Attachment II

Marked Up Ginna Station Technical Specifications

Included pages (note that not all pages are changed):

3.9-4
3.9-5
B3.9-10
B3.9-11
B3.9-12
B3.9-13
B3.9-14
B3.9-15

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch shall be either:

1. bolted in place with at least one access door closed, or

2. isolated by a closure plate, or equivalent that restricts air flow from containment;

b. One door in the personnel air lock shall be closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Ventilation Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 5, there are no accidents of concern which require containment. In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be bolted in place. Good engineering practice dictates that a minimum of 4 bolts be used to hold the equipment hatch in place and that the bolts be approximately equally spaced. As an alternative, the equipment hatch opening can be isolated by a closure plate, or by an equivalent isolation barrier (e.g., overhead door assembly), that restricts air flow from containment.

(continued)

BASES

BACKGROUND (continued)

The containment equipment and personnel air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of plant shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed in the personnel and equipment hatch (unless the equipment hatch is isolated by a closure plate).

or equivalent barrier

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System includes two subsystems. The Shutdown Purge System includes a 36 inch purge penetration and a 36 inch exhaust penetration. The second subsystem, a Mini-Purge System, includes a 6 inch purge penetration and a 6 inch exhaust penetration. During MODES 1, 2, 3, and 4, the shutdown purge and exhaust penetrations are isolated by a blind flange with two O-rings that provide the necessary boundary. The two air operated valves in each of the two mini-purge penetrations can be opened intermittently, but are closed automatically by the Containment Ventilation Isolation Instrumentation System. Neither of the subsystems is subject to a Specification in MODE 5.

(continued)

BASES

BACKGROUND (continued)

In MODE 6, large air exchangers are used to support refueling operations. The normal 36 inch Shutdown Purge System is used for this purpose, and each air operated valve is closed by the Containment Ventilation Isolation Instrumentation in accordance with LCO 3.3.5, "Containment Ventilation Isolation Instrumentation."

The Mini-Purge System also remains operational in MODE 6, and all four valves are also closed by the Containment Ventilation Isolation Instrumentation.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed using the criteria of Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the guideline values specified in 10 CFR 100. Standard Review Plan (SRP), Section 15.7.4, Rev. 1 (Ref. 2), requires containment closure even though this is not an assumption of the accident analyses. The acceptance limits for offsite radiation exposure is 96 rem (Ref. 3).

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement since these are assumed in the SRP.

(continued)

BASES (continued)

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that at least one valve in each of these penetrations is isolable by the Containment Ventilation Isolation System.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions, no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

or equivalent

If the containment equipment hatch (or its closure plate), air lock doors, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Ventilation Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the plant must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This SR demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked or otherwise prevented from closing (e.g., solenoid unable to vent).

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This SR demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24 month Frequency maintains consistency with other similar instrumentation and valve testing requirements. In LCO 3.3.5, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months an ACTUATION LOGIC TEST and CHANNEL CALIBRATION is performed. These Surveillances will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 15.7.

2. NUREG-800, Section 15.7.4, Rev. 1, July 1981.

3. Letter from D. M. Crutchfield, NRC, to J. Maier, RG&E, Subject: "Fuel Handling Accident Inside Containment," dated October 7, 1981.

Attachment III

Proposed Technical Specifications

Included pages (note that not all pages are changed):

3.9-4
3.9-5
B3.9-10
B3.9-11
B3.9-12
B3.9-13
B3.9-14
B3.9-15

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch shall be either:

1. bolted in place with at least one access door closed, or

2. isolated by a closure plate, or equivalent, that restricts air flow from containment;

b. One door in the personnel air lock shall be closed; and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Ventilation Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	24 months

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 5, there are no accidents of concern which require containment. In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be bolted in place. Good engineering practice dictates that a minimum of 4 bolts be used to hold the equipment hatch in place and that the bolts be approximately equally spaced. As an alternative, the equipment hatch opening can be isolated by a closure plate, or by an equivalent isolation barrier (e.g., overhead door assembly), that restricts air flow from containment.

(continued)

BASES

BACKGROUND (continued)

The containment equipment and personnel air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of plant shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, containment closure is required; therefore, the door interlock mechanism may remain disabled, but one air lock door must always remain closed in the personnel and equipment hatch (unless the equipment hatch is isolated by a closure plate or equivalent barrier).

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System includes two subsystems. The Shutdown Purge System includes a 36 inch purge penetration and a 36 inch exhaust penetration. The second subsystem, a Mini-Purge System, includes a 6 inch purge penetration and a 6 inch exhaust penetration. During MODES 1, 2, 3, and 4, the shutdown purge and exhaust penetrations are isolated by a blind flange with two O-rings that provide the necessary boundary. The two air operated valves in each of the two mini-purge penetrations can be opened intermittently, but are closed automatically by the Containment Ventilation Isolation Instrumentation System. Neither of the subsystems is subject to a Specification in MODE 5.

(continued)

BASES

BACKGROUND (continued)

In MODE 6, large air exchangers are used to support refueling operations. The normal 36 inch Shutdown Purge System is used for this purpose, and each air operated valve is closed by the Containment Ventilation Isolation Instrumentation in accordance with LCO 3.3.5, "Containment Ventilation Isolation Instrumentation."

The Mini-Purge System also remains operational in MODE 6, and all four valves are also closed by the Containment Ventilation Isolation Instrumentation.

The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed using the criteria of Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are within the guideline values specified in 10 CFR 100. Standard Review Plan (SRP), Section 15.7.4, Rev. 1 (Ref. 2), requires containment closure even though this is not an assumption of the accident analyses. The acceptance limits for offsite radiation exposure is 96 rem (Ref. 3).

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement since these are assumed in the SRP.

(continued)

BASES (continued)

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that at least one valve in each of these penetrations is isolable by the Containment Ventilation Isolation System.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions, no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

If the containment equipment hatch (or its closure plate or equivalent), air lock doors, or any containment penetration that provides direct access from the containment atmosphere to the outside atmosphere is not in the required status, including the Containment Ventilation Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the plant must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This SR demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked or otherwise prevented from closing (e.g., solenoid unable to vent).

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.

SR 3.9.3.2

This SR demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 24 month Frequency maintains consistency with other similar instrumentation and valve testing requirements. In LCO 3.3.5, the Containment Ventilation Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 24 months an ACTUATION LOGIC TEST and CHANNEL CALIBRATION is performed. These Surveillances will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 15.7.

2. NUREG-800, Section 15.7.4, Rev. 1, July 1981.

3. Letter from D. M. Crutchfield, NRC, to J. Maier, RG&E, Subject: "Fuel Handling Accident Inside Containment," dated October 7, 1981.

Attachment IV

Fuel Handling Accident Inside Containment Dose Consequence Analysis for Ginna Station

Fuel Handling Accident Inside Containment
Dose Consequence Analysis for Ginna Station

1.0 BACKGROUND

The Ginna Station Technical Specifications require that containment "closure" be in effect during movement of irradiated fuel assemblies within containment and during core alterations. Containment "closure" is defined in LCO 3.9.3 as isolating, or providing automatic isolation capability, for all penetrations which provide direct access from the containment atmosphere to the outside atmosphere. Specific requirements are provided for the personnel and equipment air locks in that at least one door in the associated air lock must be closed. A closure plate which restricts air flow from containment can also be used in place of the equipment hatch and its associated air lock door.

These technical specification requirements are provided even though containment closure is not credited in the fuel handling accident dose consequences as described in the technical specification bases. Standard Review Plan (SRP) 15.7.4 states that if a "fuel handling accident will occur only when the containment is isolated, no radiological consequences need be calculated." The SRP further states that if "containment will be open during fuel handling operations, as with a containment purge exhaust system" it should be verified that the resulting doses are "well within" 10 CFR Part 100 limits (i.e., 75 rem for the thyroid and 6 rem for the whole-body). Since Ginna Station was designed and built prior to the SRP, it does not (and is not required to) meet these limits. As such, the fuel handling accident within containment for Ginna Station shows an offsite dose of 96 rem thyroid which has been previously accepted by the NRC as being "well within" 10 CFR Part 100 limits (UFSAR Section 15.7.3.3.2.2).

In support of the upcoming steam generator replacement outage, RG&E is requesting to revise the existing technical specification requirements related to the equipment hatch to allow use of an installed overhead door assembly to isolate the equipment hatch opening. This overhead door assembly is similar to those used in BWR containments and in the Ginna Station Auxiliary Building (which also has technical specification requirements associated with it during movement of irradiated fuel assemblies within the Auxiliary Building). With the use of the overhead door assembly, containment "closure" is essentially maintained since the bases for LCO 3.9.3 specifically state that "since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required" for the isolation devices used to meet this LCO. However, it is recognized that the overhead door assembly could create a larger release path from containment than from other penetrations due to its size and the small gaps which exist between the door and equipment hatch opening. Therefore, the purpose of this analysis is to calculate whole body and thyroid doses in the control room and for the exclusion area boundary (EAB) with the equipment hatch opening isolated using the installed overhead door assembly in place of the current options provided in technical specifications.

In performing this analysis, several issues must be addressed. These are briefly discussed below:

1. The Ginna Station UFSAR (Section 2.3.4.2) provides an atmospheric dispersion factor (i.e., X/Q) value of 4.8E-4 sec/m^3 for the EAB during the first 2 hours of an accident. The X/Q value for the control room is provided in UFSAR Table 6.4-1 as 1.9E-4 sec/m^3. This value is less than that for the EAB even though the control room is closer to the release source. Therefore, this analysis must determine an appropriate value to use in order to calculate control room doses.

2. A leakage rate from containment via the overhead door assembly must be determined during the two hour release following the fuel handling accident. This includes consideration of leakage from other available pathways (e.g., purge lines which may be opened).

3. The analysis must be performed using the higher enrichment fuel with higher peaking factors that is being used for the upcoming fuel cycle.

These issues are evaluated in detail below.

2.0 DETERMINATION OF X/Q VALUES

The basic equation used for the spatial distribution of materials released under the influence of the building wake is:

$$X/Q = K/AU$$

where:

K is the nondimensional concentration coefficient which is a function of the configuration of nearby structures, the release and intake locations, and the wind location;

A is the reference building area; and

U is the wind velocity at the top of the reference building.

The plant configuration is shown in Figure 1. The critical features as shown on this figure is that the control room intake source is on top of the control room building.

The K values used in the above equation for the purpose of this analysis are from Dr. J. Halitsky based on field and wind tunnel experiments (see Chapter 5 of Meteorology and Atomic Energy, 1968, TID-24190). Table 1 provides the K values for each wind direction for the release configuration being evaluated at Ginna.

The Table 1 K values and the control building area of 2,964 m^2 were input in the above equation and processed through 3 years of hourly averaged weather data from the Ginna site weather tower (1992 - 1994). The resulting X/Q values at the control room intake on top of the control building structure were then plotted in Figure 2. As shown by this figure, the 5% probable value of X/Q is 3.0E-4 sec/m^3.

The above calculated value is larger than the control room X/Q value provided in UFSAR Table 6.4-1, but smaller than the X/Q value provided in the Ginna Station UFSAR for the EAB. The value calculated for the EAB is based on generic (i.e., conservative) assumptions which if were similarly evaluated using plant specific considerations would be lowered (see UFSAR Section 2.3.4.2.2). Therefore, the UFSAR X/Q value for the EAB will be conservatively used in this analysis for both the EAB and the control room.

3.0 DETERMINATION OF CONTAINMENT LEAKAGE RATES

The existing UFSAR analysis for fuel handling accidents assumes that containment is not isolated at the time of the event. However, for the purpose of this analysis, containment will be in the configuration required by technical specifications. Therefore, a leakage rate from containment must be determined as discussed below:

3.1 Mini-Purge and Shutdown Purge Penetrations

These penetrations may be opened during fuel movement provided that they are capable of being closed by an OPERABLE Containment Ventilation Isolation Signal. The isolation valves for these penetrations are designed to close within 2 seconds from the time a signal is generated or a total of 5 seconds following the event (see UFSAR Table 6.2-15). This time limit is consistent with Branch Technical Position CSB 6-4 to ensure that the impact on dose analyses is minimized. Therefore, leakage through these pathways is considered negligible and ignored for the purpose of this analysis.

3.2 Equipment Hatch Opening

The installed overhead door assembly has small gaps between the door and the equipment hatch opening. These gaps create the potential for a release path from containment. As such, the leak rate through these gaps must be determined.

The first step is to estimate the size of the gaps. An estimate of a 1/4 inch gap for the entire 88 feet circumference of the opening is used. This results in a leakage area of 1.83 ft^2. It should be noted that this 1/4" gap is indicative of the type of containment barrier which the overhead door assembly is intended to provide and not an acceptance limit for its operability due to the other conservatisms used in this analysis. The next step is to determine the leakage through this area. As discussed in the bases for LCO 3.9.3, containment pressurization is not assumed for a fuel handling accident. However, some pressure differential must be assumed to create a leakage path to the outside environment. For conservatism, two sources of pressure differential were considered as shown below:

a. For the first three minutes following the fuel handling accident, containment is assumed to be pressurized by 0.5 psig above the outside atmosphere. This 3 minutes is the time it takes for the containment free volume to equalize with the outside atmosphere through the 1.83 ft^2 gap. This equates to a leakage rate of 8240 cfm.

b. For the remaining two hour duration, a leakage rate based on the maximum temperature difference between containment (120°F) and the outside atmosphere (-2°F) is assumed. This equates to a leakage rate of 320 cfm.

3.3 Other Containment Penetrations

The remaining containment penetrations which provide direct access from containment to the outside environment must be closed by a manual or automatic isolation valve, blind flange, or equivalent per technical specifications. As discussed in the bases for LCO 3.9.3, these isolation devices do not have to meet 10 CFR 50, Appendix J leakage requirements such that leakage through these penetrations can exist.

At Ginna Station, most penetrations which provide direct access to the outside environment do not have isolation capability other than the installed containment isolation valves. Therefore, LCO 3.9.3 is typically met through the use of these installed containment isolation valves. In addition, due to the conservative assumptions for the equipment hatch, leakage through these paths was ignored.

4.0 DOSE CALCULATIONS

The source of fission products released from the fuel after 100 hours decay and the assumptions for iodine form and removal are based on Regulatory Guide 1.25. This analysis assumes that the fission product inventory from the highest rated assembly was released to the reactor cavity water. Table 2 shows the assumed source term for each isotope.

The whole body dose calculations were made using the following relations and dose factors from Tables 3 and 4.

$$D_{thy} = X/Q \; f_{BR} \, f_{cr} \sum_{i=1}^{n} Q_i \, f_{DFi}$$

where:

f_{BR} is the active breathing rate (assumed to be 3.47E-4 m^3/sec);

f_{cr} is the average relative concentration in the control room based on the in-leakage, filtration, and total volume of the control room (see Table 3);

Q_i is the curie quantity in the containment available for leakage;

f_{DFi} is the adult thyroid inhalation dose factor for the isotope; and

n is the number of iodine isotopes.

The whole body dose conservatively combines both the gamma and beta contributions in the following relationship for a semi-infinite plume:

$$D_{wb} = X/Q \; [0.23 \sum_{i=1}^{n} Q_i \, \bar{E}_{\beta i} + 0.25 \sum_{i=1}^{n} Q_i \, \bar{E}_{\gamma i}]$$

where $\bar{E}_\beta$ and $\bar{E}_\gamma$ are the average beta and gamma energies in units of mev per disintegration. Values for these parameters are given in Table 4.

5.0 RESULTS

The results of the analysis for the various combinations are provided in Table 3. The three cases which were run are described below with the differences between the three cases strictly due to assumptions with respect to the control room ventilation configurations:

5.1 Case 1

The control room was assumed to be in the normal configuration at the time of the fuel handling accident. This results in 2000 cfm of unfiltered air being taken into the control room for the first 30 seconds following the fuel handling accident until the control room is isolated and the system switches to emergency recirculation mode with air being forced through a charcoal system.

5.2 Case 2

The control room was assumed to be isolated prior to the fuel handling accident and subsequently placed in the recirculation mode.

5.3 Case 3

The control room was assumed to be isolated prior to the fuel handling accident with the recirculation mode unavailable.

The results from each of these cases demonstrate that the EAB doses remain far below the SRP requirements of 75 rem at the thyroid and that the control room doses remain far below the limits of General Design Criteria 19 of 30 rem thyroid.

TABLE 1. CONCENTRATION COEFFICIENTS (K) VALUES FOR GINNA CONTROL ROOM GIVEN A RELEASE AT THE EQUIPMENT HATCH	
WIND DIRECTION, θ (WINDS FROM)	K
0	3.0
22.5	4.5
45	0.0
67.5	0.0
90	0.0
112.5	0.0
135	0.0
157.5	0.0
180	0.375
202.5	2.25
225	4.5
247.5	2.25
270	1.5
292.5	0.75
315	0.375
337.5	0.0
Reference Area, A = 2964 m^2 Reference Height for Wind = 36m AGL	

TABLE 2. FUEL GAP INVENTORY RELEASED AT TIME OF GINNA FUEL HANDLING ACCIDENT (100 Hours After Shutdown)		
ISOTOPE	CURIES RELEASED TO REACTOR CAVITY OR SPENT FUEL POOL WATER	CURIES RELEASED TO CONTAINMENT ATMOSPHERE
I-131	5.16E+04	5.16E+02
I-132	Negligible	Negligible
I-133	4.50E+03	4.50E+01
I-134	Negligible	Negligible
I-135	3.22E+00	3.22E-02
Kr-85m	3.12E-03	3.12E-03
Kr-85	1.72E+03	1.72E+03
Kr-87	Negligible	Negligible
Kr-88	Negligible	Negligible
Xe-131m	6.35E+02	6.35E+02
Xe-133m	1.55E+03	1.55E+03
Xe-133	8.33E+04	8.33E+04
Xe-135m	5.16E-01	5.16E-01
Xe-135	1.56E+02	1.56E+02
Xe-138	Negligible	Negligible

TABLE 3. ASSUMPTIONS FOR GINNA FUEL HANDLING ACCIDENT INSIDE CONTAINMENT			
	CASE 1	CASE 2	CASE 3
Containment Volume (ft^3)	1.0E6	1.0E6	1.0E6
Containment Leak Rate During First 3 min (cfm)	8240	8240	8240
Containment Leak Rate During Last 117 min (cfm)	320	320	320
Control Room In-Leakage (cfm)	46	46	46
Control Room Volume (ft^3)	45,784	45,784	45,784
Control Room Recirculation Rate (cfm)	2000	2000	0
Iodine Filter Efficiency - elemental (rem) - organic (rem)	0.9 0.7	0.9 0.7	0.9 0.7
Fraction of Iodine that is - elemental (rem) - organic (rem)	0.75 0.25	0.75 0.25	0.75 0.25
Initial (30 sec) Control Room Flow (cfm)	2000	0	0
Breathing Rate (m^3/sec)	3.47E-4	3.47E-4	3.47E-4
EAB X/Q (sec/m^3)	4.8E-4	4.8E-4	4.8E-4
Control Room Intake X/Q (sec/m^3)	4.8E-4	4.8E-4	4.8E-4
Whole Body Dose ($\beta + \gamma$) - EAB (rem) - Control Room (rem)	0.12 0.05	0.12 0.007	0.12 0.007
Thyroid Dose - EAB (rem) - Control Room (rem)	8.1 0.79	8.1 0.16	8.1 0.53

TABLE 4. ISOTOPIC DATA FOR FUEL HANDLING ACCIDENT			
ISOTOPE	$\overline{E}_\beta$ (MEV/DIS)	$\overline{E}_\gamma$ (MEV/DIS)	THYROID DOSE FACTOR (REM/CI)
I-131	1.91E-1	3.80E-1	1.48E6
I-132	4.90E-1	2.29E0	5.35E4
I-133	4.09E-1	6.07E-1	4.05E5
I-134	6.08E-1	2.63E0	2.50E4
I-135	3.69E-1	1.57E0	1.24E5
Kr-85m	2.70E-1	1.56E-1	0
Kr-85	2.70E-1	2.20E-3	0
Kr-87	1.42E0	7.84E-1	0
Kr-88	4.00E-1	1.93E0	0
Xe-131m	1.50E-1	1.98E-2	0
Xe-133m	2.00E-1	4.16E-2	0
Xe-133	1.40E-1	4.48E-2	0
Xe-135m	1.00E-1	4.28E-1	0
Xe-135	3.40E-1	2.44E-1	0
Xe-138	6.50E-1	1.17E0	0

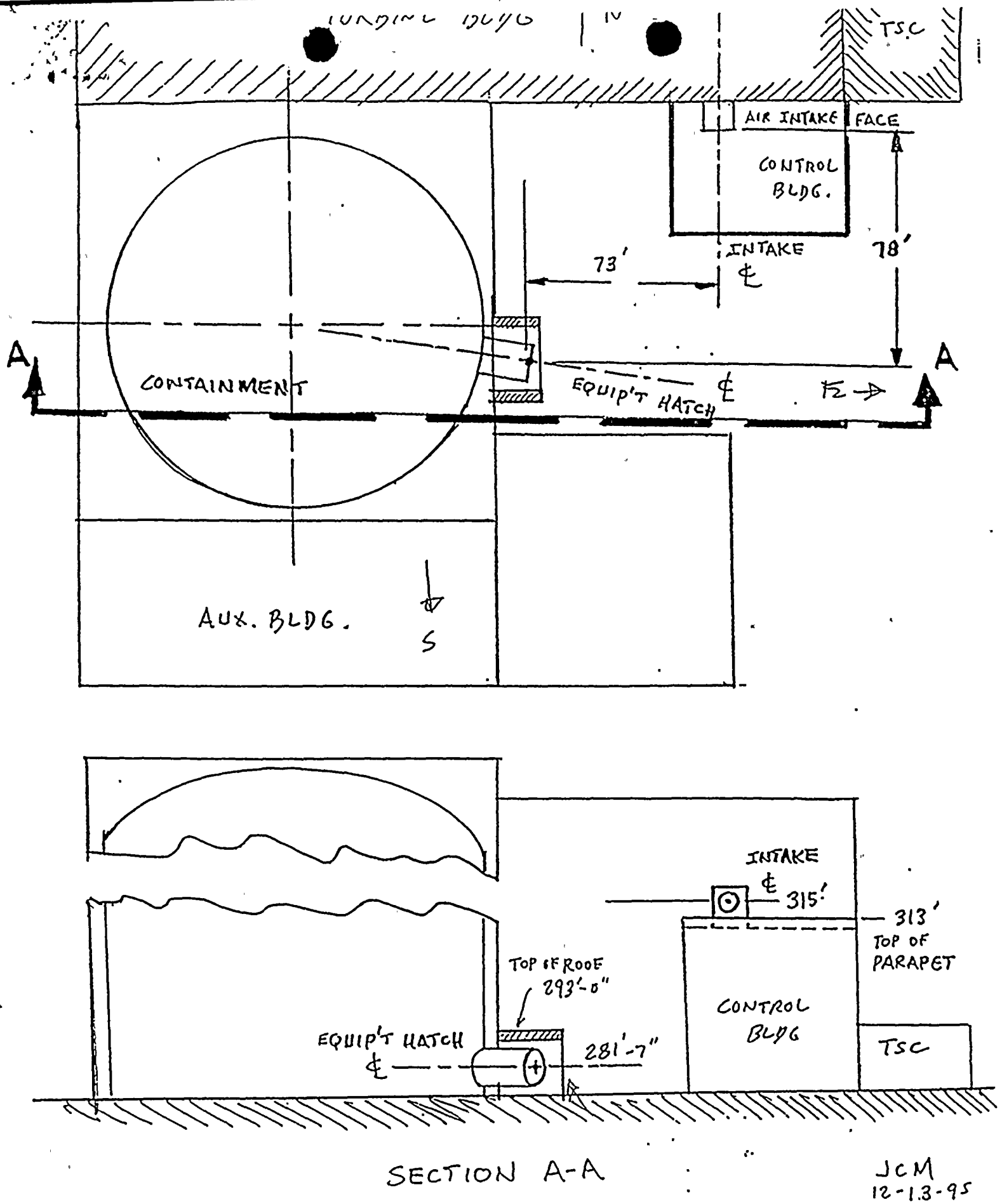
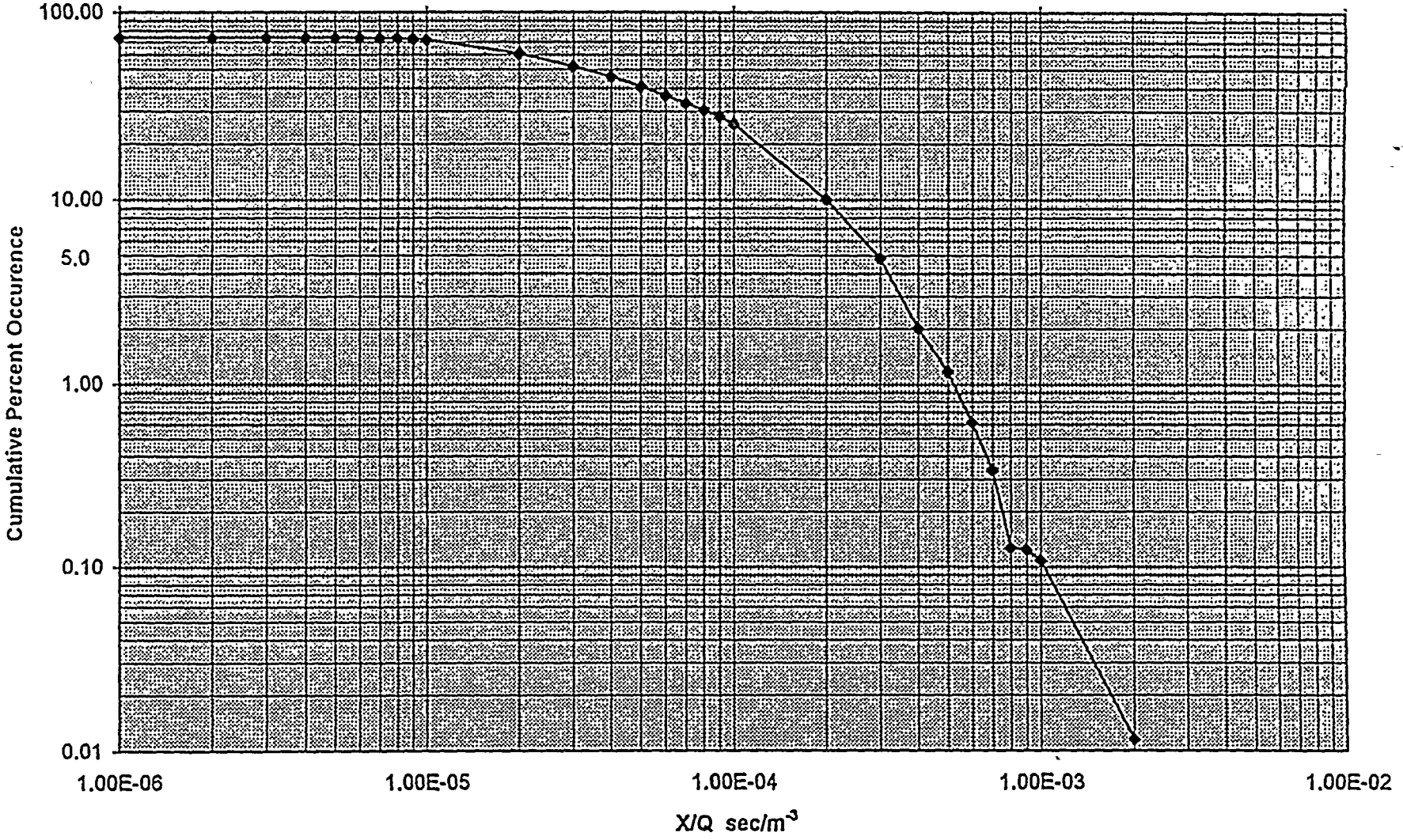


FIGURE 1. PLANT CONFIGURATION

Figure 2 - X/Q Using 1992-1994 Meteorological Data



1/15/96

25851 hours
36 m release