12 RADIATION PROTECTION

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE

12.1.1 ALARA PROGRAM

The Radiation Protection program at Ginna Station shall ensure that internal and external radiation exposures to station personnel, contractor personnel, and the general population resulting from station operation, including anticipated operational occurrences, will be within applicable limits and will be as low as is reasonably achievable (ALARA).

Rochester Gas and Electric Corporation (RG&E) has developed a Nuclear Directive entitled ALARA, which establishes the Nuclear Operations Group policy with regard to maintaining occupational radiation exposure ALARA and defines the authority and responsibilities of corporate and plant personnel relevant to the Nuclear Directive.

The Nuclear Directive applies to all RG&E personnel, contractors, vendors, and station visitors who are subject to occupational radiation exposure at the Ginna Station and to those personnel whose assignments involve design, construction, or operational activities having a significant impact on the current or future occupational radiation exposure of station personnel.

12.1.2 ORGANIZATIONAL RESPONSIBILITIES

The President is responsible for assuring the occupational radiation exposure of all personnel in the Nuclear Operations Group to be ALARA in accordance with the Nuclear Directive.

The Vice President, Nuclear Operations Group, is responsible for development, implementation, and maintenance of the ALARA Program and is responsible to ensure the ALARA commitments of the Nuclear Directive are properly implemented in the Nuclear Operations Group.

12.1.3 RADIATION PROTECTION PROGRAM

The bases of the Ginna Station Radiation Protection Program are that doses to personnel will be maintained within the limits of 10 CFR 20 and that the radiation protection designs and program features are consistent with the guidelines of Regulatory Guide 8.8, Revision 3. Shielding is provided to reduce levels of radiation. Ventilation is arranged to control the flow of potentially contaminated air. Radiation monitoring systems are used to measure levels of radiation in potentially occupied areas and to measure airborne radioactivity throughout the plant. Use of respiratory protective equipment is used as stipulated in Regulatory Guide 8.15. The Radiation Protection Program is provided for plant personnel and visitors during reactor operation, maintenance, MODE 6 (Refueling), radwaste handling, and inservice inspection. ALARA procedures are in place that govern all activities in restricted areas at Ginna Station. The Radiation Protection Program is organized and maintained to meet the requirements of 10 CFR 20 with approved exceptions specified in the Technical Specifications. The program is adhered to for all operations involving personnel radiation exposure.

12.2 RADIATION SOURCES

The radiation source assumptions discussed below are those used for the post-accident shielding design review(*Reference 1*) performed according to the Three Mile Island Lessons Learned Short-Term Requirements (NUREG 0737, Item II.B.2).

The activity assumed for liquid source-term calculation is based on 100% of the noble gas inventory, 50% of the halogen core inventory, and 1% of all other nuclides in the core inventory. The activity assumed for gaseous source-term calculation is based on 100% of the noble gas core inventory and 25% of the halogen core inventory.

Two liquid source terms were used in the evaluation. For systems which contain postaccident recirculation fluid, the source term was based on diluting the liquid inventory discussed above with 303,800 gal of fluid filling the containment sump from the refueling water storage tank (RWST), both accumulators, and a boric acid storage tank. These systems include the following:

A. Residual heat removal.

B. Containment spray recirculation.

C. High-pressure injection.

D. Nuclear sampling (residual heat removal process fluid).

For systems which can contain fluid from the reactor coolant system but do not take suction on the containment sump, the source term was based on diluting the liquid inventory with the 46,600 gal in the reactor coolant system. This source was used for the nuclear sampling system (reactor coolant fluid) and the portion of the chemical and volume control system associated with reactor coolant degassing.

Gaseous source terms were determined for containment and for the waste gas system. The containment airborne source term was based on diluting the gaseous inventory discussed previously with the air contained in the containment free volume (970,000 ft^3). The waste gas system source term was determined for a reactor coolant degassing operation by calculating the quantity of activity entering the volume control tank via the normal letdown path and assuming that a quantity equal to the stripping fraction (based upon a stripping efficiency of 1.0) enters the vapor space and is immediately purged to the vent header system. The stripping fraction is defined as

$$SF = (CR - CL) / CR$$

where: **SF = stripping fraction**

CR = concentration entering volume control tank in liquid

CL = concentration leaving volume control tank in liquid

Table 12.2-1 provides the shielding source terms for liquid and gaseous radioactivity which were calculated using the assumptions above.

A Technical Specification change included the isolation of the lines from the boric acid storage tank to the safety injection pumps. Consequently, the quantity of boric acid assumed in the analysis from a boric acid storage tank would not enter the containment sump. Since approximately 1% (3000 gal) of the total of 303,800 gal would not enter the sump in the current plant configuration, the dose rates would be expected to increase approximately 1% and be manifested by the containment sump concentration values in Table 12.2-1. This small increase is not significant in terms of the shielding previously provided.

The impact of core power uprate on post accident radiation source terms was evaluated in *Reference 2*.

The uprate source terms are based on an analyzed core power level of 1811 MWt, and use of an 18-month fuel cycle. Consistent with original licensing basis, the activity assumed for the liquid source-term calculation is based on 100% of the noble gas core inventory, 50% of the halogen core inventory, and 1% of all the other nuclides in the core inventory, and the activity assumed for the gaseous source-term calculation is based on 100% of the noble gas core inventory and 25% of the halogen core inventory. The estimated isotopic activity of dose significant isotopes following core power uprate in the liquid source and the gaseous source are presented in Table 12.2-2.

REFERENCES FOR SECTION 12.2

1. Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post-Accident Operations Outside Containment at R.E. Ginna Nuclear Power Plant, dated December 31, 1979.

2. Letter from Constellation Energy to the NRC, subject: Licensing Amendment Request Regrading Extended Uprate, dated July 7, 2005.

Table 12.2-1
SHIELDING SOURCE TERMS (T=0) (HISTORICAL)

Isotope	Liquid[a] Source Activity (Ci)	Gaseous[b] Source Activity (Ci)	Containment Sump Concentration ($\mu Ci/cm^3$)	Reactor Coolant Concentration ($\mu Ci/cm^3$)	Containment Airborne Concentration ($\mu Ci/cm^3$)	Waste Gas Concentration ($\mu Ci/cm^3$)
^{84}Br	3.9×10^6	2.0×10^7	3.5×10^3	2.2×10^4	7.2×10	3.4×10^3
^{87}Kr	1.9×10^7	1.9×10^7	1.7×10^4	1.1×10^5	7.0×10^2	6.7×10^5
^{133}Te	2.3×10^5	---	2.1×10^2	1.3×10^3	---	---
^{134}Cs	8.9×10^4	---	8.0×10	5.0×10^2	---	--
^{136}Cs	2.5×10^4	---	2.2×10	1.4×10^2	---	---
^{137}Cs	3.7×10^4	---	3.3×10	2.1×10^2	---	---
^{139}Ba	7.9×10^5	---	7.0×10^2	4.4×10^3	---	---
^{83}Br	1.7×10^6	8.4×10^5	1.5×10^3	9.5×10^3	3.0×10	1.9×10^2
^{83m}Kr	3.3×10^6	3.3×10^6	3.0×10^3	1.9×10^4	1.2×10^2	1.1×10^5
^{85m}Kr	1.0×10^7	1.0×10^7	8.9×10^3	5.6×10^4	3.6×10^2	2.6×10^5
^{85}Kr	3.9×10^5	3.9×10^5	3.5×10^2	2.2×10^3	1.4×10	5.9×10^3
^{88}Kr	2.9×10^7	2.9×10^7	2.6×10^4	1.6×10^5	1.0×10^3	8.5×10^5
^{88}Rb	2.8×10^5	---	2.5×10^2	1.6×10^3	---	---
^{89}Rb	3.9×10^5	---	3.2×10^2	2.0×10^3	---	---
^{89}Sr	3.9×10^5	---	3.5×10^2	2.2×10^3	---	---
^{90}Sr	2.6×10^4	---	2.3×10	1.5×10^2	---	---
^{90}Y	2.6×10^4	---	2.3×10	1.5×10^2	---	---
^{92}Sr	4.8×10^5	---	4.3×10^2	2.7×10^3	---	---

Isotope	Liquid[a] Source Activity (Ci)	Gaseous[b] Source Activity (Ci)	Containment Sump Concentration ($\mu Ci/cm^3$)	Reactor Coolant Concentration ($\mu Ci/cm^3$)	Containment Airborne Concentration ($\mu Ci/cm^3$)	Waste Gas Concentration ($\mu Ci/cm^3$)
^{92}Y	5.2×10^5	---	4.7×10^2	3.0×10^2	---	---
^{93}Sr	5.8×10^5	---	5.2×10^2	3.3×10^3	---	---
^{99}Mo	7.9×10^5	---	7.0×10^2	4.4×10^3	---	---
^{99m}Tc	6.8×10^5	---	6.1×10^2	3.9×10^3	---	---
^{103}Ru	5.8×10^5	---	5.2×10^2	3.3×10^3	---	---
^{103m}Rh	5.8×10^5	---	5.2×10^2	3.2×10^3	---	--
^{106}Ru	2.2×10^5	---	1.9×10^2	1.2×10^3	---	---
^{106}Rh	2.2×10^5	---	1.9×10^2	1.2×10^3	---	---
^{132}Te	5.8×10^5	---	5.2×10^2	3.3×10^3	---	---
^{132}I	3.1×10^7	1.5×10^7	2.7×10^4	1.8×10^5	5.6×10^2	2.7×10^4
^{134}Te	8.4×10^5	---	7.5×10^2	4.8×10^3	---	---
^{134}I	4.7×10^7	2.3×10^7	4.2×10^4	2.6×10^5	8.5×10^2	4.0×10^4
^{138}Xe	7.9×10^7	7.9×10^7	7.0×10^4	4.4×10^5	2.9×10^3	3.4×10^6
^{138}Cs	7.9×10^5	---	7.0×10^2	4.4×10^3	---	---
^{140}Ba	7.3×10^5	---	6.6×10^2	4.2×10^3	---	---
^{140}La	7.9×10^5	---	7.0×10^2	4.4×10^3	---	---
^{143}Ce	6.3×10^5	---	5.6×10^2	3.6×10^3	---	---
^{143}Pr	6.3×10^5	---	5.6×10^2	3.6×10^3	---	---
^{144}Ce	4.8×10^5	---	4.3×10^2	2.7×10^3	---	---
^{144}Pr	6.3×10^5	---	5.6×10^2	3.6×10^3	---	---

Isotope	Liquid[a] Source Activity (Ci)	Gaseous[b] Source Activity (Ci)	Containment Sump Concentration ($\mu Ci/cm^3$)	Reactor Coolant Concentration ($\mu Ci/cm^3$)	Containment Airborne Concentration ($\mu Ci/cm^3$)	Waste Gas Concentration ($\mu Ci/cm^3$)
^{91}Sr	4.8×10^5	---	4.3×10^2	2.7×10^3	---	---
^{91m}Y	---	---	---	---	---	---
^{91}Y	5.0×10^5	---	4.5×10^2	2.8 x 103	---	---
^{95}Zr	6.8×10^5	---	6.1×10^2	3.9×10^3	---	---
^{95m}Nb	---	---	---	---	---	---
^{95}Nb	6.8×10^5	---	6.1×10^2	3.9×10^3	---	---
^{97}Zr	6.8×10^5	---	6.1×10^2	3.9×10^3	---	---
^{97m}Nb	---	---	---	---	---	---
^{97}Nb	6.8×10^5	---	6.1×10^2	3.9×10^3	---	---
^{105}Ru	4.5×10^5	---	4.0×10^2	2.6×10^3	---	---
^{105m}Rh	4.5×10^5	---	4.0×10^2	2.6×10^3	---	---
^{105}Rh	2.9×10^5	---	2.8×10^2	1.6×10^3	---	---
^{131}Te	3.8×10^5	---	3.4×10^2	2.1×10^3	---	--
^{131}I	2.2×10^7	1.1×10^7	1.9×10^4	1.2×10^5	3.9×10^2	1.9×10^4
^{131m}Xe	3.4×10^5	3.3×10^5	3.0×10^2	1.9×10^3	1.2 x 10	5.8×10^3
^{133}I	4.1×10^7	2.0×10^7	3.7×10^4	2.3×10^5	7.4×10^2	3.5×10^4
^{133m}Xe	2.0×10^6	2.0×10^6	1.8×10^3	1.1×10^4	7.2 x 10	3.7×10^4
^{133}Xe	7.9×10^7	7.9×10^7	7.0×10^4	4.4×10^5	2.9×10^3	1.4×10^6
^{135}I	3.6×10^7	1.8×10^7	3.2×10^4	2.0×10^5	6.6×10^2	3.1×10^4
^{135m}Xe	2.2×10^7	2.2×10^7	2.0×10^4	1.2×10^5	8.0×10^2	9.1×10^5

Isotope	Liquid[a] Source Activity (Ci)	Gaseous[b] Source Activity (Ci)	Containment Sump Concentration ($\mu Ci/cm^3$)	Reactor Coolant Concentration ($\mu Ci/cm^3$)	Containment Airborne Concentration ($\mu Ci/cm^3$)	Waste Gas Concentration ($\mu Ci/cm^3$)
^{135}Xe	1.5×10^7	1.5×10^7	1.4×10^4	8.6×10^4	5.5×10^2	2.6×10^4
^{141}Ba	6.8×10^5	---	6.1×10^2	3.9×10^3	---	---
^{141}La	6.8×10^5	---	6.1×10^2	3.9×10^3	---	---
^{141}Ce	7.3×10^5	--	6.6×10^2	4.2×10^3	---	---

Note: The information presented in the above Table represents the Source Terms used in the original post-accident shielding assessment documented in Reference 1 and is retained herein for historical purposes only.

a. Based on 100% noble gas core inventory, 50% halogen core inventory, and 1% of all other core inventory.

b. Based on 100% noble gas core inventory, and 25% halogen core inventory.

Table 12.2-2
SHIELDING SOURCE ACTIVITY at T=0 hrs - Power Level 1811 MWt

Isotope[a]	Liquid[b] Source Activity (CI)	Gaseous[c] Source Activity (CI)
KR-83M	6.46E+06	6.46E+06
KR-85	5.85E+05	5.85E+05
KR-85M	1.36E+07	1.36E+07
KR-87	2.62E+07	2.62E+07
KR-88	3.68E+07	3.68E+07
KR-89	4.50E+07	4.50E+07
KR-90	4.45E+07	4.45E+07
XE-131M	5.59E+05	5.59E+05
XE-133	1.01E+08	1.01E+08
XE-133M	3.17E+06	3.17E+06
XE-135	2.56E+07	2.56E+07
XE-135M	2.04E+07	2.04E+07
XE-137	9.08E+07	9.08E+07
XE-138	8.61E+07	8.61E+07
BR-82	1.69E+05	8.45E+04
BR-83	3.21E+06	1.61E+06
BR-85	6.70E+06	>3.35E+06
I-129	9.15E-01	4.58E-01
I-130	1.05E+06	5.25E+05
I-131	2.54E+07	1.27E+07
I-132	3.76E+07	1.88E+07
I-133	5.15E+07	2.58E+07
I-134	5.70E+07	2.85E+07
I-135	4.86E+07	2.43E+07
I-136	2.31E+07	1.15E+07
RB-86	1.30E+03	--

Isotope[a]	Liquid[b] Source Activity (CI)	Gaseous[c] Source Activity (CI)
RB-88	3.74E+05	--
RB-89	4.80E+05	--
RB-90	4.66E+05	--
RB-90M	1.09E+05	--
CS-132	1.42E+02	--
CS-134	1.10E+05	--
CS-134M	3.08E+04	--
CS-135M	1.98E+04	--
CS-136	3.24E+04	--
CS-137	6.30E+04	--
CS-138	9.54E+05	>--
CS-139	9.03E+05	--
CS-140	8.13E+05	--
GA-72	3.53E+01	--
AS-76	2.22E+01	--
GE-77	5.16E+02	--
SE-83	2.50E+04	--
AG-110M	2.20E+03	--
AG-110	8.48E+04	--
AG-111	3.44E+04	--
AG-112	1.84E+04	--
CD-115M	9.00E+02	--
IN-115M	9.77E+03	--
CD-115	9.73E+03	--
SN-121	8.53E+03	--
SB-122	1.34E+03	--
SB-123	1.85E+03	--
SB-124	8.11E+02	--
SB-125	7.62E+03	--

Isotope[a]	Liquid[b] Source Activity (CI)	Gaseous[c] Source Activity (CI)
SN-125	6.86E+03	--
TE-127M	7.02E+03	--
SB-127	5.45E+04	--
SN-127	3.49E+04	--
TE-127	5.40E+04	--
TE-129M	2.38E+04	--
SB-129	1.63E+05	--
TE-129	1.62E+05	--
SB-130M	2.39E+05	--
SB-130	5.30E+04	--
TE-131M	7.32E+04	--
SB-131	4.26E+05	--
TE-131	4.52E+05	--
SB-132M	1.67E+05	--
SB-132	2.54E+05	--
TE-132	7.22E+05	--
TE-133M	3.82E+05	--
SB-133	2.98E+05	--
TE-133<	6.10E+05	--
TE-134	8.65E+05	--
SR-89	4.95E+05	--
SR-90	4.63E+04	--
SR-91	6.20E+05	--
SR-92	6.70E+05	--
SR-93	7.57E+05	--
SR-93	7.14E+05	--
BA-137M	5.97E+04	--
BA-139	9.31E+05	--
BA-140	8.94E+05	--
BA-142	8.03E+05	--

Isotope[a]	Liquid[b] Source Activity (CI)	Gaseous[c] Source Activity (CI)
MO-99	9.67E+05	--
TC-99M	8.49E+05	--
MO-101	8.67E+05	--
TC-101	8.68E+05	--
RH-103M	6.95E+05	--
TC-104	6.43E+05	--
RH-105	4.68E+05	--
RH-105M	1.46E+05	--
TC-105	5.17E+05	--
RH-106	2.92E+05	--
RU-103	7.71E+05	--
RU-106	2.61E+05>	--
PD-109	1.64E+05	--
CE-141	8.48E+05	--
CE-143	7.89E+05	--
CE-144	6.46E+05	--
NP-239	1.05E+07	--
PU-238	2.23E+03	--
PU-239	1.85E+02	--
PU-240	2.79E+02	--
PU-241	6.15E+04	--
PU-242	8.40E-01	--
Y-90	4.84E+04	--
Y-91M	3.66E+05	--
Y-91	6.38E+05	--
Y-92	6.73E+05	--
Y-93	7.73E+05>	--
Y-94	7.80E+05	--
NB-95M	6.07E+03	--
NB-95	8.66E+05	--

Isotope[a]	Liquid[b] Source Activity (CI)	Gaseous[c] Source Activity (CI)
Y-95	8.38E+05	--
ZR-95	8.60E+05	--
NB-97M	8.15E+05	--
NB-97	8.68E+05	--
ZR-97	8.59E+05	--
LA-140	9.26E+05	--
LA-141	8.35E+05	--
LA-142	8.22E+05	--
PR-142	4.76E+04	--
LA-143	7.86E+05	--
PR-143	7.74E+05>	--
PR-144	6.52E+05	--
ND-147	3.39E+05	--
PM-147	6.62E+04	--
PM-148M	1.59E+04	--
PM-148	1.45E+05	--
PM-149	3.19E+05	--
PM-151	1.00E+05	--
SM-153	2.58E+05	--
EU-154	6.14E+03	--
EU-155	4.14E+03	--
EU-156	1.03E+05	--
EU-157	1.34E+04	--
EU-158	4.49E+03	--
EU-159	2.38E+03	
GD-159	3.25E+03	--
TB-160	5.61E+02	--
HO-166	7.80E+01	--
AM-241	7.55E+01	--
CM-242	1.75E+04	--

Isotope[a]	Liquid[b] Source Activity (CI)	Gaseous[c] Source Activity (CI)
M-244	2.06E+03	--

a. Based on 100% noble gas core inventory, 50% halogen core inventory, and 1% of all other core inventory

b. Based on 100% noble gas core inventory, 25% halogen core inventory

c. Dose significant isotopes

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 DESIGN CRITERIA

12.3.1.1 Conformance to 1967 Design Criteria

The following design criteria were used during the licensing of Ginna Station. They represent the Atomic Industrial Forum version of proposed criteria issued by the AEC for comment on July 10, 1967, and discussed in Section 3.1.1.

CRITERION: Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels (AIF-GDC 18).

Monitoring and alarm instrumentation is provided for fuel and waste storage and handling areas to detect inadequate cooling and to detect excessive radiation levels. Radiation monitors are provided to maintain surveillance over the release operation.

The spent fuel pool (SFP) cooling system flow is monitored to ensure proper operation as described in Section 9.1.3.

A controlled ventilation system removes gaseous radioactivity from the atmosphere and fuel storage and waste treatment areas of the auxiliary building and discharges it to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate high activity alarms on the control board annunciator, as described in Sections 11.5 and 12.3.

CRITERION: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities (AIF-GDC 68).

Auxiliary shielding for the waste disposal system and its storage components is designed to limit the dose rates as required by personnel access, testing, operation, and maintenance requirements.

Gamma radiation is continuously monitored in the auxiliary building. A high-level signal is alarmed locally and annunciated in the control room.

12.3.1.2 Conformance to 1972 Design Criteria

Conformance to the requirements of 1972 General Design Criteria 19, 61, and 63 of 10 CFR 50, Appendix A, is discussed in Section 3.1.2. With respect to these general design criteria:

A. Sufficient shielding, distance, and containment integrity are provided to ensure that control room personnel shall not be subject to doses under postulated accident conditions during the occupancy of the control room which, in the aggregate, would exceed the limits of GDC-19.

B. All areas of the plant are designed with suitable shielding for radiation protection based on anticipated radiation rates and occupancy, and individual components which contain

significant radioactivity are located in confined areas which are adequately ventilated through appropriate filtering systems.

C. Radiation monitors and alarms are provided as required to warn personnel of impending excessive levels of radiation or airborne activity.

Conformance to Safety/Regulatory Guides in effect as of August 1972 is discussed in Section 1.8.

12.3.2 SHIELDING

12.3.2.1 Design Basis

Radiation shielding is designed for operation at maximum calculated thermal power and to limit the MODES 1 and 2 radiation levels at the site boundary to below those levels allowed for continuous nonoccupational exposure.

The original design basis for normal operation plant shielding is safe operation at a core power level of 1520 MWt, a one year fuel cycle length, and conservative reactor coolant source terms assuming 1% fuel defects. The design basis target dose rates in plant areas and the associated shielding design, presented in Section 12.3.2.2 below, are based on the above design basis.

Power uprate represents a change from the original design basis. The assessment of impact of power uprate on adequacy of existing plant shielding was evaluated based on scaling techniques that took into consideration the radiation source terms used in the original plant shielding design as discussed in UFSAR Section 12.3.2.1.1 through 12.3.2.2.5 below, and the uprate source terms, specifically, the design basis fission and corrosion product activity concentrations in the reactor coolant at the analyzed core power level of 1811 MWt and with an 18-month fuel cycle length (*Reference 4*).

Inside containment, continued adequacy of the primary shield following uprate was determined based on usage of a low leakage fuel management scheme, and review of fluence calculations that confirmed that the original design calculations remain bounding for uprate conditions. Due to the conservative analytical techniques used to establish original plant shielding design, and a comparison of the design basis target dose rates to pre-uprate survey data, the existing reactor secondary shield and the fuel handling shields were also determined adequate to address the approximately 19% increase in radiation source terms expected due to the uprate.

Shielding adequacy outside containment (where the radiation sources are either the reactor coolant itself, or down-stream sources originating from coolant activity), was determined by an evaluation that compared the uprate design primary coolant source terms (fission and activation products) to the original design basis primary coolant source terms. Three sources were considered: total primary coolant, degassed primary coolant and the primary coolant noble gas source. Due to the change in isotropic compositions and gamma energy spectrum between the original and the uprate reactor coolant fluid, the comparison was based on the dose rate resulting from the above sources shielded by 0, 1, 2, and 3 ft of concrete for representative source geometry.

The comparison showed that the ratio of the calculated dose rates resulting from the uprate source to the original design basis source, for the various design basis source term/shielding configurations discussed above, range from 1.0 to 2.4. However, since the design basis uprate primary coolant activity is a very conservative source term (i.e., based on 1% failed fuel, a 2% margin for power uncertainty and an additional 4% margin for fuel management schemes), credit was taken for a more realistic but limiting upper bound primary coolant activity based on the plant Technical Specifications.

The uprate assessment conclude that the Plant Technical Specifications will limit the uprate reactor coolant, degassed reactor coolant and reactor coolant gas activity and the associated dose rates assuming various shielding configurations, to less than or equal to the original design basis values.

Thus, taking into consideration the conservative analytical techniques used to establish the original shielding design, and the Plant Technical Specifications which typically restrict the reactor coolant activity to levels significantly less than 1% fuel defects, the increase in the core power level and fuel cycle length is expected to have no significant impact on plant shielding adequacy and safe plant operation.

It is noted that operating personnel at the station are protected by adequate shielding, monitoring, and operating procedures. Individual worker exposure is maintained within acceptable limits by the site ALARA Program, which controls access to radiation areas. Procedural controls compensate, as necessary, for increased radiation levels to ensure that operator exposure remains ALARA, and that the normal operation radiation zones are labeled and controlled for access in accordance with the requirements of 10CFR20 related to allowable operator exposure and access control.

All plant areas capable of personnel occupancy are considered as one of the five zones of radiation level listed in Table 12.3-1.

Typical Zone 0 areas are the turbine building and turbine plant service areas. Typical Zone I areas are the offices and control room. Zone II areas include the local control spaces in the auxiliary building and the operating deck of the containment during reactor shutdown. Areas designated Zone III include grade level areas adjacent to the containment structures, fuel handling areas, and intermittently occupied work areas. Typical Zone IV areas are the region adjacent to the reactor coolant system after reactor shutdown, waste drumming areas, and volume control tank spaces.

All radiation and high-radiation areas are appropriately marked and isolated in accordance with 10 CFR 20 and other applicable regulations.

The shielding is divided into five categories according to function. These categories include the primary shielding, the secondary shielding, the containment structure, the fuel transfer (water and pool) shielding, and the auxiliary shielding.

In addition, plant shielding ensures that in the event of postulated accidents, the integrated offsite exposure due to the contained activity does not exceed established criteria. See UFSAR Section 12.3.2.2.6 for additional detail.

12.3.2.2 Shielding Design

12.3.2.2.1 Primary Shield

The primary shield consists of the core baffle, water annuli, barrel-thermal shield (all of which are within the reactor vessel), the reactor vessel wall, and a concrete structure surrounding the reactor vessel.

The primary shield is designed to

A. Reduce the neutron fluxes incident on the reactor vessel to limit the radiation induced increase in transition temperature.

B. Attenuate the neutron flux sufficiently to prevent excessive activation of plant components.

C. Limit gamma radiation from the reactor vessel to avoid excessive temperature gradients or dehydration of the concrete structure surrounding the reactor vessel.

D. Reduce the residual radiation from the core, reactor internals, and reactor vessel to levels that will permit access to the region between the primary and secondary shields after plant shutdown.

E. Reduce radiation leakage to obtain optimum division of the shielding between the primary and secondary shields.

The primary shield immediately surrounding the reactor vessel consists of an annular reinforced-concrete structure extending from the base of the containment to an elevation of 252.5 ft. The lower portion of the shield is a minimum thickness of 6.5 ft of regular concrete (density = 2.3 g/cm^3) and is an integral part of the main structural concrete support for the reactor vessel. It extends upward to join the concrete cavity over the reactor. The reactor cavity, which is approximately rectangular in shape, extends upward to the operating floor with vertical walls 4-ft thick, except in the area adjacent to fuel handling, where the thickness is increased to 6 ft. A steel plate is provided at each point where the four reactor coolant pipes penetrate the primary shield.

The primary concrete shield is air cooled to prevent overheating and dehydration from the heat generated by radiation adsorption in the concrete. Eight "windows" have been provided in the primary shield for insertion of the out-of-core nuclear instrumentation. Cooling for the primary shield concrete and the nuclear instrumentation is provided by the reactor compartment coolers.

The primary shield neutron fluxes and design parameters are listed in Table 12.3-2.

12.3.2.2.2 Secondary Shield

The secondary shield surrounds the reactor coolant loops and the primary shield. It consists of interior walls surrounding the reactor coolant loop in the containment structure and the operating floor.

The main function of the secondary shielding is to attenuate the radiation originating in the reactor and the reactor coolant. The major source in the reactor coolant is the Nitrogen-16

activity, which is produced by neutron activation of oxygen during passage of the coolant through the core. The secondary shielding will help to limit the full-power dose rate outside the containment due to activity within the containment to less than 1 mrem/hr.

The secondary shield design parameters are listed in Table 12.3-3.

12.3.2.2.3 **Containment Structure**

The containment structure consists of a 3 ft 6 in. thick reinforced-concrete cylinder capped by a 2 ft 6 in. thick hemispherical reinforced-concrete dome. These thicknesses are nominal values. The true relevant engineering values are dependent on the specific location in the structure and the loading condition that is present. It also includes supplemental shields in front of the containment penetrations and 20-in. walls and roof of the control room.

The main purpose of the containment structure is to ensure safe radiation levels outside the containment following a design-basis accident.

The equipment access hatch is shielded by a 3-ft thick concrete shadow shield and a 1-ft thick concrete roof to reduce scattered dose levels in the event of a loss-of-coolant accident.

The containment structure design parameters are listed in Table 12.3-4.

12.3.2.2.4 **Fuel Handling (Water and Pool) Shield**

During fuel handling, shielding is provided to facilitate the removal and transfer of spent fuel assemblies and control rods from the reactor vessel to the spent fuel pool (SFP). It is designed to attenuate radiation from spent fuel, control rods, and reactor vessel internals to less than 2.5 mrem/hr at the refueling cavity water surface and less than 1.0 mrem/hr in the auxiliary building general areas.

The refueling cavity, flooded to an elevation of 277.0 ft during MODE 6 (Refueling) operations, provides a water shield above the components being withdrawn from the reactor vessel. The water height during MODE 6 (Refueling) is at least 23 ft above the reactor vessel flange. This height ensures that a minimum of 10 ft 2 in. of water will be above the active fuel of a withdrawn fuel assembly.

The refueling canal is a passageway connected to the reactor cavity and extending to the inside surface of the containment. The canal is formed by two concrete walls each 6-ft thick, which extend upward to the same height as the reactor cavity. During MODE 6 (Refueling) the canal is flooded with borated water to the same height as the reactor cavity.

The spent fuel assemblies and control rod clusters are remotely removed from the containment through the horizontal spent fuel transfer tube and placed in the spent fuel pool (SFP). Concrete, 6-ft thick, shields the spent fuel transfer tube. Additional lead and steel shielding has been added to the joint where the 6-ft thick walls met the containment wall in the auxiliary and intermediate buildings. The additional shielding is designed to attenuate radiation to less than 100 mrem/hr adjacent to the joint. This shielding is designed to protect personnel from radiation during the time a spent fuel assembly is passing through the main concrete support of the containment and the transfer tube.

Radial shielding during fuel transfer is provided by the water and concrete walls of the fuel transfer tube. An equivalent of 6 ft of regular concrete is provided to ensure a maximum dose value of 1.0 mrem/hr in the auxiliary building areas adjacent to the spent fuel pool (SFP).

Spent fuel is stored in the spent fuel pool (SFP) which is located adjacent to the containment. Shielding for the spent fuel storage pool is provided by 6-ft thick concrete walls and a minimum water level of 12 ft 6 in. above the spent fuel assemblies when withdrawn for movement.

The fuel handling shield design parameters are listed in Table 12.3-5.

12.3.2.2.5 **Auxiliary Shielding**

The auxiliary shielding consists of concrete walls around certain components and piping which process reactor coolant. In some cases, the concrete block walls are removable to allow personnel access to equipment during maintenance periods. The function of the auxiliary shielding is to protect personnel working near various system components in the chemical and volume control system, the residual heat removal system, the waste disposal system, and the sampling system. Access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded so that compartments may be entered without having to shut down.

The shield material provided throughout the auxiliary building is regular concrete (density = 2.3 g/cm^3). The principal auxiliary shielding provided is tabulated in Table 12.3-6.

12.3.2.2.6 **Shielding Design Modifications**

Rochester Gas and Electric conducted a plant radiation and shielding design review of vital areas and equipment in order to ensure adequate personnel access to vital areas and protection of safety equipment for post design-basis accident operations. This was done in response to NUREG 0737, Item II.B.2. As a result of the review, the following shielding modifications were made.

A. Increase of height of shield wall across from the nuclear sample room.

B. Installation of a lead door on the penetration area shield.

C. Relocation of the vent header line in the auxiliary building intermediate level and shielding of a portion of the vent header.

D. Installation of lead shielding on the east wall of the count room.

The shielding design review and the corrective actions taken for postaccident vital area access, which was based on the original licensed power level and a 12-month fuel cycle length, were reviewed by the NRC and the staff concluded that the requirements of NUREG 0737, Item II.B.2.2 had been met (*Reference 1*). The impact of core power uprate to 1775 MWt and operation with an 18 month fuel cycle has been evaluated and it is determined that the post accident operator exposure will remain within the regulatory limits set by NUREG 0737, Item II.B.2 (*Reference 4*).

In 1978, NRC IE Bulletin 78-08 (*Reference 2*) had requested that actions be taken to identify potential high radiation areas associated with fuel element transfer tubes. Surveys taken at that time did not identify any areas of concern. However, in 1999, radiation streaming was discovered adjacent to the fuel transfer tube, which could create a high radiation area during the movement of irradiated spent fuel. Rochester Gas and Electric provided a revised response to Bulletin 78-08 (*Reference 3*). Additional shielding was installed to attenuate this radiation streaming. As stated in Section 12.3.2.2.4, the shielding will attenuate the radiation to less than 1.0 mrem/hr in general areas, and to less than 100 mrem/hr adjacent to the joint during the time a spent fuel assembly is passing through the main concrete support of the containment and the transfer tube.

12.3.2.2.7 **Containment Accessibility Procedure**

The containment is completely closed whenever the core is critical or whenever the primary system is above MODE 5 (Cold Shutdown) with nuclear fuel in place. Limited access to the containment through personnel air locks is possible with the reactor above MODE 5 (Cold Shutdown). This type of access is restricted to the areas external to the reactor equipment compartment, primarily for inspection, testing and maintenance.

Prior to personnel entry into the containment, the containment atmosphere is sampled and purged, if necessary, using the containment mini-purge system to reduce the concentration of radioactive gases and airborne particulates (see Sections 6.2.4.4.9 and 9.4.1.2.9). This and the containment auxiliary filtering system (see Section 9.4.1.2.6) are used to reduce activity levels inside the containment. The containment mini-purge system is designed to achieve a 99% reduction (0.01 dilution factor) of contaminants in the containment atmosphere in 38 hour for 2000 cfm operation, in 51 hour for 1500 cfm, and in 76 hour for 1000 cfm. After containment entry has been achieved, with the unit in cold or MODE 6 (Refueling) shutdown, the blind flanges for the containment shutdown purge system may be removed and that system may be utilized for purging (see Section 9.4.1.2.8).

The primary reactor shield is designed so that access to the primary equipment compartment is limited by the activity of the primary system equipment and not the reactor. Opening of the containment equipment hatch or both doors simultaneously in the personnel locks can occur during MODE 5 (Cold Shutdown) or MODE 6 (Refueling) conditions including during core alterations and during movement of irradiated fuel assemblies within containment in accordance with the requirements of Technical Specification 3.9.3 (Containment Penetration).

MODE 6 (Refueling) is carried out with borated water in the refueling cavity so that the core is always in a substantially subcritical condition. Operations which may change core reactivity, such as replacement of fuel or control rods, are performed only when the containment penetrations are configured as described in the Technical Specifications. The fuel transfer penetration tube is under a 40-ft water seal at this time.

12.3.3 VENTILATION

The ventilation systems in Ginna Station are discussed in Section 9.4. The radiation protection aspects during normal full-power operation are associated with the following systems:

A. Gas collection and decay tank system.

B. Plant ventilation system.

C. Containment ventilation system.

D. Air ejector and gland seal exhaust system.

A block diagram of the major systems is shown in the Technical Specifications.

12.3.3.1 Gas Collection and Decay Tank System

All tanks that receive primary coolant are provided with a nitrogen cover gas. These tanks include the three chemical and volume control system holdup tanks, the volume control tank, the pressurizer relief tank, and the reactor coolant drain tank. Most of the gas is contained in the chemical and volume control tanks. The cover gas must be maintained oxygen-free to prevent explosive mixtures with hydrogen released from the primary coolant. When primary coolant is letdown to the chemical and volume control tanks, the cover gas is displaced into the vent header and compressed into one of four 470-ft^3 gas decay tanks. Gas in the tanks can be compressed to 100 psig.

When the liquid in the chemical and volume control tank is processed, the gas is returned from the decay tank or from a nitrogen supply backup if decay tank gas is not available. In practice, most of the stored gas is reused. When sufficient excess gas has accumulated, the decay tank is isolated and held for further decay. It is then sampled for iodine and radiogas and released through a charcoal filter to the auxiliary building vent system (plant vent). The normal line-up of the decay tanks is one on service/reuse, one on standby, one held for decay, and one in reserve. If the activity level in the plant vent is excessive (as determined by the plant vent monitor), the release is automatically terminated.

12.3.3.2 Plant Ventilation System

The plant ventilation system supplies outside air to the intermediate building hot side and auxiliary building and exhausts the air to the plant vent stack. All air exhausted from the auxiliary building is processed by high efficiency particulate air and charcoal filters prior to release to the plant vent. The normal exhaust flow rate in the plant vent is about 75,000 cfm. The flow rate is used in calculating offsite dose and is checked regularly to verify that offsite dose calculations use a valid flow rate.

The plant vent is continuously sampled for air particulate, radiogas, and Iodine-131 by on-line monitors. There is also a grab sample iodine monitor, which contains a particulate filter and a charcoal filter (fixed filters). Release determinations are made from laboratory analysis of the filters from the grab sample iodine monitor. The lab measurements serve as a calibration check for the on-line readings of the particulate monitor (moving filter) and iodine monitor. Releases are determined directly for radiogas from the radiogas monitor. Gas monitor calibration is checked by laboratory analysis of a weekly grab sample. Lab analysis and the radiation monitoring system are used, as appropriate, for release measurements for all releases. Several continuous air monitors are provided for the auxiliary building. One monitor is located on each of the three levels and has channels that continuously monitor for iodine, particulate, or radiogas activity or a combination of these activities.

12.3.3.3 Containment Ventilation System

The containment building atmosphere is continuously cooled and cleaned by a recirculating air system located inside the building (containment air cooling system). The system consists of four separate air handling units. Each unit has separate inlet dampers and is equipped with a water cooled heat exchanger, a demister, a bank of high efficiency particulate air filters, and a fan. The cleaned and cooled air is recycled within containment via a common exhaust plenum and distribution system.

An auxiliary filtering system is also provided in the containment building. It consists of two charcoal filter units with a combined air handling capacity of 10,200 cfm. The containment building atmosphere may be recirculated through these units to reduce iodine concentrations if entry to the containment or purging is planned.

When purging of the containment is required, the containment mini-purge system is utilized (see Section 9.4.1.2.9).

The containment atmosphere is continuously sampled for air particulate, radiogas, and Iodine-131 by on-line monitors. There is also a grab sample iodine monitor, which contains a particulate filter and a charcoal filter (fixed filters). The filters from the grab sample iodine monitor are removed once a week and are analyzed in the laboratory to determine and verify concentrations in the containment atmosphere. Before any containment entry is made, the containment auxiliary charcoal filter system is placed in operation, and grab samples of the containment atmosphere are used, in conjunction with the radiation monitoring system, to determine the iodine, gaseous, and particulate activity levels. Results are used to determine stay time or to determine if operation of the containment mini-purge system is necessary.

12.3.3.4 Air Ejector and Gland Seal Exhaust System

The air ejector and gland seal exhausts are combined and discharged through a vent on top of the turbine building. A gross radioactivity monitor is provided external to the vent pipe (a sodium iodide scintillation detector in a snowplow-type lead shield). See Section 11.5.2.2.8 for additional information.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

12.3.4.1 Introduction

The function of the radiation monitoring system is to detect any plant problem which may lead to a radiation hazard and/or release of radioactivity to the environment. The system also warns the operators of this hazard so that appropriate actions may be taken. To accomplish this function, the system utilizes both area and process radiation monitors. See table 6-4 of CY-GI-170-300 (Offsite Dose Calculation Manual, revision 36), for a list of area radiation monitors. The process radiation monitors are discussed in Section 11.5.

The area radiation monitors are used to indicate general radiation levels in a specified area of the plant over a range of 10^{-5} or 10^{-4} to 10^{4} R/hr except the containment high-range radiation monitors (R-29 and R-30), which have a range of 1 to 10^{7} R/hr. These ranges meet the instrument range recommendations of Regulatory Guide 1.97. The monitors are set to alarm

(visually and audibly) at the location and in the control room when a significant increase in radiation level occurs. This system is also designed to alarm in the control room if a meter or detector should fail. Various monitors also perform control functions, primarily isolating a flowpath in order to limit the spread of contamination.

In addition, and in accordance with the requirements of NUREG 0737, Item III.D.3.3, mobile air monitors having a single channel analyzer, calibrated to the Iodine-131 energy, are located in various areas throughout the plant to detect the presence of iodine. Portable air samplers with charcoal and silver zeolite cartridges are available in various locations throughout the plant.

12.3.4.2 Description

The area radiation monitor alarm setpoints are calculated by methods described in the Offsite Dose Calculation Manual (ODCM). The alarm setpoints are listed in plant procedures.

Area monitors R-1 and R-3 are in low background radiation areas of less than 1 mrem/hr. Alarm setpoints will give personnel sufficient warning of changing radiation levels while preventing spurious alarms.

Area monitor R-7 alarm setpoint for the in-core detector area is set low enough so as to alarm when flux mapping is in progress but high enough to be off as the in-core detectors are decaying. This is because if one area radiation monitor channel is alarming, the audible alarm on the control board will not sound if another channel goes on alarm.

All of the other channel alarms are set sufficiently above the normal operating radiation levels so as to prevent spurious alarms while at the same time set to provide alarms for any significant increases of radiation levels that might occur.

The letdown monitor alarm channel, R-9, is located on the letdown line in the sodium hydroxide tank room. Its function is to detect a change in the radioactivity of the primary coolant and alert the control room. The alarm setpoint is set at a pre-determined value above the normal background for an early indication of an increase in failed fuel.

R-26 is the area monitor for the all-volatile-treatment mixed bed demineralizers, regeneration tank, and waste tank. Its alarm setpoint will warn personnel if these tanks become radiation areas due to primary to secondary leakage.

Monitors R-29 and R-30 are the containment high range radiation monitors (1 to 10^7 R/hr). The alert alarm is set at 10 R/hr corresponding to the top scale of the low-level area monitors in the containment (R-2 and R-7). A high alarm of 100 R/hr gives notice of a significant release of fission gases to the containment. These monitors have been installed in accordance with the requirements of NUREG 0737, Item II.F.1.2.C.

Area radiation monitors R-31 and R-32 are the A and B steam line radiation monitors. Output from the associated detectors is displayed on control room indicators and the plant process computer system (PPCS). Alarm setpoints are set within the working range of the readout device just above the ambient radiation level to maximize sensitivity and avoid spurious alarms.

12.3.4.3 Radiation Monitoring System Detectors

All of the monitors in the radiation monitoring system use one of four types of detectors: ion chamber, scintillation detectors, Geiger-Mueller tubes, or a semiconductor detector. The detector type is selected as appropriate for each application based on a variety of factors including Process or Area monitor, isotopes of interest and measurement range.

The sensitivity of the detectors is such that they can be used in radiation fields from 0.1 mrem/hr up to 10^4 R/hr where the energy of the gamma radiation can vary approximately between 50 KeV and 3 MeV.

Process monitor detector assemblies contain a check source that can be exposed to the detector by a solenoid-operated shutter. The source is used for verification of system function, but not for calibration.

12.3.5 EQUIPMENT AND SYSTEM DECONTAMINATION

12.3.5.1 Design Basis

Activity outside the core could result from fission products from defective fuel elements, fission products from tramp uranium left on the cladding in small quantities during fabrication, products of n - γ or n - p reactions on the water or impurities in the water, and activated corrosion products. Fission products in the reactor coolant associated with MODES 1 and 2 and tramp uranium are generally removed with the coolant or in subsequent flushing of the system to be decontaminated. The products of water activation are not long-lived and may be removed by natural decay during reactor cooldown and subsequent flushing procedures. Activated corrosion products are the primary source of the remaining activity.

The corrosion products contain radioisotopes from the reactor coolant which have been absorbed on or have diffused into the oxide film. The oxide film, essentially magnetite with oxides of other metals including chromium and nickel, can be removed by chemical means.

Water from the primary coolant system and the spent fuel pool (SFP) is the primary potential source of contamination in addition to the corrosion film of the primary coolant system. The contamination could be spread by various means when access is required. Contact while working on primary system components could result in contamination of the equipment, tools, and clothing of the personnel involved in the maintenance. Also, leakage from the system during operation or spillage during maintenance could contaminate the immediate areas and could contribute to the contamination of the equipment, tools, and clothing.

12.3.5.2 Methods of Decontamination

Surface contaminants which are found on equipment in the primary system and the spent fuel pool (SFP) that are in contact with the water are removed by conventional techniques of flushing and scrubbing as required. Tools are decontaminated by flushing and scrubbing since the contaminates are generally on the surface only of nonporous materials. Personnel and their clothing are decontaminated according to the standard radiation protection requirements.

Those areas of the plant which are susceptible to spillage of radioactive fluids are painted with a sealant to facilitate decontamination that may be required. Generally, washing and flushing of the surfaces are sufficient to remove any radioactivity present.

The corrosion films generally are tightly adhering surface contaminants and must be removed by chemical or mechanical processes. Since decontamination experience with reactors is continually being gained, specific procedures may change for each decontamination case.

Portable components may be cleaned with a combination of chemical and mechanical methods if required.

12.3.5.3 Decontamination Facilities

Decontamination facilities onsite consist of an equipment pit and a cask pit located adjacent to the spent fuel storage pool. In the stainless-steel-lined equipment pit, fuel handling tools

and other tools can be cleaned and decontaminated. Decontamination of small tools and pieces of equipment can be accomplished in the contaminated storage building(CSB) (see Section 12.5.4).

In the cask decontamination pit, the outside surfaces of the shipping casks are decontaminated, if necessary, by using steam, water detergent solutions, and manual scrubbing to the extent required. When the outside of the casks are decontaminated, the casks are removed by the auxiliary building crane and hauled away.

A decontamination shower and washroom for the personnel is located adjacent to the radiation control area locker room. Personnel decontamination kits with instructions for their use are in an area adjacent to the radiation control area locker room.

REFERENCES FOR SECTION 12.3

1. Letter from D. M. Crutchfield, NRC, to R. W. Kober, RG&E, Subject: Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Postaccident Operations (NUREG 0737, Item II.B.2), dated May 23, 1984.

2. NRC IE Bulletin 78-08, Radiation Levels From Fuel Element Transfer Tubes, dated June 12, 1978.

3. Letter from R.C. Mecredy, RG&E, to G.S. Vissing, NRC, Subject: Revised Response to NRC IE Bulletin 78-08, Radiation Levels from Fuel Element Transfer Tubes, dated December 27, 2000.

4. Letter from Constellation Energy to the NRC, subject: Licensing Amendment Request Regarding Extended Power Uprate, July 7, 2005.

Table 12.3-1
PLANT ZONE CLASSIFICATIONS

Zone	Condition of Occupancy	Maximum Dose Rate (1% Failed Fuel) (mrem/hr)
0	Continuous access	0.1
I	Continuous access	1.0
II	Periodic access	2.5
III	Limited access	15
IV	Restricted access	>15

Table 12.3-2a
PRIMARY SHIELD NEUTRON FLUXES AND DESIGN PARAMETERS - CALCULATED NEUTRON FLUXES (HISTORICAL)

Energy Group	*Incident Fluxes (n/cm^2-sec)*	*Leakage Fluxes (n/cm^2-sec)*
E > 1 MeV	2.2×10^9	7.5×10^2
5.3 keV≤ E ≤ 1 MeV	2.3×10^{10}	1.6×10^3
0.625 eV ≤ E ≤ 5.3 keV	1.4×10^{10}	2.7×10^3
E < 0.625 eV	1.9×10^{10}	9.8×10^5

Note: The above table presents the input parameters used to support the primary shield design for the original license application. The calculated fluxes were based on a core configuration that included fresh fuel on the core periphery, providing the greatest contribution to neutron and gamma leakage. With the low leakage fuel management scheme utilized for the uprated core, the existing primary shielding remains conservative.

Table 12.3-2b
PRIMARY SHIELD NEUTRON FLUXES AND DESIGN PARAMETERS - DESIGN PARAMETERS (HISTORICAL)

Core thermal power	1520 MWt
Active core height	144 in.
Effective core diameter	98.16 in.
Baffle wall thickness	1.125 in.
Barrel wall thickness	1.75 in.
Thermal shield wall thickness	3.50 in.
Reactor vessel I.D.	132.0 in.
Reactor vessel wall thickness	6.50 in.
Reactor coolant cold-leg temperature	552°F
Reactor coolant hot-leg temperature	607°F
Maximum thermal neutron flux exiting primary concrete	10^6 n/cm^2-sec
Reactor shutdown dose exiting primary concrete	<15 mrem/hr

Note: The above table presents the input parameters used to support the primary shield design for the original license application. The conservative analytical techniques typically used to establish shielding requirements ensure that core power uprate to 1775 MWt and operation with an 18-month fuel cycle will have no significant impact on the required thickness of the primary shielding.

Table 12.3-3
SECONDARY SHIELD DESIGN PARAMETERS (HISTORICAL)

Core power density	85 W/cm^3
Reactor coolant liquid volume (hot)	6245 ft^3
Reactor coolant transit times	
Core	0.850 sec
Core exit to steam-generator inlet	2.013 sec
Steam-generator inlet channel	0.596 sec
Steam-generator tubes	3.234 sec
Steam-generator tubes to vessel inlet	2.570 sec
Vessel inlet to core	1.774 sec
Total out of core	10.187 sec
Full-power dose rate outside secondary shield	< 1 mrem/hr

Note: The above table presents the input parameters used to support the secondary shield design for the original license application. The conservative analytical techniques typically used to establish shielding requirements ensure that core power uprate to 1775 MWt and operation with an 18-month fuel cycle will have no significant impact on the required thickness of the secondary shielding.

Table 12.3-4
CONTAINMENT STRUCTURE DESIGN PARAMETERS (HISTORICAL)

Core thermal power	1520 MWt
Minimum full power operating time	1000 days
Equivalent fraction of core melting	1.0
Fission product fractional releases	
Noble gases	1.0
Halogens	0.5
Remaining fission product inventory	0.01
Cleanup rate following accident	0
Maximum integrated dose (infinite exposure) in the control room	< 2 rem

Note: The above table presents the input parameters used to support post-accident containment shield design for the original license application. The impact of core power uprate to 1775 MWt and operation with an 18 month fuel cycle has been evaluated and the maximum operator dose for vital access missions remain within the regulatory limit of 5 Rem.

Table 12.3-5
REFUELING CANAL AND SPENT FUEL POOL DESIGN PARAMETERS (HISTORICAL)

Total number of fuel assemblies	121
Minimum full power exposure	1000 days
Minimum time between shutdown and fuel handling	100 hr
Maximum dose rate (due to spent fuel) adjacent to spent fuel pool (east side of transfer canal)	1.0 mrem/hr
Maximum dose rate (due to spent fuel) at water surface	2.5 mrem/hr
Maximum dose rate adjacent to the north wall (intermediate building south)[a]	15 mrem/hr
Maximum dose rate adjacent to the west wall[a]	15 mrem/hr

Note: The above table presents the input parameters used to support the spent fuel pool and refueling canal shield design for the original license application. The conservative analytical techniques typically used to establish shielding requirements ensure that core power uprate to 1775 MWt and operation with an 18-month fuel cycle will have no significant impact on the shielding requirements associated with fuel handling.

a. Calculated

Table 12.3-6
PRINCIPAL AUXILIARY SHIELDING (HISTORICAL)

Component	Concrete Shield Thickness
Demineralizers	4 ft 0 in.
Charging pumps	2 ft 6 in.
Liquid waste holdup tanks	3 ft 3 in.
Spent regenerant chemical holdup tanks	2 ft 0 in.
Volume control tank	3 ft 6 in.
Reactor coolant filter	3 ft 6 in
Gas stripper	3 ft 6 in.
Gas decay tanks	3 ft 6 in.
Gas compressor	2 ft 6 in.
Waste evaporator (physically removed in 1999)	2 ft 6 in.
Design parameters for the auxiliary shielding include:	
Core thermal power	1520 MWt
Fraction of fuel rods containing small clad defects	0.01
Reactor coolant liquid volume (hot)	6245 ft^3
Letdown flow (normal purification)	40 gpm
Effective cesium purification flow	3 gpm
Cut-in concentration deborating demineralizer	160 ppm
Dose rate outside auxiliary building	1 mrem/hr
Dose rate in the building outside shield walls	2.5 mrem/hr

Note: The above table presents the input parameters used to support Auxiliary Building shield design for the original license application. The conservative analytical techniques typically used to establish shielding requirements, and the plant technical specifications that limit the reactor coolant activity to well below 1% fuel defects ensure that core power uprate to 1775 MWt and operation with an 18 month fuel cycle will have no significant impact on the required thickness of the auxiliary building shielding.

12.4 DOSE ASSESSMENT

12.4.1 OPERATION IN MODES 1 AND 2

Radiation surveys are made of plant areas on a periodic basis by radiation protection personnel. Measurements are recorded on floor plan maps which are prepared for this purpose. The radiation survey information presented in Sections 12.4.1 and 12.4.2 represent survey performed prior to the uprate. Following power uprate, the maximum increase in radiation level is expected to be 19%. Results of radiation surveys taken during 1983 are summarized in Tables 12.4-1 and 12.4-2. Results of radiation surveys taken during 2019 are summarized in Table 12.4-1a.

Inspection of the data in Table 12.4-2 shows that radiation levels in the control room, office building, and turbine building were negligible (e.g., all readings were less than 0.05 mrem/hr). Radiation levels in the intermediate building were also relatively low. The next highest area reading observed was 4 mrem/hr at the entrance to the nuclear sample room. Most of the other areas in the intermediate building were less than 1 mrem/hr. The spent fuel pool (SFP) skimmer filter read several hundred mrem/hr.

Radiation levels in the auxiliary building depend on the nature of the activities in progress and radioactivity levels in process systems and waste handling equipment. Radiation levels in the spent fuel pool (SFP) and refueling area were generally less than 2 mrem/hr. Radiation levels in accessible areas on the three operating floors were generally less than 2 mrem/hr. Several hundred mrem/hr may be present in rooms containing process lines and equipment such as the NaOH tank room, volume control tank room, reactor coolant filter room, nonregenerative heat exchanger room, spent resin tank room, waste holdup room, and drum storage area.

Radiation levels from equipment in the auxiliary building such as the waste holdup tank, contaminated valves and lines in the letdown system, demineralizer units, volume control tank, reactor coolant filters, spent resin units, and storage tanks may approach high values. For example, 10 rem/hr was noted from the reactor coolant filter, 2 rem/hr from the bottom of the waste holdup tank, and 50 rem/hr from the spent resin tanks.

Radiation levels in containment depend on several factors, including the power level, radioactivity levels in the primary coolant (fuel defects), the amount of leakage from the primary system, and the status of the various ventilation systems (e.g., recycle cooling system, auxiliary cleanup system, and purge system). Under full power and secured conditions the radiation levels on the operating floor are generally on the order of 10 to 50 mrem/hr with local areas near equipment or lines approaching 100 to 200 mrem/hr in a few areas. Radiation levels in accessible areas of the intermediate and basement floor areas are also in the same order of magnitude. High levels may be present from contaminated lines, in shielded areas in the basement level, and under the primary coolant loops. High levels from neutron streaming are present around the reactor head and at the bottom of the reactor vessel in sump A.

12.4.2 FUEL HANDLING OPERATIONS

The radiation levels associated with fuel handling are low because the spent fuel is handled remotely using several feet of water for shielding. During transfer of spent fuel from the

reactor to the storage pool, radiation levels are 1 to 2.5 mrem/hr at the manipulator cranes and 1 to 2 mrem/hr at the surface of the water. These dose rates are not changed by movement of spent fuel. The general floor working areas have a radiation level of 1 to 10 mrem/hr.

Radiation levels associated with fuel inspection are essentially the same as for fuel transfer. Inspection can be done by underwater television cameras and leak testing or "sipping" is also done remotely, under water.

During the fuel shipping process, spent fuel would be loaded into a shipping cask remotely, under water. Most of the exposure would be received during the decontamination of the shipping cask after it was removed from the storage pool.

The radiation levels when handling fuel with maximum burnup are not expected to be significantly different from those detailed above. No more than a 10% increase in radiation levels is expected with maximum burnup fuel.

12.4.3 POSTACCIDENT CONDITIONS

12.4.3.1 Summary

Doses to personnel during postaccident access to vital areas were evaluated as part of the plant shielding design review submitted to the NRC in December 1979. (See *Reference 1*). This evaluation identified the most critical areas requiring personnel access after the onset of extreme accident conditions following a postulated major release of radioactivity into the containment building. Consideration was given to areas where predetermined postaccident functions would be performed (nuclear sample room, chemistry laboratory, count room, control room, and air sample penetrations) and to areas where personnel could be called upon to execute certain accident-mitigating or short-term recovery tasks (hydrogen recombiner panel, radwaste panel, and building ventilation filters). Potential radiation exposures were determined for each task and included additional exposure due to accessing the areas considered.

As a result of subsequent plant modifications, the list of areas that need operator access has changed. Several of the access requirements listed in the 1979 Design Review Report (*Reference 1*) are no longer required due to changes in licensing basis since 1979, specifically access requirements associated with post-accident sampling and access requirements to the hydrogen recombiner panel. The access requirements for sampling were predicated upon the perceived need for samples of the containment sump, containment atmosphere, and reactor coolant system within a relatively short period of time after an accident occurred. However, post TMI studies have shown that other means can be employed to determine the degree of core damage and classify events for emergency planning purposes. Consequently, the Post Accident Sampling System (PASS) was removed from the Technical Specifications in Amendment 81 using the Consolidated Line Item Improvement Process (CLIIP) per TSTF-366. WCAP-14986, and the associated NRC SER (*Reference 4*) provided the technical justification. Hydrogen recombiners were removed from the plant Technical Specifications in Amendment 90 per TSTF-447, and the associated NRC SER (*Reference 5*). In general, post TMI information determined that hydrogen production following a design basis Loss-of-Coolant Accident was slow enough that other means could be employed to reduce the concentration to below combustible limits if needed. In the event of a severe accident, the rate of

hydrogen production exceeds the capability of the recombiners, causing the recombiners to become an unwanted ignition source. Therefore, entry into this area is no longer considered necessary for short term post accident operations.

Also, a review of the Emergency Operating Procedures indicated that there are no Emergency Operating Procedure steps that discuss the need to change the auxiliary building, spent fuel pool or control room accident filters. In addition, the Emergency Operating Procedure review established that the vital area access requirements noted in the 1979 Report are not considered "required" steps but "enhancements" to be undertaken only if the environment is considered acceptable by Health Physics (HP) personnel.

Regarding the time when access may be envisioned, the review determined that immediate access was not required for any of the operator actions listed in the 1979 Design Review Report. Based on the above review, and except as noted, it was determined that for purposes of demonstrating availability, the earliest access time that needed to be addressed was at 1 day following the accident. The earliest access time that needed to be addressed for the radwaste panel was 10 days following the accident.

The updated list of operator access requirements includes an additional action identified subsequent to the issuance of the 1979 Design Review Report, i.e., throttling of the service water flow to the component cooling water heat exchangers to support cooling of the residual heat removal system. This action must be completed prior to initiation of the recirculation phase during which sump water is recirculated back to the reactor coolant system following a Loss-of-Coolant accident.

12.4.3.2 Methodology

12.4.3.2.1 Calculation of Dose Rates

Dose rates for the areas of interest were calculated in the December 1979 Design Review Report by determining the potential contributing sources at a representative location and using the appropriate source term from Table 12.2-1 adjusted for decay as required. The dose rate at the representative location was used as the general area dose rate for the area and chosen to envelope all critical location of interest. The SDC code (see *Reference 2*) was used in performing the dose rate calculations.

Reference 3 documents the impact of core power uprate on the post-LOCA gamma radiation dose rates developed in the 1979 Design Review Report by using source term scaling techniques. The updated dose rate and dose estimates reflect the source term assumptions discussed in the 1979 Design Review Report, as updated to reflect a core power level of 1811 MWT (includes a 2% margin to accommodate power measurement uncertainty), and operation with an 18-month fuel cycle.

12.4.3.2.2 Doses to Personnel During Postaccident Access to Vital Areas

Personnel doses received in performing a specific task in a given vital area were calculated as the sum of the doses received during travel to and from the vital area and the dose received while performing the given operation in the vital area.

The doses received during travel were determined by calculating dose rates at selected locations along the travel route (or at a single location if the dose rate along the travel route is relatively uniform) using the methodology discussed in Section 12.4.3.2.1 and multiplying the dose rates by the appropriate travel times. Travel times for access routes initially evaluated in the 1979 Design Review Report were based upon an assumed travel rate of 200 ft per minute.

The travel time associated with the additional action identified subsequent to the issuance of the 1979 Design Review Report, i.e., throttling of the service water flow to the component cooling water heat exchangers, is based on a time and motion study.

As part of the power uprate, Ginna has also updated the planned access route to the Target Areas from that described in the 1979 Design Report. The new access route follows a path from the north side of the control room to the south entrance of the auxiliary building at El 271 ft utilizing a path east of the containment outside any structures or buildings. Access between the floors of the auxiliary building will be via the east stairway. This path was selected since it was determined, based on review of the potential radiation sources, that it was more prudent, based on the concepts of ALARA, to minimize the time spent for access purposes within the auxiliary building and maximize the time spent for access purposes outside the structure.

Doses received while performing a given operation were determined by multiplying the dose rate for the given area by the time to perform the operation. Dose rates for the given vital area were calculated using the methodology discussed in Section 12.4.3.2.1. Time estimates for performing indicated tasks for access routes initially evaluated in the 1979 Design Review Report were based upon experience gained during a decade of plant operation at Ginna Station, whereas the time estimate for the additional action identified subsequent to the 1979 report is based on a time and motion study.

12.4.3.3 Areas That May Require Access for Postaccident Operations

12.4.3.3.1 Hydrogen Recombiner Control Panel (Area A)

Short-term post-accident access to the Hydrogen Recombiner Control Panel located on Elevation 253-ft of the Intermediate Building is no longer required. Long-term operation of this system (if available) would employ dose reduction efforts per station ALARA practices.

12.4.3.3.2 Postaccident Containment Air Sample Penetration No. 203 (Area B)

Short-term post-accident access to Containment Air Sample Penetration No. 203 located at Elevation 271-ft of the Intermediate Building is no longer required. Long-term sampling activities would encounter greatly reduced dose rates, and would employ dose reduction efforts per station ALARA practices.

12.4.3.3.3 Nuclear Sample Room (Area C)

Short-term post-accident access to the Nuclear Sample Room located on Elevation 271-ft of the Intermediate Building is no longer required. Long-term sampling activities would encounter greatly reduced dose rates, and would employ dose reduction efforts per station ALARA practices.

12.4.3.3.4 Primary Chemistry Laboratory (Area D)

Short-term post-accident access to the Primary Chemistry Laboratory located in the Service Building is no longer required. Long-term sampling activities would encounter greatly reduced dose rates, and would employ dose reduction efforts per station ALARA practices.

12.4.3.3.5 Count Room (Area E)

Short-term post-accident access to the Count Room (adjacent to the Nuclear Sample Room) is no longer required. Long-term sampling activities would encounter greatly reduced dose rates, and would employ dose reduction efforts per station ALARA practices.

12.4.3.3.6 Postaccident Containment Sample Penetration No. 305 (Area F)

Short-term post-accident access to Containment Sample Penetration No. 305 located at Elevation 253-ft of the Intermediate Building is no longer required. Long-term sampling activities would encounter greatly reduced dose rates, and would employ dose reduction efforts per station ALARA practices.

12.4.3.3.7 Radwaste Control Panel (Area G)

Area G, the radwaste control panel, is located on the basement floor of the auxiliary building. Immediate postaccident access to the waste disposal panel and gas analyzer is not required; however, certain longer term actions may make access necessary to this area. The waste disposal panel contains pressure gauges for the tanks using cover gas and also for the gas decay tanks and vent header. Alarms for tank and vent header pressure and gas analyzer oxygen are locally indicated with a waste disposal panel alarm given on the main control board. All gas system manual operations and releases are controlled locally at the waste panel. In addition, various equipment associated with the liquid waste system are manually controlled at this location.

During recirculation, residual heat removal piping, safety injection pumps and pipes, and containment spray pumps and pipes will make radiation levels on the basement floor of the auxiliary building very high.

It is anticipated that when occupancy is required at the radwaste control panel, an individual would spend 2 minutes in this area, once per shift. From Table 12.4-4, the radiation levels at this location are estimated to be 3900 R/hr 1 hour after the accident. A 2-minute occupancy in this area would result in a 130 rem dose. After 1 day, 3 days, and 10 days, the dose to a worker in 2 minutes would be 14 rems, 7 rems, and 2.6 rems, respectively.

Access to the radwaste control panel is estimated based on use of route 5 (Table 12.4-3). The dose received along this route is estimated to be 0.45 rem during the first hour of the accident and less than 42 mrem 1 day following the accident.

12.4.3.3.8 Safeguards Bus 16 (Area H)

Area H is on the intermediate level of the auxiliary building, around safeguards bus 16. It is not expected that any immediate access would be required to the areas of the safeguards buses or their associated motor control centers.

Radiation levels in area H will be from the safety injection pumps and associated piping directly below on the basement floor. The shine through the 18-in. concrete floor has been estimated to be about 78 R/hr at 1 hour after the accident. This radiation level drops to 3.3 R/hr at 1 day and 0.52 R/hr at 10 days (see Table 12.4-4).

In the event the chemical and volume control system is used to strip gas from the primary coolant, the waste gas piping from the volume control tank to the gas compressor and into the waste gas decay tanks will be contaminated. The waste gas decay tanks, once filled with accident source activity, will also contribute to the radiation level in area H, although to a lesser extent.

Radiation levels 30 ft from the 1-in. waste gas piping have been estimated to be 130 R/hr at 1 hour after the accident. After 1 day, these levels decrease to 7 R/hr, approximately double the radiation level coming from the floor below. The waste gas decay tanks have been estimated to contribute 3.6 R/hr to area H at 1 hour into the accident and 0.12 R/hr after 1 day.

The waste gas piping is 1 in. in diameter and could be locally shielded. One in. of lead shielding would decrease the radiation level by a factor of about 4, and 2 in. of lead by a factor of 10.

Access to area H at 1 hour after an accident is assumed to be via route discussed in UFSAR Section 12.4.3.2.2 and will result in an additional 5.7rem for ingressing and egressing the area. The majority of this dose comes from the waste gas piping (5.2 rem). If no waste gas stripping is in progress, then the access dose would be only 0.52 rem. If the waste gas piping is contaminated with stripped gases, shielding with the equivalent of 1 in. of lead on waste gas piping and 1/2 in. of lead on residual heat removal sample lines results in a dose to individuals accessing area H of just over 1.94 rem. If access to this area is required at 1 day postaccident, then the radiation dose for ingress and egress would be 0.19 rem and 0.01 rem for unshielded and shielded waste gas and residual heat removal sample piping, respectively.

12.4.3.3.9 **Safeguards Bus 14 (Area I)**

The safeguards bus 14 is on the 271-ft elevation. It has been designated as area I and is directly over area H. Occupancy in this area will be the same as explained for area H.

The radiation level in area I will be essentially from the direct dose from containment. At 1 hour, the radiation level has been estimated to be about 6.5 R/hr; after 8 hours, 0.9 R/hr; and only 0.06 R/hr after 1 day.

Access to area I is via route discussed in UFSAR Section 12.4.3.2.2 and will add 0.45 rem to the worker's dose at 1 hour following an accident and only 0.007 rem after 1 day.

12.4.3.3.10 **Postaccident Containment Air Sample Penetration (Area J)**

Short-term post-accident access to the Containment Air Sample Penetration is no longer required. Long-term sampling activities would encounter greatly reduced dose rates, and would employ dose reduction efforts per station ALARA practices.

12.4.3.3.11 Auxiliary Building Heating, Ventilation, and Air Conditioning (Area K)

Short-term post-accident access to the Auxiliary Building Heating, Ventilation and Air Conditioning Areas to change filters is no longer required. Long-term filter changes would encounter greatly reduced dose rates, and employ dose reduction efforts per station ALARA practices.

12.4.3.3.12 Spent Fuel Pool (SFP) and Auxiliary Building Heating, Ventilation, and Air Conditioning Filters (Area L and Area M)

Short-term post-accident access to the Spent Fuel Pool and Auxiliary Building Heating, Ventilation and Air Conditioning Areas to change filters is no longer required. Long-term filter changes would encounter greatly reduced dose rates, and employ dose reduction efforts per station ALARA practices.

12.4.3.3.13 Control Access High Efficiency Particulate Air and Charcoal Filters (Area N)

Short-term post-accident access to the Control Access High Efficiency Particulate Air and Charcoal Filters is no longer required. Long-term filter changes would encounter greatly reduced dose rates, and employ dose reduction efforts per station ALARA practices.

12.4.3.3.14 Control Room

An additional operator action was identified subsequent to the issuance of the 1979 Design Review Report. The task required throttling of the service water flow to the component cooling water heat exchangers to support cooling of the residual heat removal system. This action must be completed prior to initiation of the recirculation phase during which sump water is recirculated back to the reactor coolant system following a Loss-of-Coolant accident.

The earliest access time is expected to be at 10 minutes after the accident. The only radiation source at that time is the airborne source inside containment. The source term assumptions for the containment airborne source are consistent with that used in the 1979 Design Review Report, i.e., 100% of the core noble gases and 25% halogens. No credit is taken for any removal mechanisms other than decay. The access route dose estimates are based on use of the new access route discussed earlier in Section 12.4.3.2.2.

12.4.3.3.15 Control Room

Continuous occupancy is required in the control room after an accident, and direct radiation exposure from shine must be considered after a design basis accident. Per *Reference 7*, the estimated integrated dose was calculated (*Reference 8*) for shine from containment, the (CRE-ATS) filters and the plume passing the control room. The total integrated dose from these three sources is estimated to be 0.36 Rem for the duration of the accident. This dose must be added to the total airborne dose (see Table 6.4-1).

REFERENCES FOR SECTION 12.4

1. Letter from L. D. White, Jr., RG&E, to D. Ziemann, NRC, Subject: Three Mile Island Lessons Learned Short-Term Requirements, dated December 28, 1979.

2. E. D. Arnold and B. F. Maskervitz, SDC - A Shield Design Calculation Code for Fuel Handling Facilities, ORNL 3041, March 1966.

3. Letter from Constellation Energy to the NRC, subject: Licensing Amendment Request Regarding Extended Power Uprate, dated July 7, 2005.

4. Letter from R. Clark, NRC, to R. Mecredy, Ginna Station, Subject: R.E. Ginna Nuclear Power Plant - Amendment Re: Elimination of Post Accident Sampling System (TAC No. MB3387), dated January 17, 2002.

5. Letter from D. Skay, NRC, to M.G. Korsnick, Ginna LLC, Subject: R.E. Ginna Nuclear Power Plant - Amendment Eliminating Requirements for Hydrogen Recombiner and Hydrogen Monitors Using the Consolidated Line Item Improvement Process (TAC No. MC4195).

6. Letter from P. Milano, NRC, to M.G. Korsnick, Ginna LLC, Subject: R.E. Ginna Nuclear Power Plant - Amendment re: 16.8% Power Uprate (TAC No. MC7382).

7. Regulatory Guide 1.183, Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors.

8. DA-NS-2006-031, LBLOCA Direct Radiation Doses in the Control Room.

Table 12.4-1
RADIATION MONITORING SYSTEM READINGS (1983)

Monitor Number	Monitor Name or Location	1520 MWt (mrem/hr)
R1	Control room area	< 0.1
R2	Containment area	10
R3	Radio chem lab	0.1
R4	Charging pump room	5
R5	Spent fuel pool area	1
R6	Nuclear sample room	2
R7	Incore detector area	10
R8	Drumming station	3
R9	Letdown line monitor	40

Table 12.4-1a
RADIATION MONITORING SYSTEM READINGS (2019)

Monitor Number	Monitor Name or Location	1775 MWt (mrem/hr)
R26	AVT – Condensate DI resin vessel D	
R33	Intermediate Building Hot – Nuclear sample room	0.4
R34	Aux Building – Containment spray pumps	0.2
R35	Intermediate Building Hot – PASS sample panel	0.15
R49	Intermediate Building Cold – Purge Fan Area	0.1

Table 12.4-2

RADIATION SURVEY READINGS IN PLANT AREAS (1983)[a]

Area Description or Location	Readings (mrem/hr)
Control room	<0.05
Quality control office	<0.05
Results and test office	<0.05
Health physics office	<0.05
Cafeteria	<0.05
Main hall	<0.05
Turbine building	
General area	<0.05
Main steam	<0.05
Reheaters	<0.05
Condensers	<0.05
Blowdown	<0.05
Secondary sample sink	<0.05
Intermediate building areas	
Elevation 253 ft	0.05 to 3
Elevation 271 to 278 ft	0.05 to 4
Elevation 295 to 298 ft	0.05 to 4
Auxiliary building operating floor	
Spent fuel pool and decontamination pit area	1
New fuel storage area	1
Refueling water storage tank (RWST)	6
Monitoring and reactor makeup in water tank area	1
Waste handling area	5 to 10
Boric acid storage tank area	2 to 25
Waste condensate demineralizer room	3 to 130[a]
Drumming station	5 to 10
Waste storage vault	250 to 10,000
Auxiliary building intermediate floor	

Area Description or Location	Readings (mrem/hr)
Refueling water storage tank (RWST) area	0.5 to 4
Vent filter area	1
Waste line southeast end	70
Gas compressor room	1 to 60
Gas decay tank room	1
Chemical and volume control system tank rooms	1 to 6
Volume control tank room	1000 to 5000
Concentrates tank room	2 to 200[b]
Demineralizer vault	10^3 to 10^6
Reactor coolant filter room	50 to 20,000[c]
Auxiliary building basement floor	
General operating area	1 to 5
Refueling water storage tank (RWST) room	6 to 25
Valves on lines	15 to 200
Seal injection filter	1000 to 10,000
NaOH tank room	4 to 160[d]
Nonregenerative heat exchanger	15 to 250
Seal return cooler	4 to 35
Charging pump room	2 to 180
Chemical and volume control system tank room	5 to 150
Waste holdup tank	15 to 2000[e]

a. Reading on demineralizer, maximum.
b. Reading on filter, maximum.
c. 20 rem/hr at contact with filter.
d. Reading on letdown line, maximum.
e. Reading on bottom of tank, maximum.

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a. Readings taken in 1983 at 100% full power or 1520 MWt.

Table 12.4-3
Table DELETED

Table 12.4-4
EXPOSURE RATES FOR VITAL AREAS AS A FUNCTION OF TIME (R/hr)

VITAL AREA[a] Area Identification per 1979 Design Review Report	*1 Hour*	*1 Day*	*30 Days*	*6 Months*
Area A (Hydrogen Recombiner)	Access Requirement has been eliminated			
Area B-F (PASS related activities)	Access Requirement has been eliminated			
Area G (Radwaste Panel)	4000>	470	28	6.7
Area H (Safeguards Bus 16) Excluding waste gas system	78	3.3	0.21	NC[b]
Including Waste gas system (unshielded pipes)	232	10	0.37	NC[b]
Including waste gas system (1 in. lead on pipes)	120	3.8	0.19	NC[b]
Area I (Safeguards Bus 14)	7	0.06	Negligible[c]	NC[b]
Area J (PASS related activity)	Access Requirement has been eliminated			
Area K-N (HVAC Filter change out)	Access Requirement has been eliminated			
New Area-Access Requirement identified Subsequent to the 1979 Design Review Report				
Throttle SW to CCW HX	5 rem/hr at T=10 mins			

a. See Section 12.4.3 for Vital Area Description
b. NC: Not calculated
c. (0.001 R/hr)

Table 12.4-5
VITAL AREA RADIATION DOSE SUMMARY

Vital Area[a]/Location	*Time After Accident*	*Occupancy*	*Dose In Area*	*Access Dose to and From*	*Total*
Area A (Hydrogen Recombiner Control Panel)	Access Requirement has been eliminated				
Area B-F (Post Accident Sampling Access Requirements)	Access Requirement has been eliminated				
Area G (Radwaste Control Panel)	10 day	2 min	2.7 rem	Negl	2.7 rem
Area H (Safeguards Bus 16)	1 day	--	3.3 R/hr	Negl[b]	3.3 R/hr
Area I (Safeguards Bus 14)	1 day	--	0.1 R/hr	Negl[b]	0.1 R/hr
Area J (Post Accident Sampling Access Requirement)	Access Requirement has been eliminated				
Area K-N (HVAC Filter Changeout (Aux Bldg., Spent Fuel Pool, Control Room)	Access Requirement has been eliminated				
[c]Throttle SW to CCW HX	10 mins	10 mins<	0.9 rem	0.6 rem[b]	1.5 rem

a. See Section 12.4.3 for Vital Area Description

b. Areas H and I consider Access at 1 day and beyond. Area G is based on Access at least 10 days after the accident.

c. Access requirement identified subsequent to issuance of the 1979 Design Review Report.

12.5 RADIATION PROTECTION PROGRAM ADMINISTRATION

12.5.1 ORGANIZATION

The designated radiation protection manager at Ginna Station is the Manager, Radiation Protection. Personnel report to the Manager, Radiation Protection, for key areas of the radiation protection programs. The authority and responsibilities of this management position is discussed in Section 13.1.2.1.2.2.

The qualifications of individual members of the radiation protection staff meet or exceed the minimum qualification requirements for comparable positions referenced in ANSI/ANS 3.1-2014 as supplemented by Regulatory Guide 1.8, June 2019. The Manager, Radiation Protection, or members of the supervisory staff, meet or exceed the specific requirements for a radiation protection manager.

Training programs for radiation protection personnel are discussed in Section 13.2.1.

12.5.2 EXPOSURE CONTROL PROGRAM

12.5.2.1 External Exposure

The external occupational exposure control program consists of Radiation Work Permits (RWP), dosimetry, dose monitoring and review, dose limitations, and quality assurance.

The control of personnel dose within the RWP is accomplished by the use of personnel dosimeters, radiation surveys, timekeeping and/or stay times, and a records system for documenting dose. The dosimeter used primarily for dose of legal record for those individuals required to be monitored in accordance with 10CFR20.1502 shall be accredited by the National Voluntary Laboratory Program (NVLAP) for ionizing radiation dosimetry. Individuals for whom monitoring is not required in accordance with 10CFR20.1502 may be monitored using Self Reading Dosimeters (SRD) only. If the dosimeter is lost or damaged, a backup dosimeter or other means of dose reconstruction may be used as the dose of record. The system of records used to control personnel external exposure is through individual dose history files and continuously undated database of dose and RWP history. The exposure limitations program consists of procedures establishing policy for annual external dose guideline limits and RWP criteria to meet the ALARA concept.

The quality assurance for external exposure control consists of the calibration of dosimeters and portable survey instruments, and quality control checks. The program is also evaluated for compliance by audits and surveillance. The program is evaluated for performance improvement by self-assessments.

12.5.2.2 Internal Exposure

The internal occupational exposure program consists of using RWPs, dosimetry records, internal exposure review, internal exposure limitations, and quality assurance of whole body analysis equipment. The control of personnel internal exposure within the RWP is accomplished by the use of airborne radioactive air sampling to identify and control airborne radioactivity, time keeping and/or stay times, and a records system for documenting any internal dose. The method of monitoring internal exposure is the radiological analysis of air

samples to calculate derived air concentration (DAC) hours. Internal dose of record may be assigned from DAC hours or from in-vivo or in-vitro radiological analyses. The system of records used to control personnel internal exposure is through individual dose history files, computer database of DAC hour tracking and RWP history, whole body count results, air sample results, and air contamination surveys. The exposure limitations consist of procedures establishing policy for DAC limits for posting as an airborne area, RWP criteria to meet the ALARA concept, and DAC hour limits for requiring whole body counting, internal dose assignments or in-vitro analysis.

The quality assurance of internal exposure consists of the calibration of portable survey instruments, air sampling equipment, counting instruments, whole body counters, and the use of quality control checks for calibrated equipment. The program is evaluated for compliance by audits and surveillances, and is evaluated for performance improvement by self-assessments.

12.5.2.3 Respiratory Protection

A respiratory protection program is in effect in accordance with 29 CFR 1910.134, Regulatory Guide 8.15 and NUREG 0041, under the supervision of a Health Physicist. This program is implemented by written procedures on the use, selection, fitting and testing, maintenance, and operation of respirators.

12.5.2.4 Radioactive Sources Control

All radioactive sources are under the control of the Manager, Radiation Protection and/or a designated Health Physicist. Handling of all sources is by trained personnel following established plant procedures or Radiation Work Permits. These procedures or permits specify precautions, protective clothing, dosimetry, and permissible locations for use, where applicable. Procedures include requirements for receipt of radioactive materials, use of sources, and inventory of sources. Sources that contain quantities of byproduct material listed in 10 CFR 30.70, Schedule A or 10 CFR 30.71, Schedule B, and all other sources are leak tested in accordance with plant procedures.

12.5.2.5 Medical Examinations

All prospective employees must pass a physical examination given by the Medical Department. This includes medical history, radiation history, physical, electrocardiogram (if considered necessary), special eye examination, and lab analyses. Special health reexamination or bioassay tests are required as determined by the Medical Department.

Employees required to wear respirators must pass a medical check. This includes the pulmonary function testing process and completion of a medical questionnaire. Reexaminations are required annually.

12.5.3 SURVEILLANCE PROGRAM

12.5.3.1 Surveys

Radiation and contamination surveys are performed in accordance with written procedures. The procedures include the methodology for conducting the survey as well as minimum survey frequencies for all areas of the station.

12.5.3.2 Radiation Work Permits

Radiation Work Permits are used for entry into all restricted areas of Ginna Station such as radiological surveys by radiation protection personnel, tours by operations personnel, or inspections by station personnel and for specific tasks or jobs such as maintenance work on reactor coolant systems, special plant evolutions (resin transfers, etc.), or plant modifications. The duration of the approval period and the approval process for Radiation Work Permits are specified in radiation protection procedures.

12.5.3.3 Access Control, Posting, and Labeling

12.5.3.3.1 Restricted Areas

The plant site is divided into categories, the unrestricted area, the restricted area, and controlled areas. A controlled area may be established outside the restricted area but inside the site boundary for any reason.

The restricted area encompasses all plant areas access to which is controlled to limit personnel exposure to radioactive materials and radiation. Access to the restricted area is limited to those persons authorized for entry by station supervisors and radiation protection personnel. Entry to and exit from the restricted areas in the intermediate building, auxiliary building, and containment are through the designated access control point only.

Any area inside the restricted area in which radioactive materials and radiation are present is surveyed, classified, and conspicuously posted with the appropriate radiation caution sign.

The general area of the service facilities is designed to provide adequate personnel decontamination and change areas. The Radiation Protection Access Control Point is an area designed to provide personnel monitoring, tools and materials monitoring, worker point of contact for entry, use of Radiation Work Permits for controlling worker exposure, and personnel decontamination. The Radiation Protection group controls access and egress for workers into restricted areas to ensure radioactive materials are controlled in the restricted area or otherwise properly controlled in accordance with plant procedures.

12.5.3.3.2 Access Control

Dosimeter readings are recorded when personnel enter or exit the restricted area on the Radiation Work Permit used for entry, except for an emergency entry when dose may be recorded after exiting the restricted area. Locations within the restricted area that exceed 1000 disintegrations per minute/100 cm^2 of removable contamination are considered

contaminated areas and are posted with step-off pads and/or signs. The limit for removable contamination outside the restricted area is 1000 disintegrations per minute/100 cm^2.

Administrative and physical security measures are employed to prevent unauthorized entry of personnel to any high-radiation area or very-high-radiation areas (greater than 500 rads in 1 hour at 1 meter). These measures include the following:

A. Areas in which radiation levels are so high that individuals might receive doses in excess of 100 mrem in 1 hour are barricaded and conspicuously posted as high-radiation areas. Administrative controls require the issuance of a Radiation Work Permit prior to entering any high-radiation area.
B. Entrances to locations where the above value exceeds 1 rem in 1 hour are conspicuously posted and, in addition, doors or barriers are locked to prevent unauthorized entry. Keys to these doors or barriers are kept under special administrative control. In addition, very high radiation areas shall include additional administrative controls to prevent unauthorized access.
C. Any individual or group of individuals entering a high-radiation area is provided with either a radiation monitoring device which continuously indicates the radiation dose rate in the area or a radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received for areas with established dose rates or is accompanied by an individual qualified in radiation protection procedures with a dose rate monitoring device.

12.5.3.3.3 **Protective Apparel**

Personnel entering a contaminated area are required to wear protective clothing. The nature of the work to be done is the governing factor in the selection of protective clothing to be worn by individuals. The protective apparel available are shoe covers, head covers, gloves, and coveralls or lab coats. Additional items of specialized apparel such as plastic or rubber suits, face shields, and respirators are available for operations involving high-level contamination. In all cases, radiation protection personnel shall evaluate the radiological conditions and specify the required items of protective clothing to be worn on the appropriate Radiation Work Permit.

Respiratory protective devices may be required in a situation arising from plant operations in which an airborne radioactive area exists or is expected. The use of engineering controls should be the primary means of controlling airborne radioactive areas. In all cases, the airborne concentrations are monitored by radiation protection personnel and the necessary controls or protective devices are specified on the Radiation Work Permit according to concentration and type of airborne contaminants present.

12.5.4 RADIATION PROTECTION FACILITIES AND EQUIPMENT

Locker rooms are used to store items of personal clothing not required or allowed in the restricted area. Dressing/undressing areas, decontamination sink, showers, and PCMs (personnel contamination monitors) are provided for men and women in an area adjacent to the intermediate building, hot side.

The work areas are located adjacent to the restricted area access and are interconnected. The areas consist of an office area, chemistry, count room, radiochemistry laboratory, and chemistry laboratory. The first aid room, general work area, storage area, dosimetry/records area, staff offices and manager's offices are in the administration building in close proximity to the work areas.

The count room contains a gamma spectroscopy system with germanium detectors. The primary sample stations are located in the restricted area on the other side of the radiochemistry laboratory wall. A pass box with an alarm system makes it possible for samples to be transferred to the radiochemistry laboratory with minimal personnel handling times.

Radiation protection facilities include calibration and source checking facilities. Calibration sources include a self-contained, interlocked cabinet type gamma calibrator, gamma calibrators and several smaller sources for free air irradiation at lower dose rates.

A hot shop was historically used for storage and decontamination of contaminated tools and equipment. The hot shop at one time contained a decontamination sink, ultrasonic cleaner, and automatic dishwasher. The contaminated storage building (CSB) is currently used for storage and decontamination of tools, materials and equipment. The CSB is connected to the auxiliary building (AB) by an enclosed walkway and fire rated doors. There is also a fire rated door in the CSB that provides access to the adjacent canister preparation building (CPB). The CSB has a floor area of approximately 3600 ft^2.

A radioactive material storage building (RMSB) is currently used for storage of tools, materials and equipment. The building is located on the west side of the plant south west of the engineering building. The RMSB has a floor area of approximately 2400 ft^2. (See Figure 1.2-1)

A Radioactive Material Storage Facility is located west of the Plant, outside of the Protected Area fence. This facility is used for storage of Radioactive Material. The facility is controlled to prevent unauthorized access and unauthorized removal of licensed radioactive material.

The spent fuel cask decontamination pit in the auxiliary building is used for large equipment decontamination (see Section 12.3.5).

Portable radiation survey instruments meeting current industry standards are available for routine monitoring functions. Ginna's instruments use various detectors such as Geiger-Mueller, ion chamber, scintillation, proportional counter, and newer technology to measure radiation. They are capable of measuring alpha, beta, gamma, neutron, and X-ray radiations from background to very high levels of radiation in the thousands of R/hr range.

Table 12.5-1
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