473, RETS for BWRs[9]
NUREG-0133, Preparation of RETS for Nuclear Power Plants.[10]

The following NRC guidelines were also used in the ODCM review: "General Contents of the Offsite Dose Calculation Manual," Revision 1,[11] and Regulatory Guide 1.109, Revision 1.[12]

As specified in NUREG-0473, the ODCM is to be developed by the Licensee to document the methodology and approaches used to calculate offsite doses and to maintain the operability of the radioactive effluent systems. As a minimum, the ODCM should provide equations and methodology for the following:

- Alarm and trip setpoints on effluent instrumentation,
- Liquid effluent concentrations in unrestricted areas,
- Gaseous effluent dose rates at or beyond the site boundary,
- Liquid and gaseous effluent dose contributions,
- Liquid and gaseous effluent dose projections.

In addition, the ODCM should contain flow diagrams that define the treatment paths and the components of the radioactive liquid, gaseous, and solid waste management systems. These flow diagrams should be consistent with the systems being used at the plant. A description and the location of samples in support of the environmental monitoring program are also needed in the ODCM.

3. EVALUATION

JAFNPP is one of three operating BWR units located on the joint Nine Mile Point - James A. FitzPatrick site. The other units are Nine Mile Point Units 1 (NMP-1) and 2 (NMP-2) and are located about 3000 ft to the west of JAFNPP. The ODCM provides the methodology to calculate radiation doses to individuals in the vicinity of JAFNPP. It also provides methodology for calculating effluent monitor setpoints and allowable release rates to ensure compliance with the Technical Specifications, Appendix B, of the New York Power Authority, JAFNPP Docket Number 50-333, and 10 CFR 20 release criteria.

Liquid Effluent Pathways

JAFNPP is located on the southern shore of Lake Ontario in Oswego County, New York, approximately 36 miles northwest of Syracuse, New York. JAFNPP has been in commercial operation since July of 1975. Lake Ontario supplies water to the once-through circulating water system and the service water system which are used as the dilution water for liquid radwaste discharges. All radioactive liquid releases enter Lake Ontario where the circulating water discharge tunnel terminates on the lake bottom approximately 1400 ft from the shoreline.

The liquid radwaste system collects, monitors, and processes all potentially radioactive liquid wastes in a controlled manner. Once the wastes are collected in waste tanks, they are processed by the liquid radwaste treatment system and are discharged from one of the following tanks:

- Floor Drain Sample Tank,
- Waste Sample Tank,
- Laundry Drain Tank.

According to Figure F-5 of the ODCM (Figure 1 of this report) all

liquid radwaste effluent meant for release passes through the monitored liquid radwaste effluent line. According to the text in the ODCM, the radwaste effluents are then diluted with the circulating water and service water before release to Lake Ontario. However, Figure F-5 of the ODCM does not show the point where the liquid radwaste effluent line joins the circulating water and the service water lines for dilution prior to discharge.

A simplified flow diagram of the solid radwaste treatment system is not included in the ODCM.

Liquid Effluent Monitor Setpoints

Sections 3.3.3 and 3.3.4 of the ODCM contain the methodology used to determine the setpoints for the radwaste liquid effluent and service water effluent monitors in compliance with Surveillance Technical Specification Appendix B, Section 2.1.a. There is an automatic termination of release function for the liquid radwaste effluent line as indicated in Table 2.1-1 in Appendix B of the Technical Specification.

The setpoint for the liquid radwaste effluent line monitor is based on the radionuclides identified in each batch of liquid waste prior to release. For continuous releases weekly composite samples are collected continuously in proportion to the rate of flow of the effluent stream and are analyzed. Simultaneous releases are accounted for during both batch and continuous releases.

In Section 3.3.3.b, the equation $S = 0.5 \times C/F_L$ is given to determine the alarm/trip setpoint for the radwaste liquid effluent monitor. However, when $F_L > 0.5$, the value of the setpoint would be less than the expected count rate for the undiluted effluent at concentration C and the monitor would therefore alarm/trip continuously.

In Section 3.3.3.1.g a reference is made to Equation 3.3, however the equation identifier is missing for Equation 3.3 in Section 3.3.3.b.

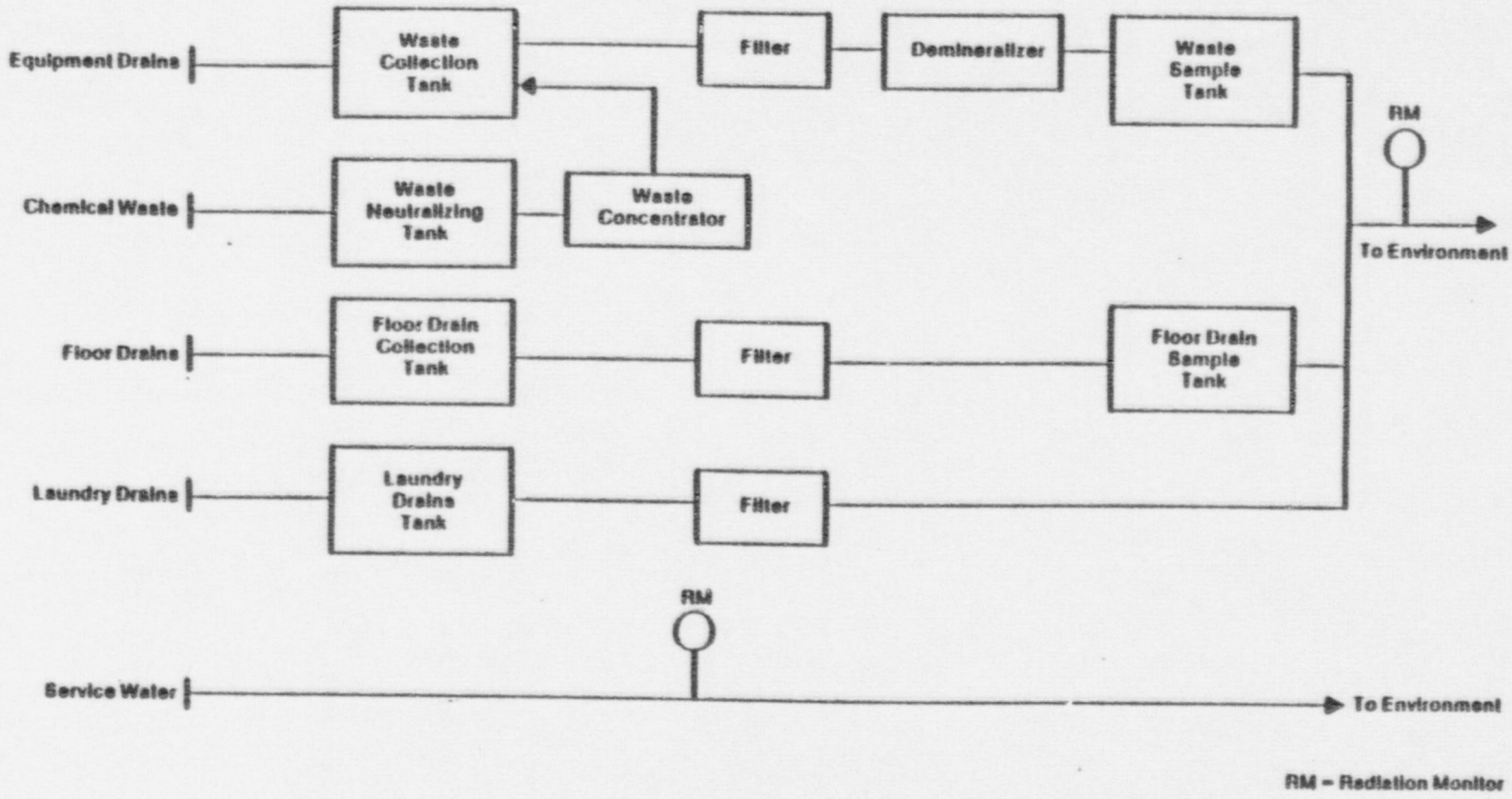

Figure 1. Liquid Radwaste Treatment Systems, Effluent Paths, and Controls for James A. Fitzpatrick Nuclear Power Plant. (Taken from Figure F-5 of the JAFNPP ODCM, Rev. 2 dated June 1985.)

In Section 3.3.4 in Equation 3-3.c a summation over the nuclide index "i" is missing.

With the exception of the issues identified, the methodology for determining the alarm/trip setpoints for the liquid effluent radioactive monitors is within the guidelines of NUREG-0133 and is considered acceptable.

Gaseous Effluent Pathways

According to Technical Specification 3.1.a, there are five gaseous environmental release points at JAFNPP:

- Main Stack,
- Refuel Floor Vent,
- Reactor Building Vent,
- Turbine Building Vent,
- Radwaste Building Vent.

According to Appendix F, Table F-1, in the ODCM, the main stack has a height of 385-ft above grade and is considered an elevated release, whereas the other four are considered ground releases. The Technical Specifications and Appendix F, Table F-1 identify the five gaseous effluent release points listed above. However, Figures F-1, F-2 and F-4 identify only four gaseous release points. Figure F-4 also shows a simplified diagram of the gaseous radwaste treatment system. Figure 2 in this report is a reproduction of Figure F-4. Another diagram is needed however, showing a diagram of the components and pathways for the Standby Gas Treatment System (SBGTS).

Technical Specification Appendix B, Section 3.6.a requires that the offgas treatment system be used to reduce the concentration of radioactive materials in gaseous effluents prior to release from the plant within 24 hrs after startup of the second turbine driven feedwater pump.

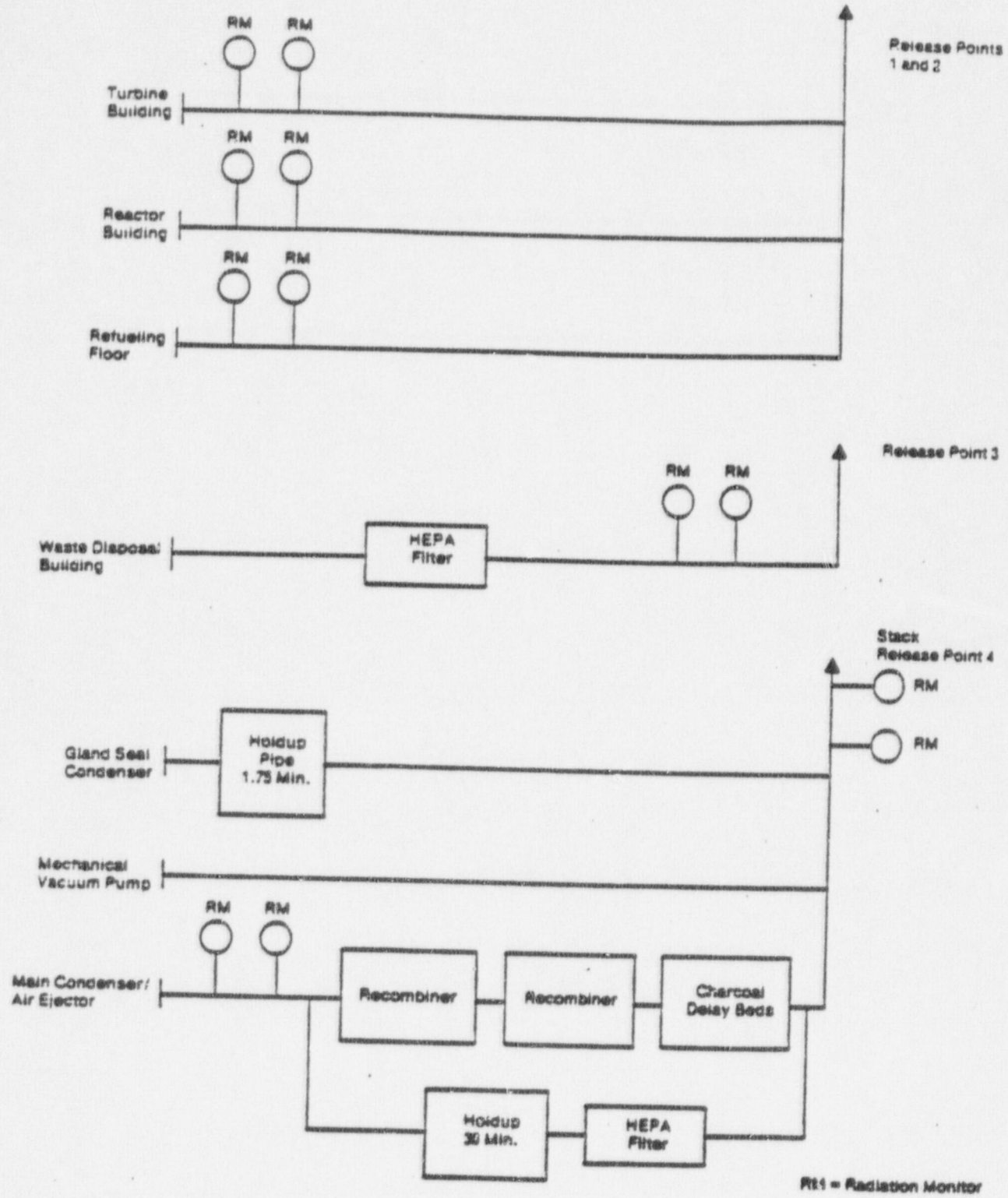

Figure 2. Gaseous Radwaste Treatment Systems, Effluent Paths, and Controls for James A. Fitzpatrick Nuclear Power Plant. (Taken from Figure F-4 of the JAFNPP ODCM, Rev. 2 dated June 1985.)

According to the Bases statement, "this is due to the fact that excess air in-leakages in the main condenser as a result of operating only one turbine driven feedwater pump will exceed offgas treatment system limitations and consequently render the system inoperable. Startup of the second turbine driven feedwater pump will decrease air in-leakage and assure offgas treatment system availability." It is not clear why the feedwater pumps affect the in-leakage rate.

The Technical Specifications identify noble gas monitors, iodine samplers and particulate samplers to monitor gaseous effluent releases. Stack and vent effluents are continuously surveyed during release of noble gases. Iodine and particulate samplers are routinely analyzed in accordance with Technical Specification Appendix B, Table 3.2-1

Gaseous Effluent Monitor Setpoints

Section 4.3.2 of the ODCM contains the methodology used to determine the setpoints for the noble gas effluent monitors as required by Technical Specification Appendix B, Section 3.1.d. Technical Specification Appendix B, Section 3.1.a states that radioactive gaseous wastes released to the environment from each of the five effluent release points shall be monitored and recorded. The monitors for releases from the refuel area exhaust and reactor building area exhaust have automatic isolation for the secondary containment prior to initiation of the SBGTS.

The total body dose rate limit is stated as being more restrictive than the skin dose rate and is used for determining the setpoints for the noble gas monitors. The noble gas monitors maximum setpoints are based on the maximum concentration required to yield the total body dose rate limit. The calculated setpoints account for simultaneous releases.

Technical Specification Appendix B, Section 3.2.a states that the dose rate limits apply to areas "at or beyond the site boundary." Additionally, Figure 5.1-1 in the Technical Specifications shows the site

boundary to include the JAFNPP as well as NMP-1 and NMP-2. Therefore, it appears that the contribution from NMP-1 and NMP-2 must be considered when demonstrating compliance to the dose rate limit of Technical Specification Appendix B Section 3.2.a and not the dose rate from JAFNPP only.

In Section 4.3.2.b.(2) the value for the elevated release rate should be 2.1E+6 μCi/sec instead of 2.4E+6 μCi/sec.

With the exception of the uncertainty of including contributions for NMP-1 and NMP-2 when demonstrating compliance to the dose rate limit of Technical Specification Appendix B Section 3.2.a, the methods for determining the setpoints for the noble gas effluent monitors are, in general, in agreement with the guidelines of NUREG-0133 and are considered acceptable.

Concentrations in Liquid Effluents

Section 3.2.2 contains the methodology for determining the radioactivity concentrations in liquid effluents as required by Surveillance Technical Specification Appendix B, Section 2.2.b. In Section 3.2.2.a the word "percentage" should be removed from the definition for f_2. Regardless, the methodology is within the guidelines of NUREG-0133 and is considered acceptable.

Dose Rates in Gaseous Effluents

Section 4.3.1 contains the methodology for determining dose rates from noble gases to the total body and skin as required by Surveillance Technical Specification Appendix B Section 3.2.a.

In Section 4.3.1.b.(2), the definition for DR_{TB} for Equation 4-5 indicates that this quantity is the dose rate from nuclide i, whereas it in fact is the dose rate to the total body from all noble gas radionuclides.

The definition of DR_{SKIN}, for Equation 4-6, indicates that this quantity is the the dose rate from nuclide i, whereas it is in fact the dose rate to the skin from all noble gas radionuclides. Also, the quantity DR_{SKIN_i} is not defined for Equation 4-6.

In Section 4.3.1.c.(7) in the equation for DR_{SKIN_i} the release rate quantity, $\dot{Q}_i$ is missing.

With the exception of the discrepancies indicated above, the methodology for determining the dose rate due to the release of radioactive noble gases is in agreement with the guidelines of NUREG-0133 and is considered acceptable.

Section 4.3.3.b contains the methodology for determining the instantaneous dose rate due to the release of tritium, iodine-131, iodine-133 and all radionuclides in particulate form with half lives greater than 8 days for the inhalation pathway as required by Surveillance Technical Specification Appendix B, Section 3.2.b.

In Section 4.3.3 the text does not specifically indicate tritium as required in Technical Specification Appendix B, Section 3.2.b.

In Section 4.3.3.b, in Equation 4-9 the quantity $\dot{Q}_i$ should be the total release rate for nuclide i from all release points in order to determine a total dose rate. Also, in Equation 4-10, which determines the limiting iodine release rate, there should be a summation due to all radionuclides.

With the exception of the indicated discrepancies, the methodology for determining the dose rates due to the release of tritium, iodine-131, iodine-133 and all radionuclides in particulate form with half lives greater than 8 days is in agreement with the guidelines of NUREG-0133 and is considered acceptable.

Dose Due to Liquid Effluents

Section 3.4.1.b contains the methodology for determining the dose due to radioactive material released in liquid effluents as required by Surveillance Technical Specification Appendix B, Section 2.3.a.

JAFNPP identifies fish and potable water pathways for the dose calculations, designating the adult as the maximum exposed individual and the total body and liver as the critical organs.

In Section 3.4.1.b the quantity Δt_1 in Equation 3-4 should be the number of hours in the calendar year instead of the number of hours during liquid effluent releases for the calendar year. Also, a summation of the dose due to all radionuclides is missing from Equation 3-4.

In Section 3.4.1.b the reference for the derivation of the quantity $A_{i\tau}$ should be Appendix A, Table L-4.1 instead of Appendix A.

In Section 3.4.2.b.(1) the definition of Δt_1 for Equation 3-5 should be the number of hours of the reporting period (e.g. monthly, quarterly, annually) and not the number of hours over which the release occurs. Also, the quantity $(DF)_1$ should be the total volume of dilution released during the reporting period. A summation of the dose due to all radionuclides should be included in Equation 3-5.

The methodology in Section 3.4.2.c should be made consistent with the previous comments in this review concerning the definition of time period Δt_1 identified in Sections 3.4.1.b and 3.4.2.b.(1).

Appendix A, Table L-1 lists the maximum permissible concentrations in water in unrestricted areas. The following discrepancies were detected by the reviewer:

- The first Zn-69 should be listed as Zn-65.

- The value for insoluble Y-93 should be 3E-4 μCi/ml instead of 3E-5 μCi/ml.

Appendix A, Tables L-2 and L-3 contain liquid dose conversion factors, $A_{i\tau}$ for all age groups, pathways and radionuclides. Using the methodology given in Appendix A, Table L-4.1 the reviewer (with the exception of the boating pathway) was unable to reproduce a large fraction of the dose conversion factors. It is recommended that the Licensee ensure the validity of each and every dose conversion factor.

Appendix A, Table L-4.1 gives a derivation of the liquid dose factor $A_{i\tau}$. The following errors were detected by the reviewer:

- In the definition for Z in Table L-4.1, the quantity M_c is not defined.

- The units for the quantity λ_i should be sec^{-1} instead of secs.

- A reference should be provided supporting the quantities Dw, and Df in Equation L-4.1.

- In the calculational example for swimming, the equation referred to should be Equation L-4.1 instead of Equation L-4.4. Also, for the quantity DFi the table referred to should be Table L-4.4 instead of Table L-4-2.

- The purpose for Table L-4.2 is unclear since all the activity values are unity and the table is not referred to in the ODCM.

- The following tables are not referred to in the ODCM: Tables L-4.3, L-5, L-5.1, and L-5.2.

- The ODCM states that the dose conversion factors tabulated in Table L-4.4 for swimming and boating are generated using the

Impell computer code. Either the methodology and parameters contained in the Impell code should be submitted for review or the Licensee should include the reference containing NRC approval for use of this code.

Primarily due to the incorrect definition for Δt_1 and the values in the dose factor tables, the methodology for calculating the dose due to the release of radioactivity in liquid effluents is not in agreement with the guidelines of NUREG-0133 and is not considered acceptable.

Dose Due to Gaseous Effluents

Sections 4.4.1.b.(1), 4.4.2.a and 4.4.2.b contain the methodologies for calculating the cumulative gamma and beta air doses due to the release of radioactive noble gases as required by Surveillance Technical Specification Appendix B, Section 3.3.a.

In Section 4.4.2.a.(2)(a), the reference for the detailed explanation and evaluation of the quantity M_{eff} should be Appendix E instead of Appendix C. Also, the value for (X/Q) on Page 42 of the ODCM, for the elevated release should be 3.8E-8 sec/m^3 instead of 3.8E-7 sec/m^3.

In Section 4.4.2.b.(2) the second half of the first paragraph is evidently out of place since it discusses iodines and particulates in a section describing noble gas beta air dose methodology.

With the exception of these two discrepancies, the method for calculating the air dose due to the release of radioactive noble gases is in agreement with the guidelines of NUREG-0133 and is considered acceptable.

Sections 4.4.1.b.(2) and 4.4.2.c contain the methodologies for calculating the cumulative dose due to the release of I-131, I-133, tritium, and radionuclides in particulate form with half-lives greater than eight days as required by Surveillance Technical Specification

Appendix B, Section 3.4.a.

In Sections 4.4.1.b.(2) and 4.4.2.c, tritium has been omitted from the radionuclides included for determining cumulative dose but is required by Technical Specification Appendix B, Section 3.4.b.

In Section 4.4.1.b.(2) the reference for the derivation of $R_{i\tau}$, should be to Appendix B, Table G-7.1B instead of Appendix B, Table G-8.

In Section 4.4.2.c.(2) the subscript D should be removed from the quantity (D/Q) in Equations 4-25 and 4-26.

Appendix B, Tables G-4, G-5, and G-6 contain gaseous dose conversion factors, $P_{i\tau}$ and $R_{i\tau}$ for all age groups, pathways, and radionuclides. Using the methodology given in Appendix A, Table G-7 the reviewer was unable to reproduce a large fraction of the dose conversion factors. It is recommended that the Licensee check the validity of each and every dose conversion factor.

Appendix B, Tables G-7 gives a derivation of gaseous dose factors, P_i and R_i. The following discrepancies were detected by the reviewer:

- In Table G-7.1A tritium has been omitted from the radionuclides included for determining the inhalation dose factor but is required by the Technical Specifications.

- In Table G-7.1B methodology for determining the gaseous dose factors due to tritium in the cow-milk, cow-meat, goat-milk and fruit-vegetable pathways is missing.

- In Table G-7.1B, Section I the quantity (DFAi) should be written $(DFAi)_a$ to denote the age group.

- In Table G-7.1B, Section III in the example calculation, the quantity $(DFLi)_i$ should be written $(DFLi)_a$.

- In Table G-7.1B the R_i factor values determined in the example calculations for Co-60 (child - total body) for all pathways except inhalation are referenced as being listed in Appendix B, Table G-6 in the adult portion instead of the child portion of the table.

- The purpose of Table G-7.2 is unclear since all the activity values are unity and the table is not referred to in the ODCM.

- The purpose for some of the pages in Table G-7.3 is not clear and also the table is not referred to in the ODCM.

The methodology for calculating the cumulative dose due to the release of I-131, I-133, tritium, and radionuclides in particulate form with half-lives greater than eight days is in agreement with the guidelines of Regulatory Guide 1.109, Revision 1 and NUREG-0133 and is considered acceptable. However, since the reviewer was unable to verify the dose factors it is uncertain if the methodology can be used to demonstrate compliance to the Technical Specifications.

Dose Projections

Section 3.5.2 of the ODCM describes the method used to project doses due to the expected releases of radioactive liquid effluents to determine when the liquid radwaste treatment system should be operated as required in Surveillance Technical Specification Appendix B, Section 2.4.a. The methodology for determining the dose projection due to liquid radwaste effluents is in agreement with NUREG-0133 and is considered acceptable.

Section 4.5.2 of the ODCM describes the method used to project doses due to release of radioactive gases when the offgas treatment system is not in use as required in Surveillance Technical Specification Appendix B, Section 3.6.a. It does not appear that there are ventilation exhaust treatment systems and the gaseous radwaste treatment system is in use whenever the second steam driven feedwater pump is in operation.

Therefore, it is not clear why dose projections are required by the Technical Specifications since the purpose for the dose projections for a BWR is to determine required use of the ventilation exhaust treatment systems. Section 4.5.2.a refers to the limits of Technical Specification Appendix B, Section 3.6.a whereas no limits are found in this Technical Specification. Regardless, the methods for determining dose projections are within the guidelines of NUREG-0133 and are considered acceptable.

Total Dose

Section 5.0 of the ODCM contains the methodology for calculating the total dose contributions including direct radiation as required by Surveillance Technical Specification Appendix B, Section 5.1.a.

In Section 5.1 in the first sentence of the first paragraph the text should read: "Dose evaluation to demonstrate compliance with the 40 CFR 190 dose limits need only be performed if the quarterly or annual doses ... exceed twice the dose limits of Technical Specifications Appendix B, Sections 2.3.a, 3.3.a or 3.4.a respectively", in order to be consistent with the Technical Specifications.

With the exception of the indicated discrepancy, the methodology is in agreement with the requirements of the Technical Specifications and is considered acceptable.

Environmental Monitoring Program

Table H-1 in Appendix H of the ODCM contains specific parameters of distance and the direction sector from the site and additional information for each and every sample identified in Surveillance Technical Specification Section 6.1. The environmental monitoring program is in compliance with Technical Specification Appendix B, Table 6.1-1 and is considered acceptable.

Summary

In summary, the Licensee's ODCM uses documented and approved methods that are, in general, consistent with the methodology and guidance in NUREG-0133 and Regulatory Guide 1.109, Revision 1. However, due primarily to the incorrect definition of Δt_1 and the erroneous values for the liquid and gaseous dose factors it is recommended that the NRC request another revision.

4. CONCLUSIONS

The Licensee's ODCM updated through Revision 5, dated March 1987 for JAFNPP was reviewed. It was determined that the ODCM uses methods that are, in general, consistent with the guidelines of NUREG-0133. The methodology in most sections of the ODCM is acceptable for use in demonstrating compliance to the radiological effluent technical specifications. However, it is recommended that another revision to the ODCM be submitted to address the discrepancies identified in the review.

- In Section 3.2.2.a the word "percentage" should be removed from the definition for f_2.

- In Section 3.3.3.b, the equation $S = 0.5 \times C/F_L$ is given to determine the alarm/trip setpoint for the radwaste liquid effluent monitor. However, when $F_L > 0.5$, the value of the setpoint would be less than the expected count rate for the undiluted effluent at concentration C and the monitor would therefore alarm/trip continuously.

- In Section 3.3.3.1.g a reference is made to Equation 3.3, however the equation identifier is missing for Equation 3.3 in Section 3.3.3.b.

- In Section 3.3.4, a summation over the nuclide index "i" in Equation 3-3.c, is missing.

- In Section 3.4.1.b the reference for the derivation of the quantity A_{iT} should be more specifically Appendix A, Table L-4.1 instead of Appendix A.

- In Section 3.4.1.b the quantity Δt_1 in Equation 3-4 should be the number of hours in the calendar year instead of the number of hours during liquid effluent releases for the calendar year.

Also, a summation of the dose due to all radionuclides is missing from Equation 3-4.

- In Section 3.4.2.b.(1) the definition of Δt_1 for Equation 3-5 should be the number of hours of the reporting period (e.g. monthly, quarterly, annually) and not the number of hours over which the release occurs. Also, the quantity $(DF)_1$ should be the total volume of dilution released during the reporting period. A summation of the dose due to all radionuclides should be included in Equation 3-5.

- The methodology in Section 3.4.2.c should be made to be consistent with the previous comments in this review concerning the definition of the time period Δt_1 identified in Sections 3.4.1.b and 3.4.2.b.(1).

- A simplified flow diagram of the solid radwaste treatment system is not included in the ODCM.

- In Section 4.3.1.b.(2), the definition for DR_{TB} for Equation 4-5 indicates that this quantity is the dose rate from nuclide i, whereas it in fact is the dose rate to the total body from <u>all</u> noble gas radionuclides.

- The definition of DR_{SKIN}, for Equation 4-6, indicates that this quantity is the the dose rate from nuclide i, whereas it is in fact the dose rate to the skin from <u>all</u> noble gas radionuclides.

- In Section 4.3.1.b.(2), the quantity DR_{SKIN_i} is not defined for Equation 4-6.

- In Section 4.3.1.c.(7), in the equation for DR_{SKIN_i}, the parameter $\dot{Q}$ is missing.

- Technical Specification Appendix B, Section 3.2.a states that the dose rate limits apply to areas "at or beyond the site boundary." Additionally, Figure 5.1-1 in the Technical Specifications shows the site boundary to include the JAFNPP as well as NMP-1 and NMP-2. Therefore, it appears that the contribution from NMP-1 and NMP-2 must be considered when demonstrating compliance to the dose rate limit of Technical Specification Appendix B Section 3.2.a and not the dose rate from JAFNPP only.

- In Section 4.3.2.b.(2) the value for the elevated release rate should be 2.1E+6 μCi/sec instead of 2.4E+6 μCi/sec.

- In Section 4.3.3 the text does not specifically indicate tritium as required in Technical Specification Appendix B, Section 3.2.a.b.

- In Section 4.3.3.b, in Equation 4-9 the quantity $\dot{Q}_i$ should be the total release rate for nuclide i from all release points, in order to determine a total dose rate.

- In Section 4.3.3.b in Equation 4-10, which determines the limiting iodine release rate, there should be a summation due to all radionuclides.

- In Sections 4.4.1.b.(2) and 4.4.2.c, tritium has been omitted from the radionuclides included for determining cumulative dose but is required by Technical Specification Appendix B, Section 3.4.b.

- In Section 4.4.1.b.(2) the reference for the derivation of R_{iT}, should be to Appendix B, Table G-7.1B instead of Appendix B, Table G-8.

- In Section 4.4.2.a.(2)(a), the reference for the detailed explanation and evaluation of the quantity M_{eff} should be Appendix E instead of Appendix C.

- In Section 4.4.1.a.(2) the value for (X/Q) on Page 42 of the ODCM, for the elevated release should be 3.8E-8 sec/m^3 instead of 3.8E-7 sec/m^3.

- In Section 4.4.2.b.(2) the second half of the first paragraph seems to be displaced since it discusses iodines and particulates in a section describing noble gas beta air dose methodology.

- In Section 4.4.2.c.(2) the subscript D should be removed from the quantity (D/Q) in Equations 4-25 and 4-26.

- Section 4.5.2.a refers to the limits of Technical Specification Appendix B, Section 3.6.a however, no limits are found in this Technical Specification.

- In Section 5.1 in the first sentence of the first paragraph the text should read: "Dose evaluation to demonstrate compliance with the 40 CFR 190 dose limits need only be performed if the quarterly or annual doses ... exceed twice the dose limits of Technical Specifications Appendix B, Sections 2.3.a, 3.3.a or 3.4.a respectively", in order to be consistent with the Technical Specifications.

- Appendix A, Table L-1 lists the maximum permissible concentrations in water in unrestricted areas. The following discrepancies were detected by the reviewer:

  - The first Zn-69 should be listed as Zn-65.

  - The value for insoluble Y-93 should be 3E-4 μCi/ml instead of 3E-5 μCi/ml.

- Appendix A, Tables L-2 and L-3 contain liquid dose conversion factors, $A_{i\tau}$ for all age groups, pathways and radionuclides. Using the methodology given in Appendix A, Table L-4.1 the reviewer (with the exception of the boating pathway) was unable to reproduce a large fraction of the dose conversion factors. It is recommended that the Licensee ensure the validity of each and every dose conversion factor.

- Appendix A, Table L-4.1 gives a derivation of the liquid dose factor $A_{i\tau}$. The following errors were detected by the reviewer:

  - In the definition for Z in Table L-4.1, the quantity M_c is not defined.

  - The units for the quantity λ_i should be sec^{-1} instead of secs.

  - A reference should be provided for the quantities Dw, and Df in Equation L-4.1.

  - In the calculational example for swimming, the equation referred to should be Equation L-4.1 instead of Equation L-4.4. Also, for the quantity DFi the table referred to should be Table L-4.4 instead of Table L-4-2.

  - The purpose for Table L-4.2 is unclear since all the activity values are unity and the table is not referred to on the ODCM.

  - The following tables are not referred to in the ODCM: Tables L-4.3, L-5, L-5.1, and L-5.2.

  - The ODCM states that the dose conversion factors tabulated in Table L-4.4 for swimming and boating are generated using

the Impell computer code. Either the methodology and parameters contained in the Impell code should be submitted for review or the Licensee should include the reference containing NRC approval for use of this code.

- Appendix B, Tables G-4, G-5, and G-6 contain gaseous dose conversion factors, P_{i_7} and R_{i_7} for all age groups, pathways, and radionuclides. Using the methodology given in Appendix A, Table G-7 the reviewer was unable to reproduce a large fraction of the dose conversion factors. It is recommended that the Licensee check the validity of each and every dose conversion factor.

- Appendix B, Tables G-7 gives a derivation of gaseous dose factors, P_i and R_i. The following discrepancies were detected by the reviewer:

  - In Table G-7.1A tritium has been omitted from the radionuclides included for determining the inhalation dose factor but is required by the Technical Specifications.

  - In Table G-7.1B methodology for determining the gaseous dose factors due to tritium in the cow-milk, cow-meat, goat-milk and fruit-vegetable pathways is missing.

  - In Table G-7.1B, Section I the quantity (DFAi) should be written $(DFAi)_a$ to denote the age group.

  - In Table G-7.1B, Section III in the example calculation, the quantity $(DFLi)_i$ should be written $(DFLi)_a$.

  - In Table G-7.1B the R_i factor values determined in the example calculations for Co-60 (child - total body) for all pathways except inhalation are referenced as being listed in Appendix B, Table G-6 in the adult portion instead of the

child portion of the table.

- The purpose of Table G-7.2 is unclear since all the activity values are unity and the table is not referred to in the ODCM.

- The purpose for some of the pages in Table G-7.3 is not clear and also the table is not referred to in the ODCM.

- The Technical Specifications and Appendix F, Table F-1 identify the five gaseous effluent release points listed above. However, Figures F-1, F-2 and F-4 identify only four gaseous release points.

- Figure F-4 also shows a simplified diagram of the gaseous radwaste treatment system. Another diagram is needed however, showing a diagram of the components and pathways for the Standby Gas Treatment System (SBGTS).

- In Figure F-5 of the ODCM the diagram showing the liquid effluent release paths does not show the point where the liquid radwaste effluent line joins the circulating water and the service water lines for dilution prior to discharge.

5. REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion, 'As Low As Is Reasonably Achievable,' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

2. Letter from J. P. Bayne (New York Power Authority) to D. B. Vassallo (NRC), Subject: James A. FitzPatrick Nuclear Power Plant, Off-Site Dose Calculation Manual, October 26, 1983.

3. Letter from H. I. Abelson (NRC) to J. P. Bayne (New York Power Authority), Subject: Approval of James A. FitzPatrick Nuclear Power Plant ODCM, May 29, 1985.

4. Letter from R. J. Converse (New York Power Authority) to T. E. Murley (NRC), Subject: J. A. FitzPatrick Nuclear Power Plant, Semiannual Radioactive Effluent Release Report, February 28, 1986.

5. Letter from H. I. Abelson (NRC) to J. C. Brons (New York Power Authority), Subject: Revision to the Offsite Dose Calculation Manual (ODCM), October 2, 1986.

6. Letter from R. J. Converse (New York Power Authority) to T. E. Murley (NRC), Subject: J. A. FitzPatrick Nuclear Power Plant, Semiannual Radioactive Effluent Release Report, August 29, 1986.

7. Letter from R. J. Converse (New York Power Authority) to T. E. Murley (NRC), Subject: J. A. FitzPatrick Nuclear Power Plant, Semiannual Radioactive Effluent Release Report, February 27, 1987.

8. Letter from R. J. Converse (New York Power Authority) to T. E. Murley (NRC), Subject: J. A. FitzPatrick Nuclear Power Plant, Docket No. 50-333, License No. DPR-59, Revision to the Off-Site Dose Calculation Manual (ODCM), April 2, 1987.

9. "Standard Radiological Effluent Technical Specifications for Boiling Water Reactors," Rev. 3, Draft 7", intended for contractor guidance in reviewing RETS proposals for operating reactors, NUREG-0473, September 1982.

10. "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, A Guidance Manual for Users of Standard Technical Specifications," NUREG-0133, October 1978.

11. "General Contents of the Offsite Dose Calculation Manual," Revision 1, Branch Technical Position, Radiological Assessment Branch, NRC, February 8, 1979.

12. "Calculation of Annual Doses of Evaluating Compliance with 10 CFR 50, Appendix I," Regulatory Guide 1.109, Rev. 1, October 1977.

NRC FORM 336 (2-84) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE.

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MONTH: July YEAR: 1988

5 AUTHOR(S)

T. S. Bohn, W. Serrano, A. L. Freeman, M. R. Winberg

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13 ABSTRACT (200 words or less)

The Offsite Dose Calculation Manual for the James A. FitzPatrick Nuclear Power Plant contains a description of the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitor alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program. The NRC transmitted the ODCM updated through Revision 5 to the Idaho National Engineering Laboratory for review. The ODCM, complete through Revision 5, was reviewed in its entirety by EG&G Idaho at the INEL and the results of the review are presented in this report. It was determined that the ODCM updated through Revision 5 uses methods that are, in general, in agreement with the guidelines of NUREG-0133. However, it is recommended that another revision to the ODCM be submitted to address the discrepancies identified in the review.

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Mr. John C. Brons
Executive Vice President, Nuclear Generation
Power Authority of the State of New York
123 Main Street
White Plains, New York 10601

Dear Mr. Brons:

SUBJECT: JAMES A. FITZPATRICK NUCLEAR POWER PLANT, REVISION 5 TO THE OFFSITE DOSE CALCULATION MANUAL (TAC NO. 65216)

REFERENCES: (a) October 2, 1986 letter from H. I. Abelson to J. C. Brons
(b) April 2, 1987 letter from R. J. Converse to T. E. Murley

Reference (a) documented staff acceptance with certain caveats of Revision 2 to the Offsite Dose Calculation Manual (ODCM) for the James A. FitzPatrick Nuclear Power Plant. In Reference (b) you submitted a complete Revision 5 to the ODCM and provided a response to questions raised in Reference (a).

Revision 5 to the ODCM has been reviewed for us in its entirety by our contractor, EG&G Idaho, Inc. Their discussion and comments are contained in the attached TER, EGG-PHY-8153. We agree with the contractor that Revision 5 to the ODCM uses methods that are, in general, consistent with staff guidelines and is an acceptable interim reference. However, there are a number of discrepancies and suggestions listed in the TER, some typographical in nature, others dealing with technical issues. Since these questions involve refinements to a basically sound document, there is no need for immediate action.

However, we ask that all of the points raised in the Conclusions section of the TER be addressed when the next revision to the ODCM is issued or at least prior to startup for the 1990 Refueling Outage. It is not necessary to submit the ODCM revision to us for review, but a record of the disposition of each item should be available for an audit.

This closes TAC No. 65216.

Sincerely,

ORIGINAL SIGNED BY
David E. LaBarge, Project Director
Project Directorate I-1
Division of Reactor Projects I/II

Enclosure:
Technical Evaluation Report

cc: See next page

[REVISION 5 TAC 65216]

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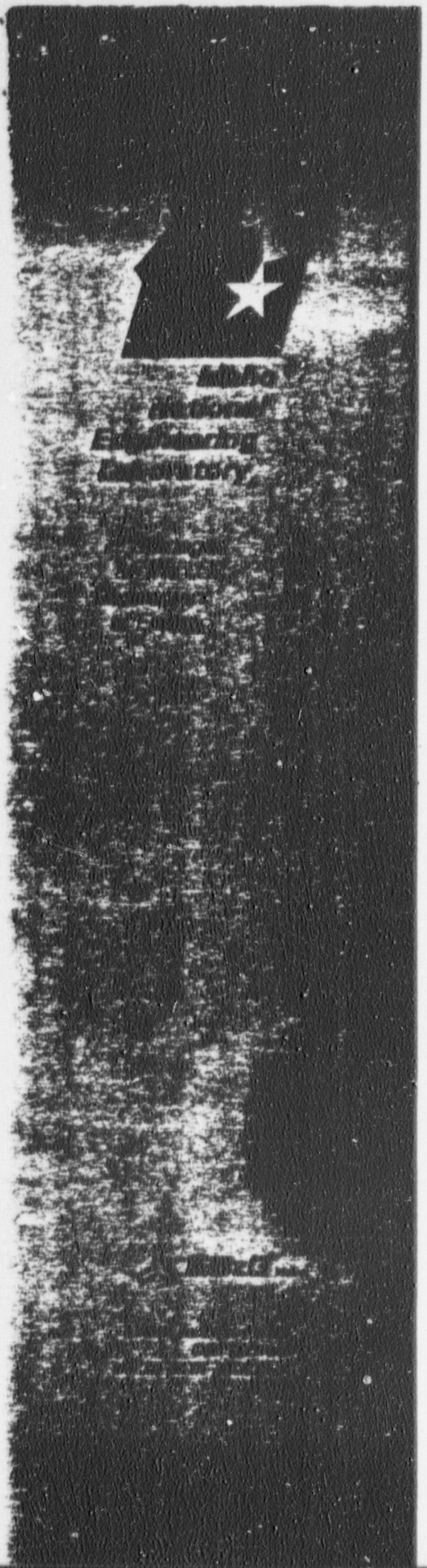

EGG-PHY-8153
July 1988

TECHNICAL EVALUATION REPORT

TECHNICAL EVALUATION REPORT FOR THE EVALUATION OF ODCM UPDATED THROUGH REVISION 5 JAMES A. FITZPATRICK NUCLEAR POWER PLANT

T. S. Bohn
W. Serrano
A. L. Freeman
M. R. Winberg

Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION

~~8905120213~~ 32pp.

EGG-PHY-8153

TECHNICAL EVALUATION REPORT

for the

EVALUATION OF ODCM UPDATED THROUGH REVISION 5

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

NRC Docket No. 50-333 NRC License No. DPR-59

T. S. Bohn
W. Serrano
A. L. Freeman
M. R. Winberg

Published July 1988

Idaho National Engineering Laboratory
EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

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ABSTRACT

The Offsite Dose Calculation Manual for the James A. FitzPatrick Nuclear Power Plant (JAFNPP) contains a description of the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitor alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program. Revision 0, dated June 15, 1983, was submitted to the NRC with letter dated October 26, 1983 and was, in general, approved by the NRC on May 29, 1985. Subsequent changes identified in Revisions 1 and 2 were reported to the NRC in the Semiannual Effluent Release Report for July-December 1985. The ODCM, updated through Revision 2 was, in general, approved on an interim basis by the NRC on October 2, 1986. Changes identified in Revisions 3 and 4 were reported to the NRC in Semiannual Effluent Release Report for January-July 1986 and July-December 1986 respectively. Also, Revision 5, dated March 1987, was submitted to the NRC in letter dated April 2, 1987. The NRC transmitted the ODCM updated through Revision 5 to the Idaho National Engineering Laboratory (INEL) for review. The ODCM, complete through Revision 5, was reviewed in its entirety by EG&G Idaho at the INEL and the results of the review are presented in this report. It was determined that the ODCM updated through Revision 5 uses methods that are, in general, in agreement with the guidelines of NUREG-0133. However, it is recommended that another revision to the ODCM be submitted to address the discrepancies identified in the review.

FOREWORD

This report is submitted as partial fulfillment of the "Review of Radiological Issues" project being contracted by the Idaho National Engineering Laboratory for the U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation. The U. S. Nuclear Regulatory Commission funded the work under FIN D6034 and NRC B&R Number 20 19 05 03.

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1. INTRODUCTION

Purpose of Review

This document reports the review and evaluation updated through Revision 5, of the Offsite Dose Calculation Manual (ODCM), submitted by the New York Power Authority, the Licensee for the James A. FitzPatrick Nuclear Power Plant (JAFNPP). The ODCM is a supplementary document for implementing the Radiological Effluent Technical Specifications (RETS) in compliance with 10 CFR 50, Appendix I requirements.[1]

Plant-Specific Background

The New York Power Authority submitted to the Nuclear Regulatory Commission (NRC) on October 26, 1983,[2] ODCM Revision 0, dated June 15, 1983 for JAFNPP. The NRC reviewed the ODCM and found it to be, in general, acceptable as stated in letter dated May 29, 1985.[3] Subsequent changes identified in Revisions 1 and 2 were reported to the NRC in the Semiannual Effluent Release Report for July-December 1985.[4] The ODCM, updated through Revision 2 was, in general, approved on an interim basis by the NRC as stated in letter dated October 2, 1986.[5] Changes identified in Revisions 3 and 4 were reported to the NRC in Semiannual Effluent Release Report for January-June 1986,[6] and July-December 1986[7], respectively. Also, Revision 5, dated March 1987, was submitted to the NRC in letter dated April 2, 1987.[8] The NRC transmitted the ODCM updated through Revision 5 to the Idaho National Engineering Laboratory (INEL) for review. The ODCM, updated through Revision 5, was reviewed in its entirety by EG&G Idaho at the INEL. The results and conclusions of the review are presented in this report.

2. REVIEW CRITERIA

Review criteria for the ODCM were provided by the NRC in two documents:

NUREG-0473, RETS for BWRs[9]
NUREG-0133, Preparation of RETS for Nuclear Power Plants.[10]

The following NRC guidelines were also used in the ODCM review: "General Contents of the Offsite Dose Calculation Manual," Revision 1,[11] and Regulatory Guide 1.109, Revision 1.[12]

As specified in NUREG-0473, the ODCM is to be developed by the Licensee to document the methodology and approaches used to calculate offsite doses and to maintain the operability of the radioactive effluent systems. As a minimum, the ODCM should provide equations and methodology for the following:

- Alarm and trip setpoints on effluent instrumentation,
- Liquid effluent concentrations in unrestricted areas,
- Gaseous effluent dose rates at or beyond the site boundary,
- Liquid and gaseous effluent dose contributions,
- Liquid and gaseous effluent dose projections.

In addition, the ODCM should contain flow diagrams that define the treatment paths and the components of the radioactive liquid, gaseous, and solid waste management systems. These flow diagrams should be consistent with the systems being used at the plant. A description and the location of samples in support of the environmental monitoring program are also needed in the ODCM.

3. EVALUATION

JAFNPP is one of three operating BWR units located on the joint Nine Mile Point - James A. FitzPatrick site. The other units are Nine Mile Point Units 1 (NMP-1) and 2 (NMP-2) and are located about 3000 ft to the west of JAFNPP. The ODCM provides the methodology to calculate radiation doses to individuals in the vicinity of JAFNPP. It also provides methodology for calculating effluent monitor setpoints and allowable release rates to ensure compliance with the Technical Specifications, Appendix B, of the New York Power Authority, JAFNPP Docket Number 50-333, and 10 CFR 20 release criteria.

Liquid Effluent Pathways

JAFNPP is located on the southern shore of Lake Ontario in Oswego County, New York, approximately 36 miles northwest of Syracuse, New York. JAFNPP has been in commercial operation since July of 1975. Lake Ontario supplies water to the once-through circulating water system and the service water system which are used as the dilution water for liquid radwaste discharges. All radioactive liquid releases enter Lake Ontario where the circulating water discharge tunnel terminates on the lake bottom approximately 1400 ft from the shoreline.

The liquid radwaste system collects, monitors, and processes all potentially radioactive liquid wastes in a controlled manner. Once the wastes are collected in waste tanks, they are processed by the liquid radwaste treatment system and are discharged from one of the following tanks:

- Floor Drain Sample Tank,
- Waste Sample Tank,
- Laundry Drain Tank.

According to Figure F-5 of the ODCM (Figure 1 of this report) all

liquid radwaste effluent meant for release passes through the monitored liquid radwaste effluent line. According to the text in the ODCM, the radwaste effluents are then diluted with the circulating water and service water before release to Lake Ontario. However, Figure F-5 of the ODCM does not show the point where the liquid radwaste effluent line joins the circulating water and the service water lines for dilution prior to discharge.

A simplified flow diagram of the solid radwaste treatment system is not included in the ODCM.

Liquid Effluent Monitor Setpoints

Sections 3.3.3 and 3.3.4 of the ODCM contain the methodology used to determine the setpoints for the radwaste liquid effluent and service water effluent monitors in compliance with Surveillance Technical Specification Appendix B, Section 2.1.a. There is an automatic termination of release function for the liquid radwaste effluent line as indicated in Table 2.1-1 in Appendix B of the Technical Specification.

The setpoint for the liquid radwaste effluent line monitor is based on the radionuclides identified in each batch of liquid waste prior to release. For continuous releases weekly composite samples are collected continuously in proportion to the rate of flow of the effluent stream and are analyzed. Simultaneous releases are accounted for during both batch and continuous releases.

In Section 3.3.3.b, the equation $S = 0.5 \times C/F_L$ is given to determine the alarm/trip setpoint for the radwaste liquid effluent monitor. However, when $F_L > 0.5$, the value of the setpoint would be less than the expected count rate for the undiluted effluent at concentration C and the monitor would therefore alarm/trip continuously.

In Section 3.3.3.1.g a reference is made to Equation 3.3, however the equation identifier is missing for Equation 3.3 in Section 3.3.3.b.

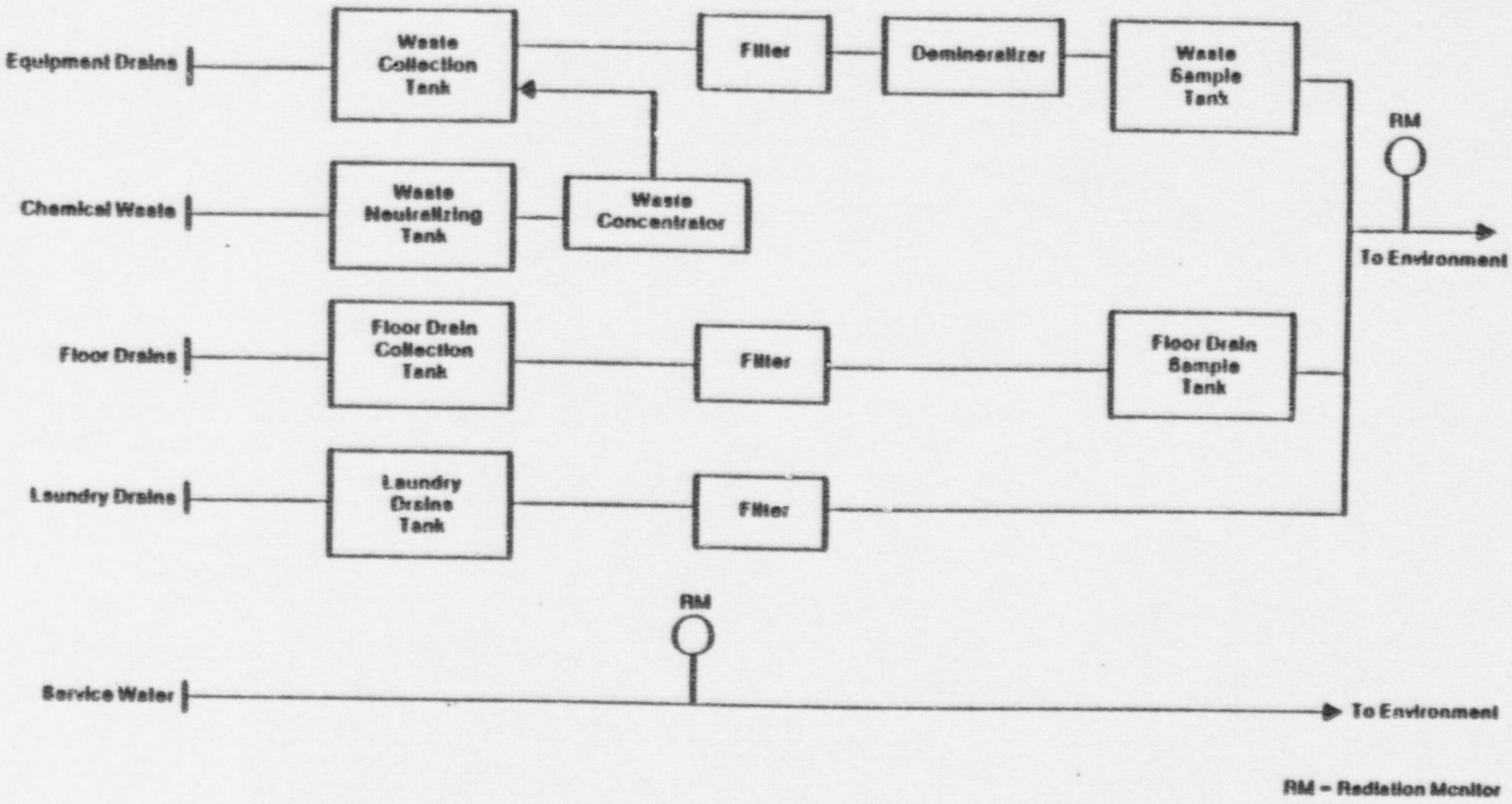

Figure 1. Liquid Radwaste Treatment Systems, Effluent Paths, and Controls for James A. Fitzpatrick Nuclear Power Plant. (Taken from Figure F-5 of the JAFNPP ODCM, Rev. 2 dated June 1985.)

In Section 3.3.4 in Equation 3-3.c a summation over the nuclide index "i" is missing.

With the exception of the issues identified, the methodology for determining the alarm/trip setpoints for the liquid effluent radioactive monitors is within the guidelines of NUREG-0133 and is considered acceptable.

Gaseous Effluent Pathways

According to Technical Specification 3.1.a, there are five gaseous environmental release points at JAFNPP:

- Main Stack,
- Refuel Floor Vent,
- Reactor Building Vent,
- Turbine Building Vent,
- Radwaste Building Vent.

According to Appendix F, Table F-1, in the ODCM, the main stack has a height of 385-ft above grade and is considered an elevated release, whereas the other four are considered ground releases. The Technical Specifications and Appendix F, Table F-1 identify the five gaseous effluent release points listed above. However, Figures F-1, F-2 and F-4 identify only four gaseous release points. Figure F-4 also shows a simplified diagram of the gaseous radwaste treatment system. Figure 2 in this report is a reproduction of Figure F-4. Another diagram is needed however, showing a diagram of the components and pathways for the Standby Gas Treatment System (SBGTS).

Technical Specification Appendix B, Section 3.6.a requires that the offgas treatment system be used to reduce the concentration of radioactive materials in gaseous effluents prior to release from the plant within 24 hrs after startup of the second turbine driven feedwater pump.

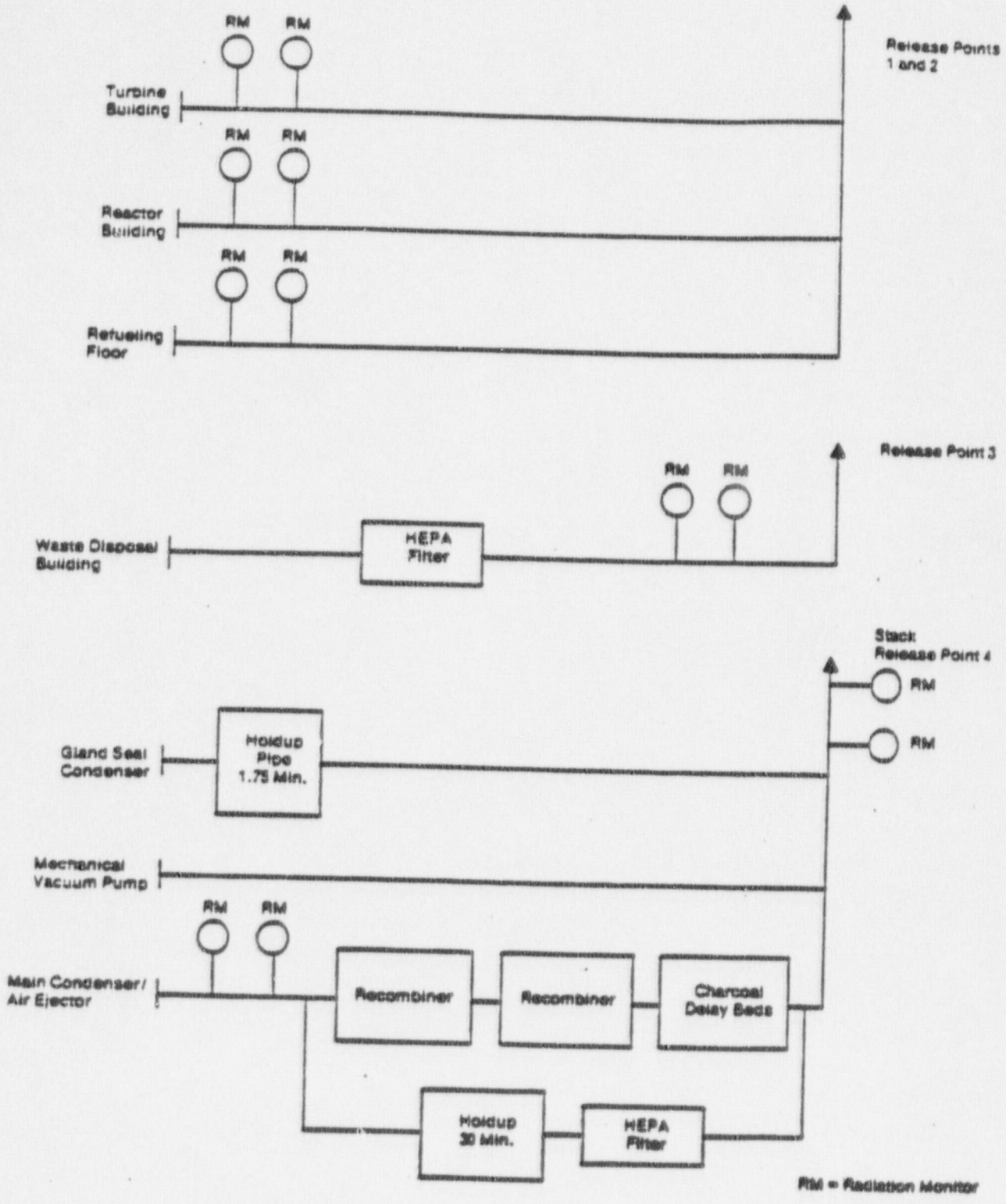

Figure 2. Gaseous Radwaste Treatment Systems, Effluent Paths, and Controls for James A. Fitzpatrick Nuclear Power Plant. (Taken from Figure F-4 of the JAFNPP ODCM, Rev. 2 dated June 1985.)

According to the Bases statement, "this is due to the fact that excess air in-leakages in the main condenser as a result of operating only one turbine driven feedwater pump will exceed offgas treatment system limitations and consequently render the system inoperable. Startup of the second turbine driven feedwater pump will decrease air in-leakage and assure offgas treatment system availability." It is not clear why the feedwater pumps affect the in-leakage rate.

The Technical Specifications identify noble gas monitors, iodine samplers and particulate samplers to monitor gaseous effluent releases. Stack and vent effluents are continuously surveyed during release of noble gases. Iodine and particulate samplers are routinely analyzed in accordance with Technical Specification Appendix B, Table 3.2-1

Gaseous Effluent Monitor Setpoints

Section 4.3.2 of the ODCM contains the methodology used to determine the setpoints for the noble gas effluent monitors as required by Technical Specification Appendix B, Section 3.1.d. Technical Specification Appendix B, Section 3.1.a states that radioactive gaseous wastes released to the environment from each of the five effluent release points shall be monitored and recorded. The monitors for releases from the refuel area exhaust and reactor building area exhaust have automatic isolation for the secondary containment prior to initiation of the SBGTS.

The total body dose rate limit is stated as being more restrictive than the skin dose rate and is used for determining the setpoints for the noble gas monitors. The noble gas monitors maximum setpoints are based on the maximum concentration required to yield the total body dose rate limit. The calculated setpoints account for simultaneous releases.

Technical Specification Appendix B, Section 3.2.a states that the dose rate limits apply to areas "at or beyond the site boundary." Additionally, Figure 5.1-1 in the Technical Specifications shows the site

boundary to include the JAFNPP as well as NMP-1 and NMP-2. Therefore, it appears that the contribution from NMP-1 and NMP-2 must be considered when demonstrating compliance to the dose rate limit of Technical Specification Appendix B Section 3.2.a and not the dose rate from JAFNPP only.

In Section 4.3.2.b.(2) the value for the elevated release rate should be 2.1E+6 μCi/sec instead of 2.4E+6 μCi/sec.

With the exception of the uncertainty of including contributions for NMP-1 and NMP-2 when demonstrating compliance to the dose rate limit of Technical Specification Appendix B Section 3.2.a, the methods for determining the setpoints for the noble gas effluent monitors are, in general, in agreement with the guidelines of NUREG-0133 and are considered acceptable.

Concentrations in Liquid Effluents

Section 3.2.2 contains the methodology for determining the radioactivity concentrations in liquid effluents as required by Surveillance Technical Specification Appendix B, Section 2.2.b. In Section 3.2.2.a the word "percentage" should be removed from the definition for f_2. Regardless, the methodology is within the guidelines of NUREG-0133 and is considered acceptable.

Dose Rates in Gaseous Effluents

Section 4.3.1 contains the methodology for determining dose rates from noble gases to the total body and skin as required by Surveillance Technical Specification Appendix B Section 3.2.a.

In Section 4.3.1.b.(2), the definition for DR_{TB} for Equation 4-5 indicates that this quantity is the dose rate from nuclide i, whereas it in fact is the dose rate to the total body from all noble gas radionuclides.

The definition of DR_{SKIN}, for Equation 4-6, indicates that this quantity is the the dose rate from nuclide i, whereas it is in fact the dose rate to the skin from all noble gas radionuclides. Also, the quantity DR_{SKIN_i} is not defined for Equation 4-6.

In Section 4.3.1.c.(7) in the equation for DR_{SKIN_i} the release rate quantity, $\dot{Q}_i$ is missing.

With the exception of the discrepancies indicated above, the methodology for determining the dose rate due to the release of radioactive noble gases is in agreement with the guidelines of NUREG-0133 and is considered acceptable.

Section 4.3.3.b contains the methodology for determining the instantaneous dose rate due to the release of tritium, iodine-131, iodine-133 and all radionuclides in particulate form with half lives greater than 8 days for the inhalation pathway as required by Surveillance Technical Specification Appendix B, Section 3.2.b.

In Section 4.3.3 the text does not specifically indicate tritium as required in Technical Specification Appendix B, Section 3.2.b.

In Section 4.3.3.b, in Equation 4-9 the quantity $\dot{Q}_i$ should be the total release rate for nuclide i from all release points in order to determine a total dose rate. Also, in Equation 4-10, which determines the limiting iodine release rate, there should be a summation due to all radionuclides.

With the exception of the indicated discrepancies, the methodology for determining the dose rates due to the release of tritium, iodine-131, iodine-133 and all radionuclides in particulate form with half lives greater than 8 days is in agreement with the guidelines of NUREG-0133 and is considered acceptable.

Dose Due to Liquid Effluents

Section 3.4.1.b contains the methodology for determining the dose due to radioactive material released in liquid effluents as required by Surveillance Technical Specification Appendix B, Section 2.3.a.

JAFNPP identifies fish and potable water pathways for the dose calculations, designating the adult as the maximum exposed individual and the total body and liver as the critical organs.

In Section 3.4.1.b the quantity Δt_1 in Equation 3-4 should be the number of hours in the calendar year instead of the number of hours during liquid effluent releases for the calendar year. Also, a summation of the dose due to all radionuclides is missing from Equation 3-4.

In Section 3.4.1.b the reference for the derivation of the quantity $A_{i\tau}$ should be Appendix A, Table L-4.1 instead of Appendix A.

In Section 3.4.2.b.(1) the definition of Δt_1 for Equation 3-5 should be the number of hours of the reporting period (e.g. monthly, quarterly, annually) and not the number of hours over which the release occurs. Also, the quantity $(DF)_1$ should be the total volume of dilution released during the reporting period. A summation of the dose due to all radionuclides should be included in Equation 3-5.

The methodology in Section 3.4.2.c should be made consistent with the previous comments in this review concerning the definition of time period Δt_1 identified in Sections 3.4.1.b and 3.4.2.b.(1).

Appendix A, Table L-1 lists the maximum permissible concentrations in water in unrestricted areas. The following discrepancies were detected by the reviewer:

- The first Zn-69 should be listed as Zn-65.

- The value for insoluble Y-93 should be 3E-4 μCi/ml instead of 3E-5 μCi/ml.

Appendix A, Tables L-2 and L-3 contain liquid dose conversion factors, $A_{i\tau}$ for all age groups, pathways and radionuclides. Using the methodology given in Appendix A, Table L-4.1 the reviewer (with the exception of the boating pathway) was unable to reproduce a large fraction of the dose conversion factors. It is recommended that the Licensee ensure the validity of each and every dose conversion factor.

Appendix A, Table L-4.1 gives a derivation of the liquid dose factor $A_{i\tau}$. The following errors were detected by the reviewer:

- In the definition for Z in Table L-4.1, the quantity M_c is not defined.

- The units for the quantity λ_i should be sec^{-1} instead of secs.

- A reference should be provided supporting the quantities Dw, and Df in Equation L-4.1.

- In the calculational example for swimming, the equation referred to should be Equation L-4.1 instead of Equation L-4.4. Also, for the quantity DFi the table referred to should be Table L-4.4 instead of Table L-4-2.

- The purpose for Table L-4.2 is unclear since all the activity values are unity and the table is not referred to in the ODCM.

- The following tables are not referred to in the ODCM: Tables L-4.3, L-5, L-5.1, and L-5.2.

- The ODCM states that the dose conversion factors tabulated in Table L-4.4 for swimming and boating are generated using the

Impell computer code. Either the methodology and parameters contained in the Impell code should be submitted for review or the Licensee should include the reference containing NRC approval for use of this code.

Primarily due to the incorrect definition for Δt_1 and the values in the dose factor tables, the methodology for calculating the dose due to the release of radioactivity in liquid effluents is not in agreement with the guidelines of NUREG-0133 and is not considered acceptable.

Dose Due to Gaseous Effluents

Sections 4.4.1.b.(1), 4.4.2.a and 4.4.2.b contain the methodologies for calculating the cumulative gamma and beta air doses due to the release of radioactive noble gases as required by Surveillance Technical Specification Appendix B, Section 3.3.a.

In Section 4.4.2.a.(2)(a), the reference for the detailed explanation and evaluation of the quantity M_{eff} should be Appendix E instead of Appendix C. Also, the value for (X/Q) on Page 42 of the ODCM, for the elevated release should be 3.8E-8 sec/m^3 instead of 3.8E-7 sec/m^3.

In Section 4.4.2.b.(2) the second half of the first paragraph is evidently out of place since it discusses iodines and particulates in a section describing noble gas beta air dose methodology.

With the exception of these two discrepancies, the method for calculating the air dose due to the release of radioactive noble gases is in agreement with the guidelines of NUREG-0133 and is considered acceptable.

Sections 4.4.1.b.(2) and 4.4.2.c contain the methodologies for calculating the cumulative dose due to the release of I-131, I-133, tritium, and radionuclides in particulate form with half-lives greater than eight days as required by Surveillance Technical Specification

Appendix B, Section 3.4.a.

In Sections 4.4.1.b.(2) and 4.4.2.c, tritium has been omitted from the radionuclides included for determining cumulative dose but is required by Technical Specification Appendix B, Section 3.4.b.

In Section 4.4.1.b.(2) the reference for the derivation of R_{i7}, should be to Appendix B, Table G-7.1B instead of Appendix B, Table G-8.

In Section 4.4.2.c.(2) the subscript D should be removed from the quantity (D/Q) in Equations 4-25 and 4-26.

Appendix B, Tables G-4, G-5, and G-6 contain gaseous dose conversion factors, P_{i_7} and R_{i_7} for all age groups, pathways, and radionuclides. Using the methodology given in Appendix A, Table G-7 the reviewer was unable to reproduce a large fraction of the dose conversion factors. It is recommended that the Licensee check the validity of each and every dose conversion factor.

Appendix B, Tables G-7 gives a derivation of gaseous dose factors, P_i and R_i. The following discrepancies were detected by the reviewer:

- In Table G-7.1A tritium has been omitted from the radionuclides included for determining the inhalation dose factor but is required by the Technical Specifications.

- In Table G-7.1B methodology for determining the gaseous dose factors due to tritium in the cow-milk, cow-meat, goat-milk and fruit-vegetable pathways is missing.

- In Table G-7.1B, Section I the quantity (DFAi) should be written $(DFAi)_a$ to denote the age group.

- In Table G-7.1B, Section III in the example calculation, the quantity $(DFLi)_i$ should be written $(DFLi)_a$.

- In Table G-7.1B the R_i factor values determined in the example calculations for Co-60 (child - total body) for all pathways except inhalation are referenced as being listed in Appendix B, Table G-6 in the adult portion instead of the child portion of the table.

- The purpose of Table G-7.2 is unclear since all the activity values are unity and the table is not referred to in the ODCM.

- The purpose for some of the pages in Table G-7.3 is not clear and also the table is not referred to in the ODCM.

The methodology for calculating the cumulative dose due to the release of I-131, I-133, tritium, and radionuclides in particulate form with half-lives greater than eight days is in agreement with the guidelines of Regulatory Guide 1.109, Revision 1 and NUREG-0133 and is considered acceptable. However, since the reviewer was unable to verify the dose factors it is uncertain if the methodology can be used to demonstrate compliance to the Technical Specifications.

Dose Projections

Section 3.5.2 of the ODCM describes the method used to project doses due to the expected releases of radioactive liquid effluents to determine when the liquid radwaste treatment system should be operated as required in Surveillance Technical Specification Appendix B, Section 2.4.a. The methodology for determining the dose projection due to liquid radwaste effluents is in agreement with NUREG-0133 and is considered acceptable.

Section 4.5.2 of the ODCM describes the method used to project doses due to release of radioactive gases when the offgas treatment system is not in use as required in Surveillance Technical Specification Appendix B, Section 3.6.a. It does not appear that there are ventilation exhaust treatment systems and the gaseous radwaste treatment system is in use whenever the second steam driven feedwater pump is in operation.

Therefore, it is not clear why dose projections are required by the Technical Specifications since the purpose for the dose projections for a BWR is to determine required use of the ventilation exhaust treatment systems. Section 4.5.2.a refers to the limits of Technical Specification Appendix B, Section 3.6.a whereas no limits are found in this Technical Specification. Regardless, the methods for determining dose projections are within the guidelines of NUREG-0133 and are considered acceptable.

Total Dose

Section 5.0 of the ODCM contains the methodology for calculating the total dose contributions including direct radiation as required by Surveillance Technical Specification Appendix B, Section 5.1.a.

In Section 5.1 in the first sentence of the first paragraph the text should read: "Dose evaluation to demonstrate compliance with the 40 CFR 190 dose limits need only be performed if the quarterly or annual doses ... exceed twice the dose limits of Technical Specifications Appendix B, Sections 2.3.a, 3.3.a or 3.4.a respectively", in order to be consistent with the Technical Specifications.

With the exception of the indicated discrepancy, the methodology is in agreement with the requirements of the Technical Specifications and is considered acceptable.

Environmental Monitoring Program

Table H-1 in Appendix H of the ODCM contains specific parameters of distance and the direction sector from the site and additional information for each and every sample identified in Surveillance Technical Specification Section 6.1. The environmental monitoring program is in compliance with Technical Specification Appendix B, Table 6.1-1 and is considered acceptable.

Summary

In summary, the Licensee's ODCM uses documented and approved methods that are, in general, consistent with the methodology and guidance in NUREG-0133 and Regulatory Guide 1.109, Revision 1. However, due primarily to the incorrect definition of Δt_1 and the erroneous values for the liquid and gaseous dose factors it is recommended that the NRC request another revision.

4. CONCLUSIONS

The Licensee's ODCM updated through Revision 5, dated March 1987 for JAFNPP was reviewed. It was determined that the ODCM uses methods that are, in general, consistent with the guidelines of NUREG-0133. The methodology in most sections of the ODCM is acceptable for use in demonstrating compliance to the radiological effluent technical specifications. However, it is recommended that another revision to the ODCM be submitted to address the discrepancies identified in the review.

- In Section 3.2.2.a the word "percentage" should be removed from the definition for f_2.

- In Section 3.3.3.b, the equation $S = 0.5 \times C/F_L$ is given to determine the alarm/trip setpoint for the radwaste liquid effluent monitor. However, when $F_L > 0.5$, the value of the setpoint would be less than the expected count rate for the undiluted effluent at concentration C and the monitor would therefore alarm/trip continuously.

- In Section 3.3.3.1.g a reference is made to Equation 3.3, however the equation identifier is missing for Equation 3.3 in Section 3.3.3.b.

- In Section 3.3.4, a summation over the nuclide index "i" in Equation 3-3.c, is missing.

- In Section 3.4.1.b the reference for the derivation of the quantity A_{iT} should be more specifically Appendix A, Table L-4.1 instead of Appendix A.

- In Section 3.4.1.b the quantity Δt_1 in Equation 3-4 should be the number of hours in the calendar year instead of the number of hours during liquid effluent releases for the calendar year.

Also, a summation of the dose due to all radionuclides is missing from Equation 3-4.

- In Section 3.4.2.b.(1) the definition of Δt_1 for Equation 3-5 should be the number of hours of the reporting period (e.g. monthly, quarterly, annually) and not the number of hours over which the release occurs. Also, the quantity $(DF)_1$ should be the total volume of dilution released during the reporting period. A summation of the dose due to all radionuclides should be included in Equation 3-5.

- The methodology in Section 3.4.2.c should be made to be consistent with the previous comments in this review concerning the definition of the time period Δt_1 identified in Sections 3.4.1.b and 3.4.2.b.(1).

- A simplified flow diagram of the solid radwaste treatment system is not included in the ODCM.

- In Section 4.3.1.b.(2) the definition for DR_{TB} for Equation 4-5 indicates that this quantity is the dose rate from nuclide i, whereas it in fact is the dose rate to the total body from <u>all</u> noble gas radionuclides.

- The definition of DR_{SKIN}, for Equation 4-6, indicates that this quantity is the the dose rate from nuclide i, whereas it is in fact the dose rate to the skin from <u>all</u> noble gas radionuclides.

- In Section 4.3.1.b.(2), the quantity DR_{SKIN_i} is not defined for Equation 4-6.

- In Section 4.3.1.c.(7), in the equation for DR_{SKIN_i}, the parameter $\dot{Q}$ is missing.

- Technical Specification Appendix B, Section 3.2.a states that the dose rate limits apply to areas "at or beyond the site boundary." Additionally, Figure 5.1-1 in the Technical Specifications shows the site boundary to include the JAFNPP as well as NMP-1 and NMP-2. Therefore, it appears that the contribution from NMP-1 and NMP-2 must be considered when demonstrating compliance to the dose rate limit of Technical Specification Appendix B Section 3.2.a and not the dose rate from JAFNPP only.

- In Section 4 3.2.b.(2) the value for the elevated release rate should be 2.1E+6 μCi/sec instead of 2.4E+6 μCi/sec.

- In Section 4.3.3 the text does not specifically indicate tritium as required in Technical Specification Appendix B, Section 3.2.a.b.

- In Section 4.3.3.b, in Equation 4-9 the quantity $\dot{Q}_i$ should be the total release rate for nuclide i from all release points, in order to determine a total dose rate.

- In Section 4.3.3.b in Equation 4-10, which determines the limiting iodine release rate, there should be a summation due to all radionuclides.

- In Sections 4.4.1.b.(2) and 4.4.2.c, tritium has been omitted from the radionuclides included for determining cumulative dose but is required by Technical Specification Appendix B, Section 3.4.b.

- In Section 4.4.1.b.(2) the reference for the derivation of R_{iT}, should be to Appendix B, Table G-7.1B instead of Appendix B, Table G-8.

- In Section 4.4.2.a.(2)(a), the reference for the detailed explanation and evaluation of the quantity M_{eff} should be Appendix E instead of Appendix C.

- In Section 4.4.1.a.(2) the value for (X/Q) on Page 42 of the ODCM, for the elevated release should be 3.8E-8 sec/m^3 instead of 3.8E-7 sec/m^3.

- In Section 4.4.2.b.(2) the second half of the first paragraph seems to be displaced since it discusses iodines and particulates in a section describing noble gas beta air dose methodology.

- In Section 4.4.2.c.(2) the subscript D should be removed from the quantity (D/Q) in Equations 4-25 and 4-26.

- Section 4.5.2.a refers to the limits of Technical Specification Appendix B, Section 3.6.a however, no limits are found in this Technical Specification.

- In Section 5.1 in the first sentence of the first paragraph the text should read: "Dose evaluation to demonstrate compliance with the 40 CFR 190 dose limits need only be performed if the quarterly or annual doses ... exceed twice the dose limits of Technical Specifications Appendix B, Sections 2.3.a, 3.3.a or 3.4.a respectively", in order to be consistent with the Technical Specifications.

- Appendix A, Table L-1 lists the maximum permissible concentrations in water in unrestricted areas. The following discrepancies were detected by the reviewer:

  - The first Zn-69 should be listed as Zn-65.

  - The value for insoluble Y-93 should be 3E-4 μCi/ml instead of 3E-5 μCi/ml.

- Appendix A, Tables L-2 and L-3 contain liquid dose conversion factors, $A_{i\tau}$ for all age groups, pathways and radionuclides. Using the methodology given in Appendix A, Table L-4.1 the reviewer (with the exception of the boating pathway) was unable to reproduce a large fraction of the dose conversion factors. It is recommended that the Licensee ensure the validity of each and every dose conversion factor.

  Appendix A, Table L-4.1 gives a derivation of the liquid dose factor $A_{i\tau}$. The following errors were detected by the reviewer:

  - In the definition for Z in Table L-4.1, the quantity M_c is not defined.

  - The units for the quantity λ_i should be sec^{-1} instead of secs.

  - A reference should be provided for the quantities Dw, and Df in Equation L-4.1.

  - In the calculational example for swimming, the equation referred to should be Equation L-4.1 instead of Equation L-4.4. Also, for the quantity DFi the table referred to should be Table L-4.4 instead of Table L-4-2.

  - The purpose for Table L-4.2 is unclear since all the activity values are unity and the table is not referred to on the ODCM.

  - The following tables are not referred to in the ODCM: Tables L-4.3, L-5, L-5.1, and L-5.2.

  - The ODCM states that the dose conversion factors tabulated in Table L-4.4 for swimming and boating are generated using

the Impell computer code. Either the methodology and parameters contained in the Impell code should be submitted for review or the Licensee should include the reference containing NRC approval for use of this code.

- Appendix B, Tables G-4, G-5, and G-6 contain gaseous dose conversion factors, P_{i_τ} and R_{i_τ} for all age groups, pathways, and radionuclides. Using the methodology given in Appendix A, Table G-7 the reviewer was unable to reproduce a large fraction of the dose conversion factors. It is recommended that the Licensee check the validity of each and every dose conversion factor.

- Appendix B, Tables G-7 gives a derivation of gaseous dose factors, P_i and R_i. The following discrepancies were detected by the reviewer:

  - In Table G-7.1A tritium has been omitted from the radionuclides included for determining the inhalation dose factor but is required by the Technical Specifications.

  - In Table G-7.1B methodology for determining the gaseous dose factors due to tritium in the cow-milk, cow-meat, goat-milk and fruit-vegetable pathways is missing.

  - In Table G-7.1B, Section I the quantity (DFAi) should be written $(DFAi)_a$ to denote the age group.

  - In Table G-7.1B, Section III in the example calculation, the quantity $(DFLi)_i$ should be written $(DFLi)_a$.

  - In Table G-7.1B the R_i factor values determined in the example calculations for Co-60 (child - total body) for all pathways except inhalation are referenced as being listed in Appendix B, Table G-6 in the adult portion instead of the

child portion of the table.

- The purpose of Table G-7.2 is unclear since all the activity values are unity and the table is not referred to in the ODCM.

- The purpose for some of the pages in Table G-7.3 is not clear and also the table is not referred to in the ODCM.

- The Technical Specifications and Appendix F, Table F-1 identify the five gaseous effluent release points listed above. However, Figures F-1, F-2 and F-4 identify only four gaseous release points.

- Figure F-4 also shows a simplified diagram of the gaseous radwaste treatment system. Another diagram is needed however, showing a diagram of the components and pathways for the Standby Gas Treatment System (SBGTS).

- In Figure F-5 of the ODCM the diagram showing the liquid effluent release paths does not show the point where the liquid radwaste effluent line joins the circulating water and the service water lines for dilution prior to discharge.

5. REFERENCES

1. Title 10, Code of Federal Regulations, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion, 'As Low As Is Reasonably Achievable,' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

2. Letter from J. P. Bayne (New York Power Authority) to D. B. Vassallo (NRC), Subject: James A. FitzPatrick Nuclear Power Plant, Off-Site Dose Calculation Manual, October 26, 1983.

3. Letter from H. I. Abelson (NRC) to J. P. Bayne (New York Power Authority), Subject: Approval of James A. FitzPatrick Nuclear Power Plant ODCM, May 29, 1985.

4. Letter from R. J. Converse (New York Power Authority) to T. E. Murley (NRC), Subject: J. A. FitzPatrick Nuclear Power Plant, Semiannual Radioactive Effluent Release Report, February 28, 1986.

5. Letter from H. I. Abelson (NRC) to J. C. Brons (New York Power Authority), Subject: Revision to the Offsite Dose Calculation Manual (ODCM), October 2, 1986.

6. Letter from R. J. Converse (New York Power Authority) to T. E. Murley (NRC), Subject: J. A. FitzPatrick Nuclear Power Plant, Semiannual Radioactive Effluent Release Report, August 29, 1986.

7. Letter from R. J. Converse (New York Power Authority) to T. E. Murley (NRC), Subject: J. A. FitzPatrick Nuclear Power Plant, Semiannual Radioactive Effluent Release Report, February 27, 1987.

8. Letter from R. J. Converse (New York Power Authority) to T. E. Murley (NRC), Subject: J. A. FitzPatrick Nuclear Power Plant, Docket No. 50-333, License No. DPR-59, Revision to the Off-Site Dose Calculation Manual (ODCM), April 2, 1987.

9. "Standard Radiological Effluent Technical Specifications for Boiling Water Reactors," Rev. 3, Draft 7", intended for contractor guidance in reviewing RETS proposals for operating reactors, NUREG-0473, September 1982.

10. "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, A Guidance Manual for Users of Standard Technical Specifications," NUREG-0133, October 1978.

11. "General Contents of the Offsite Dose Calculation Manual," Revision 1, Branch Technical Position, Radiological Assessment Branch, NRC, February 8, 1979.

12. "Calculation of Annual Doses of Evaluating Compliance with 10 CFR 50, Appendix I," Regulatory Guide 1.109, Rev. 1, October 1977.

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13 ABSTRACT (200 words or less)

The Offsite Dose Calculation Manual for the James A. FitzPatrick Nuclear Power Plant contains a description of the methodology and parameters to be used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitor alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program. The NRC transmitted the ODCM updated through Revision 5 to the Idaho National Engineering Laboratory for review. The ODCM, complete through Revision 5, was reviewed in its entirety by EG&G Idaho at the INEL and the results of the review are presented in this report. It was determined that the ODCM updated through Revision 5 uses methods that are, in general, in agreement with the guidelines of NUREG-0133. However, it is recommended that another revision to the ODCM be submitted to address the discrepancies identified in the review.

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