Specialty Materials
Honeywell
P.O. Box 430
2768 North US 45 Road
Metropolis, IL 62960

October 5, 2011

Certified Mail: 7011 0470 0000 6688 5498

ATTN: Document Control Desk
Director, Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUB-526, Docket # 40-3392

References:

1. Honeywell International Inc. Materials License, SUB-526.
2. Honeywell International Inc. Request to Use ICRP 68 for determination of DAC, ALI, and Soluble Uranium, dated July 26, 2011.
3. International Commission on Radiation Protection (ICRP) Publication 68, Dose Coefficients for Intake of Radionuclides by Worker, Annals of the ICRP, Volume 24, No. 4,1994.
4. International Commission on Radiation Protection (ICRP) Publication 60,1990 Recommendations of the International Commission on Radiological Protection, Annals of the ICRP, Volume 21, No. 1-3,1991.

Subject: HONEYWELL METROPOLIS WORKS SUPPLEMENTAL DOCUMENTATION FOR REQUEST TO USE ICRP 68 FOR DETERMINATION OF DAC, ALI, AND SOLUBLE URANIUM LIMIT

Honeywell Metropolis Works submits this supplemental documentation to follow up on its request to use ICRP Publication 68 based methodologies for determination of DAC, ALI, and Soluble Uranium intake limit, dated July 26, 2011. In addition to the previous amendment application, Honeywell proposes to grant exemptions to 10 CFR 20, Appendix B and the Organ Dose Weighting Factors in 10 CFR 20.1003 by amending Honeywell USNRC Materials License SUB-526 to include the following new License Conditions (LC):

LC-XX

Notwithstanding the Derived Air Concentration (DAC) and Annual Limit on Intake (ALI) listed in Appendix B to 10 CFR Part 20, the licensee may use adjusted DAC values and adjusted ALI values based on International Commission on Radiation Protection (ICRP) Publication 68 (Annals of the ICRP, Volume 24, No.4).

LC-(XX+1)

Notwithstanding the Organ Dose Weighting Factors listed in 10 CFR 20.1003, the licensee may use the Tissue Weighting Factors listed In ICRP Publication 60

NMSS01

(Annals of the ICRP, Volume 21, No. 1-3) for effective dose assessments based on ICRP Publication 68 based methodologies.

It's important to note that Honeywell intends to withdraw a portion of its initial request dated July 26, 2011, specifically "... to limit the weekly intake of soluble uranium to 30 mg, in place of the limit defined in Title 10, Code of Federal Regulations Part 20 Subpart C Section 1201(e)." Contrary to our earlier assumption, this modification appears unnecessary for Honeywell's transition to ICRP Publication 68 based methodologies.

Also, please disregard the proposed revisions of Honeywell's licensing basis documents attached to the initial request dated July 26, 2011; changes to all affected documents will be performed administratively following approval of this amendment.

Attached to this letter is the Technical Basis and Justification for Change for DAC and ALI Modification, including the following supporting document:

Annual Limits on Intake and Derived Air Concentrations Based on ICRP-68 Dose Coefficients (prepared by ENERCON).

As stated in the initial request, the proposed use of ICRP Publication 68 based methodologies has been reviewed and found acceptable by Honeywell Metropolis Works management and ALARA Committee. Proper implementation and use of the newer models and methods will be ensured through systematic management and ALARA Committee oversight.

Please be reminded that the most effective and seamless implementation of the newer models and methods will be possible if this amendment request would be reviewed and approved before the start of the annual monitoring period on January 1, 2012.

If you have any questions on the submitted information or other issues associated with the Honeywell Metropolis Works license amendment request, please contact Mr. Michael Greeno, Regulatory Affairs Manager, at (618) 309-5005.

Sincerely,

Larry A. Smith
Plant Manager

Enclosure

Cc: Kevin Mattern
NMSS Project Manager
Mail Stop EBB 2-C40M
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

M. Wolf
M. Greeno
B. Stokes

ATTACHMENT

Technical Basis and Justification for Change
DAC and ALI Modification

Annual Limits on Intake and Derived Air Concentrations Based on ICRP-68 Dose Coefficients (prepared by ENERCON),

Technical Basis and Justification for Change
DAC and ALI Modification

Introduction

Honeywell Metropolis Works (MTW) is applying for the amendment to USNRC Materials License SUB-526 to authorize the use of methodologies based on International Commission on Radiation Protection (ICRP) Publication 68. ICRP Publication 68 based dose coefficients would be used to assign the effective dose to workers. The use of these advanced methodologies requires adoption of adjusted Derived Air Concentration (DAC) and Annual Limit on Intake (ALI) values in place of those specified in Title 10, Code of Federal Regulations (CFR) Part 20, Appendix B. Accordingly, implementation of adjusted DAC and ALI values will exempt Honeywell from all sections of 10 CFR Parts 20 and 40 that refer to DAC and ALI quantities in Appendix B to Part 20. Adjusted ALI and DAC values may be used when considering notifications of incidents made according to 10 CFR 20.2202(a)(2), 10 CFR 20.2202(b)(2), reporting requirements listed in 10 CFR 40.60(b)(1)(ii), as well as other affected actions.

Acceptance of the newer models and methods of the effective dose assessments assumes the use of the values of the Tissue Weighting Factors versus the Organ Dose Weighting Factors. Therefore, Honeywell requests exemption to the Organ Dose Weighting Factors listed in 10 CFR 20.1003, and may use the values for the Tissue Weighting Factors stated in Table S-2 of ICRP Publication 60 (Table 1).

Honeywell proposes that these exemptions to 10 CFR 20, Appendix B and the Organ Dose Weighting Factors values listed in 10 CFR 20.1003 needed for its transition to the newer models, be documented as the new license conditions, LC- XX and LC-(XX+1).

This attachment contains the following information in support of the amendment application:

- Bioassay Program description demonstrating maturity and strength of the program;
- Bioassay Program Development aspects indicative of the program's vitality;
- Internal Exposure Statistics presenting ALARA and Bioassay Program's positive dynamics;
- *Annual Limits on Intake and Derived Air Concentrations Based on ICRP-68 Dose Coefficients* document prepared by an external consultant, ENERCON, to show the DAC and ALI calculation methods and techniques;

Changes in the Internal Dose Assessment Principles

There are certain changes in the ICRP Publication 68 based methodologies relative to those based on ICRP Publication 30, and hence from the basis of the current Part 20.

Many of the quantities used in radiation protection were renamed, although most retained their original definitions with only subtle changes. For example:

- Dose equivalent was renamed the equivalent dose;
- Effective dose equivalent was renamed the effective dose;
- Quality Factor used to convert dose to dose equivalent was renamed the Radiation Weighting Factor;
- Organ Weighting Factor was renamed the Tissue Weighting Factor;

and similar other name changes were made.

The Tissue Weighting Factors were changed both in magnitude for individual organs, and also in the number of organs specifically assigned Weighting Factors. In 10 CFR 10.1003, six individual organs and tissues are given specific Weighting Factors, and five remainder organs share a Weighting Factor. In ICRP 60 (Table 1), 12 individual organs and tissues are given specific Weighting Factors, and the remainder organs include 10 named organs. The Weighting Factors for some organs and tissues changed significantly.

The default size for inhaled radioactive particulates for occupational exposures was changed to 5 μm. It remained unchanged at 1 μm for members of the general public.

Another variation concerns the ICRP Publication 68 based absorption "types" definitions. ICRP Publication 68 defines the following absorption "types" for radioisotopes: F, M, and S. These absorption "types" only loosely correlate with the inhalation classes D, W, and Y used in ICRP Publication 30 based models. Thus:

> Type F materials are readily absorbed into the bloodstream from the respiratory tract, "Fast" rate of absorption;
>
> Type M materials have intermediate rates of absorption into the bloodstream from the respiratory tract, "Moderate" rate of absorption;
>
> Type S materials are relatively insoluble in the respiratory tract, "Slow" rate of absorption.

The approximate half-times for these absorption rates correspond to:

> Type F (Fast): 10 min (100%).
>
> Type M (Moderate): 10 min (10%); 140 day (90%).
>
> Type S (Slow): 10 min (0.1%); 7000 day (99.9%)

MTW Bioassay Program Description

The need for workers to be in a bioassay program is tied to the probability that the worker will receive intakes in excess of 10% of an ALI in one year (10 CFR 20.1502), which is referred to as the monitoring level. Internal dosimetry monitoring is required for workers who are likely to exceed the annual monitoring level. The majority of workers at MTW are

unlikely to exceed the monitoring level, but because of the nature of their work, many have a reasonable potential for exposure to airborne radioactive materials. These workers may be placed on a routine Bioassay Program even though the monitoring is not required.

MTW Bioassay Program consists of routine and special urinary uranium sampling for evaluation of employee exposure to the natural uranium compounds processed in the plant. The Program utilizes guidance provided in Regulatory Guides 8.9 and 8.11.

Hourly employees are required to provide a routine urine sample twice a month. Salaried personnel who work inside uranium processing areas are sampled monthly. Contractor personnel are sampled based on an evaluation of their work activities. The routine sampling schedule is appropriately adjusted to allow for vacations, illnesses, etc.

Special urinary uranium samples are collected following confined space entries where the air concentrations may exceed DAC, and following a UF_6 release, if employees have been exposed. In addition, employees are encouraged to submit urine samples at the end of a work shift following a suspected exposure to airborne uranium to determine if an exposure has actually occurred. Follow-up special samples are obtained if results exceed the evaluation level.

The Kinetic Phosphorescence Analyzer is used for analysis of urinary uranium.

The administrative action levels have recently been significantly reduced, and the Program currently establishes the following action levels based on uranium in the urine concentrations:

- The evaluation level is a concentration of U^{238} in the urine of 3 micrograms per liter (µg/L), which corresponds to 2% of the Annual Limit on Intake (ALI) as defined in 10 CFR 20. Employees whose urinary excretion rate exceeds the evaluation level are re-sampled for confirmation.
- The investigation level is a concentration of U^{238} in the urine of 15 µg/L, which corresponds to 10% of the ALI. If an employee's urinary excretion rate exceeds the investigation level, the intake is investigated and daily urinary uranium samples are normally obtained until the results are less than the evaluation level.
- The restriction level is a concentration of U^{238} in the urine of 45 µg/L, which corresponds to 30% of ALI and is less than 10 milligrams per week soluble uranium toxicity limit.

If Honeywell's amendment request would be granted, the action levels will be revised accordingly, but will remain consistent with guidance provided in Regulatory Guide 8.9.

Employees exposed to highly soluble UF_6 as a result of unplanned workplace releases are required to submit two special urine samples within the first 24 hours following exposure, usually at 3-6 hours post-exposure and 16-20 hours post-exposure. Work restrictions are imposed if it appears the soluble uranium weekly intake limit may be exceeded. Daily sampling is continued until the concentration is less than the evaluation level.

Exposure to UF_6 is limited by chemical toxicity to the kidney rather than annual radiation dose. The value of 35 µg/L represents the urinary uranium concentration that would be expected in urine at 14 days after an unknown intake of uranium equivalent to the weekly soluble uranium intake limit. It is extremely unlikely an unknown exposure to UF_6 could

occur in the plant due to the highly visible "smoke" produced by a small quantity of UF_6. Special bioassay samples are required following a UF_6 release.

If the chemical compound to which the employee was exposed is known, the corresponding evaluation level may be used; however, if the exposure compound is not known, UF_4 will be conservatively used as the compound of choice in accordance with the ALARA concept.

Intakes are currently calculated using the methodology provided in NUREG/CR-4884. Internal dose evaluation frequency has been recently increased to quarterly. In the event Honeywell's request would be granted, the intakes may be calculated using ICRP Publication 68 based methodologies.

Bioassay Program Development

Honeywell's management has committed significant resources to the Bioassay Program's oversight and development. The following are examples of major recent Program's enhancements:

1. Considerable increase of the number of monitored individuals over a short period of time due to the growing production needs; successful incorporation of new participants with the simultaneous improvements in monitoring quality.

2. Addition of considerable external resources necessary for the Program's management and development.

3. Development and implementation of Honeywell Metropolis Works Internal Dosimetry Technical Basis Manual resulting in the Program's technical basis development and procedural improvements.

4. Modification of internal dose calculations; accounting for uranium daughter products and contaminant contributions and improving the dose assessment completeness and accuracy.

5. Upgrade of bioassay sampling and analytical processes, including:

    a. Placement of a new bioassay facility outside of the restricted area;
    b. Acquisition of modern equipment; and
    c. Addition and training of laboratory personnel in bioassay analytical methods.

    These measures resulted in significant improvement of analysis quality and reduction of potential sample contamination.

6. Continuous administrative measures targeting effective bioassay sample collection and custody.

7. Significant reduction of three administrative action levels (examination, investigation and restriction levels) directed to a better adherence to ALARA principles.

8. Increased (to quarterly) internal dose modeling frequency to facilitate effective response to the elevated internal dose occurrences.

Internal Exposure Statistics

Tables 2, 3 and 4 illustrate MTW Honeywell Bioassay Program's statistics and dynamics in the last five years and allow observance of some apparent trends in the Program's development.

One of the most obvious trends is the Program's growth. Nearly double the number of individuals were monitored in 2010 versus 2006 (Table 3).

This quantitative growth explains the Cumulative Internal Exposure (Total CEDE, Table 2) increase that occurred since 2006. Importantly, the Total CEDE increase in 2010 versus 2006 is only 22% (in spite of the double number of Program's participants) which indicates the general reduction of internal exposures at the plant. The Average CEDE results (Table 2) remaining well below 2.0 rem also confirm this reduction tendency.

Another essential result is that in 2006-2010 there were no internal exposures in excess of 2.0 rem (Table 3). Moreover, the percent of internal exposures exceeding even 0.50 rem remains low; it was below 3% in 2010 for example.

Analysis of 2010 Acute Intakes of Soluble Uranium data (Table 4) demonstrates that approximately 90% of acute intakes were below 1.0 mg of soluble uranium. There were no acute intakes of soluble uranium that exceeded 5 mg.

Conclusion

Considerable resources have been committed by Honeywell's administration to development of improved ALARA practices. These efforts, concentrated on hazard control and protective measures, have resulted in a measurable reduction of internal exposures at the plant. MTW Bioassay Program improvement and enhancement was a critical element of this success.

Honeywell plans to further advance its ALARA practices and Bioassay Program by using the newer models and methods since Part 20 based internal dose assessments could be overly conservative. As stated in SECY-99-077, "... the newer models provide more accurate dose estimates than the models used in Part 20", and "the differences are substantial for...thorium, uranium, and some transuranic radionuclides."

Honeywell believes that the use of ICRP Publication 68 based methodologies would facilitate its ALARA practices and Bioassay Program's progress. The newer models and methods would enable MTW to perform more accurate and realistic internal dose assessments. Since protective measures are based on hazard which is proportional to dose, Honeywell would be able to refocus ALARA practices, particularly internal exposure control and protection, to concentrate on protection based on the actual hazard. These goals may be achieved providing that the requested amendment to the license would be granted.

REFERENCES

1. Title 10, Code of Federal Regulations, Parts 20 and 40.
2. International Commission on Radiation Protection (ICRP) Publication 68, Annals of the ICRP, Volume 24, No. 4,1994.
3. International Commission on Radiation Protection (ICRP) Publication 60, Annals of the ICRP, Volume 21, No. 1-3,1991.
4. Regulatory Guide 8.9, Revision 1, Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program, USNRC, July 1993.
5. Regulatory Guide 8.11, Applications of Bioassay for Uranium, USNRC, June 1974.
6. NUREG/CR-4884, Interpretation of Bioassay Measurements, USNRC, June 1987.
7. SECY-99-077, March 1999.
8. SECY-01-0148, August 2001.

Table 1. Tissue Weighting Factors

Tissue or organ	Tissue weighting factor, w_T
Gonads	0.2
Bone marrow (red)	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Oesophagus	0.05
Thyroid	0.05
Skin	0.01
Bone Surface	0.01
Remainder	0.05

Table 2. Cumulative Internal Exposures

Year	2006	2007	2008	2009	2010
Average CEDE (rem)	0.154	0.108	0.090	0.137	0.101
Total CEDE (rem)	104.545	93.133	61.722	112.161	128.400

Table 3. Internal Exposure Distribution

Year/ Dose Range (rem)	2006	2007	2008	2009	2010
No Measurable Exposure	104	192	209	174	293
< 0.100	151	343	290	311	581
0.100 - 0.249	303	231	115	186	259
0.250 - 0.499	101	82	49	101	99
0.500 - 0.749	16	14	12	27	22
0.750 - 0.999	5	4	5	12	12
1.000 - 1.999	0	0	3	6	4
> 2.000	0	0	0	0	0
Total Number of Individuals Monitored	680	866	683	817	1270

Table 4. 2010 Acute Intakes of Soluble Uranium

Intake Range (mg)	< 1	1 - 1.99	2 - 2.99	3 - 3.99	4 - 4.99	>5	Total Intakes
Number of Intakes	487	35	17	4	3	0	546

ENERCON

Excellence—Every project. Every day.

Annual Limits on Intake and Derived Air Concentrations Based On ICRP-68 Dose Coefficients

Prepared for

Robert C. Stokes
Radiation Protection Program Manager
Honeywell International, Inc.
2768 North US 45 Road
Metropolis, IL 62960

Prepared by

Ross A. Lindberg
ENERCON
4490 Old William Penn Hwy
Murrysville, PA 15668

September 22, 2011

ENERCON

Excellence—Every project. Every day.

Introduction

Current occupational intake limits are defined within Title 10, Code of Federal Regulations Part 20, Appendix B Table 1 [USNRC 1991]. All of these values are derived using guidance and methodology that is consistent with ICRP publication 26 [ICRP 1971] and ICRP publication 30 [ICRP 1979]. In 1999 an NRC memorandum entitled, To Request Commission Approval to Grant Exemptions from Portions of 10 CFR Part 20 [USNRC 1999], was published allowing licensees to request to implement new occupational intake limits based upon dose coefficients provided within ICRP publication 68 [ICRP 1995].

This report describes the methodology used and defines the annual limits on intake (ALI) and derived air concentration (DAC) values for isotopes contained within the uranium, actinium and thorium series based upon ICRP-68 methodology. Effluent concentration limits and sewer disposal limits contained within Title 10, Code of Federal Regulations Part 20, Appendix B Tables 2 and 3 are not addressed within this document.

Discussion

ICRP 68 Annexe B provides effective dose coefficients for ingested and inhaled particulates. Section 7 states;

> *The ALI (Bq) for any radionuclide can thus be obtained by dividing the annual average effective dose limit (0.02 Sv) by the dose coefficient e(50).*

$$ALI = \frac{0.02Sv}{e(50)}$$

Where:

e(50) is the dose coefficient from Annexe B of ICRP-68

However, the NRC's 1999 memorandum states;

> *It is generally agreed among the national and international scientific community that the newer models provide more accurate dose estimates than the models used in Part 20. In view of this situation, the staff recommends approving the licensee's request to use the newer models. Licensees requesting to use the newer models would still be bound to the same dose limits in Part 20, but they would presumably be making more accurate estimates of doses resulting from intakes of radioactive materials than they can using the current Part 20 Model.*

The annual dose limit to be used in the ICRP 68 provided equation will be 0.05 Sv rather than the ICRP 60 [ICRP 1991] recommended 0.02 Sv, and thus the following equation[1] will be used to define all ALI values for isotopes within the uranium, actinium and thorium series;

$$ALI = \frac{0.05Sv}{e(50)}$$

[1] The stated equation solves for the annual limit on intake in the standard units of Bq. Unit conversion to μCi is performed in order to be consistent with the replaced 10CFR20 App. B limits.

Additionally, a new assumption is defined within ICRP 68 regarding the particle size of the source term. Section 2.1 states;

> *To calculate dose coefficients for inhalation of radionuclides by workers, the reference subject is taken to be a normal nose-breathing adult male at light work. For occupational exposure the default value now recommended for the Activity Median Aerodynamic Diameter (AMAD) is 5 μm (ICRP Publication 66, Paragraph 181), which is considered to be more representative of workplace aerosols than the 1 μm default value adopted in ICRP Publication 30 (Dorrian and Bailey, 1995).*

Therefore all ALI and DAC calculations will be derived from the dose coefficients for a particle size of 5 μm AMAD.

It is important to distinguish the similarities and differences between the ICRP 30 and ICRP 68 based exposure models. In both models, the "f_1" value is defined as the fraction of a specific radioisotope that is directly absorbed to body fluids from the gastro-intestinal tract. ICRP 68 defines absorption "types" for radioisotopes. Types F, M, and S loosely correlate with the inhalation classes D, W, and Y used within ICRP 30. Type F materials are readily absorbed into the bloodstream from the respiratory tract, "Fast" rate of absorption. Type M materials have intermediate rates of absorption into the bloodstream from the respiratory tract, "Moderate" rate of absorption. Finally, Type S materials are relatively insoluble in the respiratory tract, "Slow" rate of absorption. The approximate half-times for these absorption rates correspond to:

Type F (fast):	10 min (100%)	
Type M (moderate):	10 min (10%); 140 day (90%)	
Type S (slow):	10 min (0.1%); 7000 day (99.9%)	[ICRP 1995]

An example of the methodology used to calculate an inhalation ALI for 5 μm AMAD Type M U^{238} is presented below:

$$ALI = \frac{0.05 Sv}{1.6 \times 10^{-6} Sv/Bq} = 31250 Bq$$

Where:

- 0.05 Sv is the annual occupational exposure limit
- 1.6×10^{-6} Sv/Bq is the Type M U^{238} dose coefficient, *e(50)* from ICRP 68

A unit conversion from Bq to μCi is performed in the following manner:

$$31250 Bq \times \frac{1 \mu Ci}{3.7 \times 10^{4} Bq} = 0.84 \mu Ci$$

$$ALI = 0.84 \mu Ci$$

In the calculation of DAC values, 10CFR20 Appendix B Table 1 was referenced;

> *The derived air concentration (DAC) values are derived limits intended to control chronic occupational exposures. The relationship between the DAC and the ALI is given by: DAC=ALI(in μCi)/(2000 hours per working year x 60 minutes/hour x 2 x 10^4 ml per minute)=[ALI/2.4x10^9] μCi/ml, where 2x10^4 ml is the volume of air breathed per minute at work by "Reference Man" under working conditions of "light work."*

Therefore the DAC values for the isotopes within the uranium, actinium and thorium series will be defined with the following equation;

$$DAC = \frac{ALI}{2.4\times10^{9}\,mL}$$

Continuing on from the previous example that determined the ALI, the DAC value for an inhalation of 5 μm AMAD Type M U^{238} would be calculated in the following manner:

$$DAC = \frac{0.84\mu Ci}{2.4\times10^{9}\,mL}$$

$$DAC = 3.5\times10^{-10}\,\mu Ci/mL$$

Summary

The calculated annual occupational limits and derived air concentrations are acceptable to replace the values contained within Title 10, Code of Federal Regulations Part 20, Appendix B Table 1 when intakes are determined through ICRP-68 based methodology. Effluent concentration limits and sewer disposal limits defined within 10CFR20 Appendix B Tables 2 and 3 may similarly be derived based upon methodology defined within the introduction to appendix B of 10CFR20 in conjunction with ICRP 68 dose coefficients.

Utilizing the guidance provided within ICRP 68 in conjunction with existing regulatory guidance, current ALI and DAC values are calculated[2]. The occupational limits for the uranium, actinium, and thorium series are summarized within Appendix A of this document.

[2] All ALI/DAC are rounded to 2 significant figures using standard rounding methods. This is consistent with the number of significant figures provided within the ICRP 68 dose coefficient table.

References

ICRP 1971
ICRP Publication 26: Recommendations of the International Commission on Radiological Protection
Annals of the ICRP Volume 1 Number 3

ICRP 1979
ICRP Publication 30: Limits for Intakes of Radionuclides by Workers: Part 1 Annals of the ICRP Volume 2 Number 3/4.

ICRP 1991
ICRP Publication 60: 1990 Recommendations of the International Commission on Radiological Protection
Annals of the ICRP Volume 21/1-3

ICRP 1995
ICRP Publication 68: Dose Coefficients for Intakes of Radionuclides by Workers, Annals of the ICRP Volume 24/4, Replacement of ICRP Publication 61
Replacement of ICRP Publication 61/Annals of the ICRP Vol. 24/2.

USNRC 1991
http://www.nrc.gov/reading-rm/doc-collections/cfr/part020/full-text.html#part020-1001
Source: 56 FR 23391, May 21, 1991, unless otherwise noted.

USNRC 1999
NRC Staff Requirements Memorandum, To Request Commission Approval to Grant Exemptions from Portions of 10 CFR Part 20 (SECY-99-077)

Appendix A. Annual Limits on Intake and Derived Air Concentrations[3]

	Annual Limits on Intake					Derived Air Concentration	
	Inhalation (5 µm AMAD)			Ingestion		(5 µm AMAD)	
	Type	f_1	ALI (µCi)	f_1	ALI (µCi)	Type	DAC (µCi/ml)
Uranium							
U-238	F	0.020	2.3E+00	0.020	3.1E+01	F	9.7E-10
	M	0.020	8.4E-01	0.002	1.8E+02	M	3.5E-10
	S	0.002	2.4E-01			S	9.9E-11
U-235	F	0.020	2.3E+00	0.020	2.9E+01	F	9.4E-10
	M	0.020	7.5E-01	0.002	1.6E+02	M	3.1E-10
	S	0.002	2.2E-01			S	9.2E-11
U-234	F	0.020	2.1E+00	0.020	2.8E+01	F	8.8E-10
	M	0.020	6.4E-01	0.002	1.6E+02	M	2.7E-10
	S	0.002	2.0E-01			S	8.3E-11
Protactinium							
Pa-234	M	5.0E-04	2.5E+03	5.0E-04	2.6E+03	M	1.0E-06
	S	5.0E-04	2.3E+03			S	9.7E-07
Pa-231	M	5.0E-04	1.5E-02	5.0E-04	1.9E+00	M	6.3E-12
	S	5.0E-04	7.9E-02			S	3.3E-11
Thorium							
Th-234	M	5.0E-04	2.5E+02	5.0E-04	4.0E+02	M	1.1E-07
	S	2.0E-04	2.3E+02	2.0E-04	4.0E+02	S	9.7E-08
Th-232	M	5.0E-04	4.7E-02	5.0E-04	6.1E+00	M	1.9E-11
	S	2.0E-04	1.1E-01	2.0E-04	1.5E+01	S	4.7E-11
Th-231	M	5.0E-04	3.7E+03	5.0E-04	4.0E+03	M	1.5E-06
	S	2.0E-04	3.4E+03	2.0E-04	4.0E+03	S	1.4E-06
Th-230	M	5.0E-04	4.8E-02	5.0E-04	6.4E+00	M	2.0E-11
	S	2.0E-04	1.9E-01	2.0E-04	1.6E+01	S	7.8E-11
Th-228	M	5.0E-04	5.9E-02	5.0E-04	1.9E+01	M	2.4E-11
	S	2.0E-04	4.2E-02	2.0E-04	3.9E+01	S	1.8E-11
Th-227	M	5.0E-04	2.2E-01	5.0E-04	1.5E+02	M	9.1E-11
	S	2.0E-04	1.8E-01	2.0E-04	1.6E+02	S	7.4E-11
Actinium							
Ac-228	F	5.0E-04	4.7E+01	5.0E-04	3.1E+03	F	1.9E-08
	M	5.0E-04	1.1E+02			M	4.7E-08
	S	5.0E-04	1.1E+02			S	4.7E-08
Ac-227	F	5.0E-04	2.1E-03	5.0E-04	1.2E+00	F	8.9E-13
	M	5.0E-04	9.0E-03			M	3.8E-12
	S	5.0E-04	2.9E-02			S	1.2E-11

[3] To be used in place of 10CFR20 Appendix B Table 1 Values

Appendix A (continued)

	Annual Limits on Intake					Derived Air Concentration (5 µm AMAD)	
	Inhalation (5 µm AMAD)			Ingestion			
	Type	f_1	ALI (µCi)	f_1	ALI (µCi)	Type	DAC (µCi/ml)
Radium							
Ra-228	M	0.200	7.9E-01	0.200	2.0E+00	M	3.3E-10
Ra-226	M	0.200	1.1E-01	0.200	4.8E+00	M	4.7E-11
Ra-224	M	0.200	5.6E-01	0.200	2.1E+01	M	2.3E-10
Ra-223	M	0.200	2.4E-01	0.200	1.4E+01	M	9.9E-11
Francium							
Fr-223	F	1.000	1.0E+03	1.000	5.9E+02	F	4.3E-07
Polonium							
Po-210	F	0.100	1.9E+00	0.100	5.6E+00	F	7.9E-10
	M	0.100	6.1E-01			M	2.6E-10
Bismuth							
Bi-214	F	0.050	1.1E+02	0.050	1.2E+04	F	4.7E-08
	M	0.050	6.4E+01			M	2.7E-08
Bi-212	F	0.050	9.0E+01	0.050	5.2E+03	F	3.8E-08
	M	0.050	3.5E+01			M	1.4E-08
Bi-210	F	0.050	9.7E+02	0.050	1.0E+03	F	4.0E-07
	M	0.050	2.3E+01			M	9.4E-09
Lead							
Pb-214	F	0.200	2.8E+02	0.200	9.7E+03	F	1.2E-07
Pb-212	F	0.200	4.1E+01	0.200	2.3E+02	F	1.7E-08
Pb-211	F	0.200	2.4E+02	0.200	7.5E+03	F	1.0E-07
Pb-210	F	0.200	1.2E+00	0.200	2.0E+00	F	5.1E-10