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0CAN059503

U. S. Nuclear Regulatory Commission
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Subject: Arkansas Nuclear One - Units 1 and 2
Docket Nos. 50-313 and 50-368
License Nos. DPR-51 and NPF-6
Technical Specification Change Request Concerning Open
Containment Personnel Airlock During Core Alterations

Gentlemen:

Attached for your review and approval are technical specification changes revising the Arkansas Nuclear One (ANO) Unit 1 and Unit 2 reactor containment building closure requirements during fuel movement and core alterations. These changes are intended to permit the reactor containment building personnel airlock doors to remain open during fuel handling.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

Entergy Operations requests that the effective date for this change be within 30 days of issuance. Although this request is neither exigent nor emergency, your prompt review is requested prior to the next ANO-2 refueling outage (2R11) which is currently scheduled to begin September 22, 1995.

A001

Very truly yours,

JW Yelverton

JWY/nbm
Attachments

To the best of my knowledge and belief, the statements contained in this submittal are true.

SUBSCRIBED AND SWORN TO before me, a Notary Public in and for Johnson **County and the State of Arkansas, this** 19 **day of** May**, 1995.**

Juana M. Tapp
Notary Public
My Commission Expires 11-8-2000

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ATTACHMENT

TO

0CAN059503

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NOs. DPR-51 and NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNITS ONE & TWO

DOCKET NOs. 50-313 and 50-368

DESCRIPTION OF PROPOSED CHANGES

The proposed changes to the ANO-1 and ANO-2 Technical Specifications include the following:

- Specification 3.8.6 for ANO-1 including associated bases has been revised to allow the reactor building personnel and emergency hatches to remain open during fuel handling as long as at least 23 feet of water is maintained above the fuel seated within the reactor pressure vessel.

- Specification 3.9.4 for ANO-2 has also been revised to allow the containment building personnel airlock doors to remain open during fuel handling as long as at least 23 feet of water is maintained covering the fuel (currently required by specification 3.9.9) seated within the reactor pressure vessel.

- Specifications 3.8.11 for ANO-1 including associated bases and 3/4.9.3.a for ANO-2 have been revised to increase the minimum decay time from shutdown to the movement of irradiated fuel in containment from 72 to 100 hours.

- The bases for specifications 3/4.9.9 and 3/4.9.10 for ANO-2 have been revised to reflect an increase in the assumed amount of iodine gap gas activity from 10% to 12%.

BACKGROUND

ANO-1 and ANO-2 Technical Specifications 3.8.6 and 3.9.4, respectively, require that a minimum of one containment personnel airlock door, as well as other containment penetrations, be closed during core alterations and handling or movement of irradiated fuel within the reactor containment building. The purpose of this requirement is to mitigate the consequences of a fuel handling accident. Current fuel handling accident analysis assumptions are reflected in Sections 14.2.2.3 and 15.1.23 of the ANO-1 and ANO-2 Safety Analysis Reports (SAR), respectively. Entergy Operations has performed a new fuel handling accident analysis which assumes that the containment personnel airlock doors are open at the time of the accident.

DISCUSSION OF CHANGE

This proposed change to technical specifications would allow the containment personnel airlock doors to be open during fuel movement and core alterations. The purpose of the current requirements to have one containment personnel airlock door closed during core alterations and fuel movement is to prevent the escape of radioactive material in the event of a fuel handling accident. In order to assure that the doses to the public remain within acceptable limits, Entergy Operations has performed a revised fuel handling accident analysis. The re-analysis does not assume that the containment personnel airlock doors are closed at the time

of the accident; rather, they are assumed to be open. Additionally, doses to the control room are not being considered in this submittal because they remain bounded by the loss-of-coolant-accident (LOCA) analysis for both of the ANO units.

During a refueling outage, other work in containment does not stop during fuel movement and core alterations. This requires that personnel operate the containment personnel airlock doors numerous times in order to enter and exit containment. This excessive cycling and heavy use of the containment personnel airlocks necessitates substantial maintenance on the containment personnel airlock door components.

There are a large number of people in containment during a refueling outage including during fuel movement and core alterations. Should a fuel handling accident occur, it would take a number of cycles of the containment personnel airlock to evacuate personnel from within containment. With each personnel airlock cycle, more containment air would be released, and while waiting their turn to exit, the workers would be exposed to the released activity. Alternatively, 10CFR50.54(x) could be invoked to allow both doors of the containment personnel airlock opened while the personnel inside containment are evacuated, and then close the doors. In either case, there is a release of activity into the atmosphere. Under the proposed change, the containment could be evacuated without invoking 10CFR50.54(x) and then sealed. This would reduce dose to the workers in the event of an accident while maintaining acceptable doses to the public.

The Staff has approved a similar change for Calvert Cliffs on August 31, 1994, with issuance of Amendment Nos. 194 and 171 for Unit 1 and Unit 2, respectively. As stated in the Calvert Cliffs' safety evaluation, the Staff did not review the licensee's fuel handling accident analysis, but instead performed an independent analysis. The Staff's analysis of the Calvert Cliffs fuel handling accident, used the accident source term given in Regulatory Guide (RG) 1.4, assumptions contained in RG 1.25, and the review procedures specified in Standard Review Plan (SRP) Section 15.7.5. The Staff also assumed damage of an entire fuel assembly, an instantaneous puff release of noble gases and radioiodine from the gap of the broken fuel rods as gas bubbles up through the 23 feet of water covering the fuel, all airborne radioactivity reaching the containment atmosphere was exhausted within two hours into the environment, and all radioactive material in the fuel rod gap was assumed to have decayed for at least 100 hours.

Utilizing the Staff's approach yields an acceptable offsite thyroid dose for Calvert Cliffs and unacceptable offsite thyroid doses for the ANO units. If an entire fuel assembly were assumed to be damaged, the offsite doses would be 161 Rem and 164 Rem to the thyroid and 2.29 Rem and 2.35 Rem to the whole body for ANO-1 and ANO-2, respectively. The ANO offsite thyroid doses exceed 25% of the 10CFR Part 100 limit (75 Rem); however, they are a fraction of the 10CFR Part 100 limit. The primary reason for this is that ANO's value of X/Q is approximately six times larger than Calvert Cliffs' value due to site-specific meteorological differences.

There are numerous conservatisms included in the assumptions for a fuel handling accident. Relaxing some of the very conservative assumptions yields acceptable results for the ANO units. The assumptions postulated in the revised calculation of the radiological consequences of a fuel handling accident in the reactor containment building and the spent fuel pool area are summarized below. The inherent conservatisms are discussed in the referenced notes.

- In order to remain consistent with the ANO current licensing basis, four rows of fuel rods from a single dropped assembly are assumed to be damaged. This is 56 fuel rods for ANO-1 and 60 for ANO-2. (Justification is provided in Note 1)

- Damaged fuel rods are assumed to release their gas gap activities. The gas gap activities consist of 12% iodines and 10% of noble gases except for Kr-85 which is 30%. These release fractions are consistent with RG 1.25 except for the iodines which is conservative due to consideration of extended-burnup fuel. (Conservatisms are discussed in Note 2)

- The single damaged fuel assembly is assumed to have operated with a radial peaking factor of 1.65. This 1.65 peaking factor comes from RG 1.25 and is a conservative estimation of the actual power level the highest power assembly might experience. In reality, over the life of a fuel assembly, the peaking factor changes and high burnup fuel assemblies have a peaking factor that will be significantly lower. (Conservatisms are discussed in Note 3)

- The iodine gap inventory is assumed to be composed of 99.75% inorganic species and 0.25% organic species of iodine which comes from RG 1.25. The refueling canal and spent fuel water total effective decontamination factor of iodine is assumed to be 100. This assumption is conservative and consistent with RG 1.25. (Conservatisms are discussed in Note 4)

- The radioactive material that escapes from the refueling canal to the reactor containment building is released from the building to the atmosphere over a two hour period. (Conservatisms are discussed in Note 5)

- Atmospheric dispersion factors assume a ground level release. The ANO site boundary X/Q is 6.5 E-4 seconds/cubic meter. (Conservatisms are discussed in Note 6)

- Core isotopic inventories and half lives are consistent with ORIGEN2 which is a computer code developed by Oak Ridge National Labs (TM-7175) entitled "Isotope Generation and Depletion Code." ORIGEN2 has been used to supply or verify the material composition and characteristics that formed the basis for licensing fuel cycle facilities and regulatory efforts in waste management rulemaking by the NRC in NUREG/CR-0458 and NUREG/CR-0456, the Department of Energy, and the Environmental Protection Agency.

- Core power is assumed to be approximately 103% of 2772 and 2815 MWt for ANO-1 and ANO-2, respectively. Operation at this power level continuously prior to shutdown is

assumed. The thermal power rating of 2772 MWt is in excess of the current licensed limit of 2568 MWt. This is in anticipation of future plans to uprate ANO-1.

- The accident occurs at least 100 hours after plant shutdown. Radioactive decay of the fission product inventory during this interval is taken into consideration. This is the minimum time allowed by the proposed technical specifications before fuel movement may begin.

- The refueling canal and spent fuel pool water decontamination factor of noble gases is assumed to be 1.0 (no decontamination) which is consistent with RG 1.25.

- A breathing rate of 3.47E-04 cubic meters/second was assumed which is consistent with RG 1.25.

- The minimum water depth is 23 feet above the fuel which is consistent with RG 1.25.

- No credit is taken for the following: atmospheric cleanup systems in containment (this is conservative as the containment filtration system could be used to remove iodine), deposition of the plume on the ground in transit to the site boundary, decay as the source escapes and travels to the site boundary, and plateout. (See Note 5)

Based on the above assumptions, the maximum offsite doses associated with the fuel handling accident re-analysis at 100 hours following shutdown are 43.4 Rem and 41.8 Rem to the thyroid and 0.616 Rem and 0.598 Rem to the whole body for ANO-1 and ANO-2, respectively. The offsite doses of the revised fuel handling accident analyses for both ANO units remain well within 25% of 10CFR Part 100 limits which are 75 Rem to the thyroid and 6 Rem to the whole body. In the event of a fuel handling accident, actual offsite doses will be less because at least one of the personnel airlock doors will be closed following an evacuation of containment. The analyses show that it is not necessary to have containment closure in order to show acceptable site boundary doses following a fuel handling accident.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change would allow the containment personnel airlock doors to remain open during fuel movement and core alterations. These doors are normally closed during this time period in order to prevent the escape of radioactive material in the event of a fuel handling accident. These doors are not initiators of any accident. The probability of a fuel handling accident is unaffected by the position of the containment personnel airlock doors.

The proposed change alters assumptions made in evaluating the radiological consequences of a fuel handling accident inside the reactor containment building. Allowing the containment personnel airlock doors to remain open during fuel movement and core alterations does increase, however not significantly, the consequences of a fuel handling accident inside containment. Previously, the fuel handling accident inside containment was bounded by the fuel handling accident analysis in the spent fuel pool area of the auxiliary building. Part of the dose increase has been offset by the increase in the minimum decay time before irradiated fuel may be moved inside the reactor containment building. Extending the minimum decay time actually decreases the consequences of a fuel handling accident by reducing the radioactive inventory of the irradiated fuel which could possibly be released during a fuel handling accident. The revised fuel handling accident analysis results in maximum offsite doses of 43.4 Rem and 41.8 Rem to the thyroid and 0.616 Rem and 0.598 Rem to the whole body for ANO-1 and ANO-2, respectively. The calculated offsite doses are well within the limits of 10CFR Part 100. Also, the calculated doses are larger than the actual doses which would be expected during a fuel handling accident because the calculation does not incorporate the closing of at least one of the personnel airlock doors following evacuation of containment. The proposed change would significantly reduce the dose to workers in the containment in the event of a fuel handling accident by expediting the containment evacuation process.

Therefore, this change does <u>not</u> involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed change does not involve the addition or modification of any plant equipment. Also, the proposed change would not alter the design, configuration, or method of operation of the plant.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

This proposed change has the potential for an increased dose at the site boundary due to a fuel handling accident; however, the dose remains within acceptable limits. The margin of safety as defined by 10CFR Part 100 has not been significantly reduced. There is an increase in the calculated offsite dose resulting from a fuel handling accident; however, the increase is not significant and is well within the limits specified in 10 CFR Part 100. The overall significance will be offset by the increased minimum decay time, the decreased potential radiation dose to workers, and the increased availability of the personnel airlock door in the event of a fuel handling accident. Closing at least one of the personnel airlock doors following an evacuation of containment, further reduces the offsite doses in the event of a fuel handling accident which partially compensates for the higher offsite doses calculated as a result of this proposed change.

Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

NOTES

Note 1: Number of Rods Assumed to be Damaged in a Fuel Handling Accident

The number of fuel rods assumed to be damaged in a fuel handling accident is not considered in RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident for Boiling and Pressurized Water Reactors (PWR)." Regulatory Position 4 of RG 1.13 for Spent Fuel Storage Facility Design Basis, dated December 1975, states that the design of the ventilation and filtration system should be based on the assumption that the cladding of all the fuel rods in one fuel bundle might be breached.

Standard Review Plan (SRP), NUREG-75/087 Section 15.4.1, dated November 24, 1975, states that all the pins in one fuel bundle are assumed damaged unless the applicant's analysis shows additional damage, in which case the applicant's assumptions should be adopted. This has been reworded in the present SRP, NUREG-0800 Section 15.7.4, Revision 1, dated July 1981, which states that the applicant should provide in the SAR conservative analyses of the number of rods assumed damaged both for the spent fuel storage area and inside containment. Reference 6 of NUREG-0800 states that the Staff had used the value of all rods in one assembly for a PWR as representative of the radioactivity release that could be associated with an event of this type which is specified in SRP Section 15.7.4 of NUREG-75/087. The PWR assumption was based on an earlier staff analysis which showed that there was sufficient kinetic energy in the dropping of a fuel assembly that rupture of all rods could not be discounted.

In Reference 6 of NUREG-0800, the Staff re-examines the number of rods which might be damaged in a dropping of a spent fuel assembly while lifting it out of the core. The conservatisms employed in the analysis are:

- The maximum kinetic energy calculated for a dropped fuel assembly was used for the analysis.

- All of the kinetic energy was assumed to break fuel rods. No allowance was made for deformation and energy absorption to the top and bottom structure of the fuel assembly, the spent fuel rack structure, or deformation or energy absorption in the fuel pellets.

- A spent fuel assembly was always assumed to drop onto one or more other fuel assemblies when possible (in the core or the spent fuel pool). In the canal or transfer tube, since the assembly is removed from others, damage to only one assembly can result.

- No allowance was taken for any buoyancy effects in the water. This effect would reduce the effective weight and hence the kinetic energy available for a given impact.

- No allowance was taken for the hydrodynamic drag and energy dissipation of the fuel assembly falling through water.

The Staff concluded that these significant conservatisms lead to a substantial over-estimate of the amount of damaged fuel that could result from a dropped fuel assembly.

Four rows of fuel rods were assumed to be damaged in the fuel handling accident at ANO. The ANO-1 SAR assumed 56 out of 208 rods in an assembly damaged in the fuel handling accident, and the ANO-2 SAR assumed 60 out of 236 rods in an assembly damaged in the fuel handling accident. An extensive justification for assuming only four rows of damaged fuel rods is specified in the ANO-2 SAR. Section 15.1.23.2.1 of the ANO-2 SAR concludes that for the worst case fuel assembly drop accident, no more than four rows of fuel rods would fail due to the combined bending and localized deformation which results from absorbing the kinetic energy at impact. This results in a maximum of 60 fuel rods failing.

The Combustion Engineering System SAR (CESSAR) further addresses the basis of only assuming four rows of damaged fuel rods. The CESSAR shows that when buoyant and drag forces are considered, impact stresses which result from absorbing the kinetic energy of a dropped fuel assembly which has reached its terminal velocity (acceleration equals zero) are below the yield stress of the clad and thus no fuel rod failures will occur. Because this analysis assumed the drop distance was sufficient for the fuel assembly to reach its terminal velocity, the results are applicable for any drop height.

The CESSAR also states that horizontal impact of a fuel assembly could result from a dropped fuel assembly falling from the horizontal position, or from a vertical fuel assembly rotating to the horizontal position. As in the vertical drop analysis, worst case assumptions are made for the horizontal impact velocity (based on terminal velocity) and the rotational impact velocity. The worst case bundle impact results from the horizontal drop since the kinetic energy at impact is greater for the horizontal drop than for the rotational impact. During this horizontal drop, it is postulated that the assembly strikes a protruding structure. For this analysis, a localized loading of one grid span has been assumed. An analysis of the fuel assembly drop has revealed that the most severe impact location is between the two top spacer grids since that impact area is within the fuel rod upper plenum region and the fuel pellets do not provide support for the cladding. To obtain an estimate of the number of fuel rods which might fail, the fuel assembly's grid span was modeled and calculations performed to relate the assembly's kinetic energy at impact to the resulting strain energy in the fuel rods and guide tubes. As a result of the fuel assembly drop, no more than four rows of fuel rods (60 rods) would fail due to the strain resulting from the fuel rods and guide tubes absorbing the bundles kinetic energy at impact. The failure of all 236 fuel rods in one spent fuel assembly is not credible. Due to ANO-1's similarity to ANO-2, this logic would also be applicable to ANO-1 which would justify the assumption of 56 fuel rods being damaged by a fuel handling accident.

The safety evaluation report (SER) for Amendment No. 44 for ANO-1, dated November 7, 1979, considered the consequences of a fuel handling accident inside containment. The Staff stated that the probability of the postulated fuel handling accident inside containment is small. The Staff also noted that not only have there been several hundred reactor-years of plant operating experience with only a few accidents involving spent fuel being dropped into the

core, but none of these accidents have resulted in measurable releases of activity. A review of commercial reactor incidents involving fuel handling accidents for the period from 1980 to the present confirms that this trend has continued. Only a few accidents involving spent fuel dropped into the core have occurred, but none of these accidents have resulted in measurable releases of activity.

Recently the Staff considered one assembly for the source term of a fuel handling accident in containment for Calvert Cliffs. Entergy Operations considers that the assumption of all the rods in one assembly being damaged in a fuel handling accident in containment is extremely conservative, and therefore, is maintaining the original design basis accident analysis assumption of four rows of fuel rods being damaged in the re-evaluation of the fuel handling accident for the ANO units.

Note 2: GAP Activity

Regulatory Position 1.d of RG 1.25 states that all of the gap activity in the damaged rods is released and consists of 10% of the total noble gases other than Kr-85, 30% of the Kr-85, and 10% of the total radioactive iodine in the rods at the time of the accident. These assumptions are valid for an average burnup for the peak assembly.

In the SER for Amendment No. 111 for ANO-2, dated November 27, 1990, to allow extended burnup, the Staff cites NUREG/CR-5009, dated February 1988. The Staff agrees that the only design basis accident that could be affected by the use of extended burnup fuel, even in a minor way, would be the potential thyroid doses that could result from a fuel handling accident. NUREG/CR-5009 estimated that the I-131 fuel gap activity in the peak fuel rod with extended burnup could be as high as 12%. This value is approximately 20% higher than the value normally used by the Staff in evaluating fuel handling accidents in accordance with RG 1.25. However, for the fuel handling accident, NUREG/CR-5009 concluded that the use of RG 1.25 procedures for the calculation of accident doses for extended burnup fuel may be utilized.

In the re-evaluation of the ANO fuel handling accidents, RG 1.25 assumptions for gap activity were used, with the exception that iodine was assumed to be 12% of the total inventory as opposed to 10%. This is conservative with respect to the 10% requirement of RG 1.25.

Note 3: Radial Peaking Factor

RG 1.25 states that the values assumed for individual fission product inventories are calculated assuming full power operation at the end of core life immediately preceding shutdown and such calculation should include an appropriate radial peaking factor. The minimum acceptable radial peaking factor is 1.65 for PWRs. These radial peaking factors are a direct multiplier in the calculation of source terms, so that the dose is directly proportional to the radial peaking factor assumed for the hottest assembly. The ANO-2 SAR Section 15.1.23.2.2 and the CESSAR propose a realistic assumption of 1.55 versus 1.65. A review of data from the ANO-1 and ANO-2 reload analysis reports shows that over the last three cycles,

the highest assembly radial peaking factor is 1.41. Furthermore, high burnup assemblies, for which a higher gap radioiodine inventory has been assumed, typically have a radial peaking factor that is much lower.

Note 4: Fuel Pool Decontamination Factors

RG 1.25 states that the pool decontamination factors for the inorganic and organic species are 133 and one, respectively, given an overall effective decontamination factor of 100 (i.e., 99% of the total iodine released from the damaged rods is retained by the pool water). This difference in decontamination factors for inorganic and organic iodine species results in the iodine above the fuel pool being composed of 75% inorganic and 25% organic species.

The ANO-2 SAR Section 15.1.23.2.2 states that since the fuel handling process is under water, essentially all of the iodine released from the damaged rods would be retained in the spent fuel pool water because of the preferential distribution of iodine in water over air. The ANO-2 SAR also discusses experiments in which air-stream mixtures containing iodine were bubbled through shallow depths of water. These experiments indicated that for pools very diluted in iodine (such as spent fuel water), only about 1/10,000 of the iodine reaches the water surface in the bubbles. Additional evidence from two deliberate fuel melting excursion experiments showed that only 5.0 E-5 of the iodine released from the melted fuel elements escaped to the air above the pool after passing through about eight to eleven feet of water above the core. ANO-1 SAR Section 14.2.2.3.4 states that in experiments in which air-stream mixtures were bubbled through a water pond, decontamination factors of about 1,000 for iodine were demonstrated. Based on these experiments and other similar results, 99% of the iodine released from the fuel assembly is assumed to remain in the water. Therefore, the iodine decontamination factor within the pool specified by the RG is conservative by a factor of 10-200.

The Staff recognized the conservatism of the pool decontamination factors in the SER for Amendment No. 44 for ANO-1, dated November 7, 1979. The Staff conservatively calculated the potential radiological consequences of a fuel assembly drop onto the reactor core in which it was assumed for this postulated accident that the source term is that given in RG 1.25. The Staff noted that one of the reasons that this was conservative was because the pool decontamination factor for inorganic iodine should be greater than that recommended in RG 1.25.

Note 5: Release Model

Regulatory Position 1.i of RG 1.25 states that the radioactive material that escapes from the pool to the building is released from the building over a two hour time period. RG 1.25 states that the effectiveness of features to reduce the amount of radioactive material available for release to the environment will be evaluated on an individual case basis. RG 1.25 further states in Regulatory Position 3.a(2) that no correction is made for depletion of the effluent plume of radioiodine due to deposition on the ground, or for radioactive decay of radioiodine in transit.

The ANO fuel handling accident re-evaluation has followed these guidelines, and assumed an instantaneous or puff release of activity following the fuel handling accident and employing the 0-2 hour X/Q for the exclusion area boundary of 6.5E-4 seconds/cubic meter. This is conservative as no pressurization of containment will exist to drive the radioactive material out of the containment building. In addition, no credit is taken for the reactor building purge filtration system filters. The ANO-1 and ANO-2 filtration units include a roughing filter, a HEPA filter, and a charcoal filter and the exhaust air is monitored for radioactivity by a radiation monitoring system.

The reactor building purge system is a high capacity system that is sized to change out the containment atmosphere in less than an hour. The personnel airlock is a small opening relative to the containment size. Therefore, even if the personnel airlock is open during a fuel handling accident in the reactor building, the air could be filtered prior to release.

A further conservatism results by neglecting plateout inside the reactor containment building prior to release. TID-14844 states that 50% of the iodines in containment are assumed to be available for release to the atmosphere. The remaining 50% of the iodines are assumed to adsorb onto internal surfaces of the reactor building or adhere to internal components (plateout). TID-14844 states that rather than the assumed reduction factor of two, it is estimated that removal of airborne iodines by various physical phenomena such as adsorption, adherence, and settling (plateout) could give an effect of 3-10 reduction in the calculated offsite dose. It is recognized that this is true for LOCA conditions; however, a similar logic could be applied to fuel handling accident conditions.

TID-14844 also states that cloud depletion as ground deposition (particulate fallout) is not assumed during cloud travel. Such deposition during cloud travel could reduce the dose at the low population zone distance by factors of 2-5.

Note 6: Atmospheric Dispersion Factor

The atmospheric dispersion factor (X/Q) used to evaluate the fuel handling accident is the five percentile number for the exclusion distance for a two hour diffusion model. The X/Q value for ANO is 6.5E-4 seconds/cubic meter. This value is from ANO-1 SAR Section 2.3.6.2.1 and ANO-2 SAR Section 2.3.4.3. This means that 95% of the time, the atmospheric dispersion conditions would be more favorable. Considering a similar conservatism, TID-14844 states that atmospheric dispersion is assumed to occur under inversion type weather conditions. Also, TID-14844 notes that for weather conditions which exist for 75% or so of the time at most sites, the atmospheric dispersion conditions would be more favorable by factors of 5-1000. The 50 percentile number for the exclusion distance and a two hour diffusion model is 5.0 E-5 seconds/cubic meter as specified in the ANO-2 SAR.

PROPOSED TECHNICAL SPECIFICATION CHANGES

ANO-1