ENCLOSURE 1

REVISED EMERGENCY PLAN (E-PLAN)
IMPLEMENTING PROCEDURES (EPIPs)
REC/RPSS HANDBOOK

Emergency Plan, Revision 58 (176 pages)

F3.17 – Core Damage Assessment, Revision 16 (52 pages)

F3.17.1 – Core Damage Determination, Revision 6 (21 pages)

REC/RPSS Handbook, Revision 26 (178 pages)

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1.0 DEFINITIONS

Listed below are some terms in this plan along with the definitions that should be applied to these terms when they are used in this plan.

1.1 Assessment Action - Actions taken during or after an accident to obtain and process information necessary to make decisions regarding emergency measures.

1.2 Corrective Actions - Emergency measures taken to terminate an emergency situation at or near the source in order to prevent or minimize a radioactive release, e.g., shutting down equipment, firefighting, repair and damage control, etc.

1.3 Emergency Action Level (EAL) – A predetermined, site-specific, observable threshold for a plant Initiating Condition (IC) that places the plant in a given emergency class. An EAL can be: an instrument reading; an equipment status indicator; a measurable parameter (onsite or offsite); a discrete, observable event; results of analyses; entry into specific emergency operating procedures; or another phenomenon which, if it occurs, indicates entry into a particular emergency class.

1.4 Emergency Class: - One of a minimum set of names or titles, established by the Nuclear Regulatory Commission (NRC), for grouping of normal nuclear power plant conditions according to (1) their relative radiological seriousness, and (2) the time sensitive onsite and off site radiological emergency preparedness actions necessary to respond to such conditions. The existing radiological emergency classes, in ascending order of seriousness, are called: Notification of Unusual Event (UE), Alert, Site Area Emergency (SAE), and General Emergency (GE).

1.5 Emergency Director (ED) - The Plant Manager or designee. This individual has overall responsibility and authority for managing the emergency effort within the plant. This person will also manage efforts external to the plant until the Emergency Operations Facility (EOF) Organization can relieve the ED of external tasks.

1.6 Emergency Manager (EM) - A designated member of site management. This person has the authority and responsibility for the management of (NSPM) Northern States Power Company – Minnesota overall response to an emergency. The EM will assume command and control at the Emergency Operations Facility and direct the NSPM response efforts.

1.7 Emergency Planning Zones - a defined area around the plant to facilitate emergency planning by state and local authorities, to assure that prompt and effective actions are taken to protect the public in the event of a release of radioactive material. It is defined for:

1.7.1 Plume Exposure Pathway - a 10 mile radius around the plant where the principal exposure source is: (1) whole body exposure to gamma radiation from the plume and from deposited material; and (2) internal exposure from the inhaled radionuclides deposited in the body (Short Term Exposure).

1.7.2 Ingestion Exposure Pathway - a 50 mile radius around the plant where the principal exposure would be from the ingestion of contaminated water or foods such as milk or fresh vegetables (Long Term Exposure). The ingestion exposure pathway includes the plume exposure pathway.

1.8 Emergency Worker - Any individual involved in mitigating the consequences of an emergency situation and/or minimizing or preventing exposure to the offsite population. The emergency worker category includes emergency workers at the plant as well as individuals who are engaged in public service emergency activities - firemen, policemen, medical support, and certain public officials. These are people who voluntarily place themselves as emergency workers.

1.9 Exclusion Area - The area surrounding the plant that is under direct Prairie Island Nuclear Generating Plant control. This includes the Corps of Engineering land north of plant and the islands located in the Mississippi River east of plant. It is sized such that any individual located on its boundary would not exceed 25 REM whole body or 300 REM thyroid from I-131 for two hours immediately following the design basis accident (approximately 2340 feet out to boundary).

1.10 Facility Activation - An Emergency Response Facility is activated when the minimum staff per Figures 1 and 2 is available and the facility is ready to assume its assigned Emergency Plan functions and relieve the on-shift staff of those functions. Although the facility may be ready, the on-shift staff relief may be postponed in the interest of completing critical tasks prior to turnover.

1.11 Initiating Condition (IC): - One of a predetermined subset of nuclear power plant conditions when either the potential exists for a radiological emergency, or such an emergency has occurred.

1.12 Northern States Power Company – Minnesota (NSPM) d/b/a Xcel Energy - Operator of the Prairie Island Nuclear Generating Plant.

1.13 Protective Actions - Emergency measures taken before or after a release of radioactive materials in order to prevent or minimize radiological exposures to the population.

1.14 Protective Action Guides (PAG) – Projected dose to individuals, that warrants protective action prior to and/or following a radioactive release.

1.15 Recovery Actions - Actions taken after an emergency to restore the plant to normal.

1.16 Xcel Energy - Operating Utility of Northern States Power Company – Minnesota.

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2.0 SCOPE AND PURPOSE

In accordance with license conditions, 10CFR Part 50, and NRC guidance, the Northern States Power Company – Minnesota (NSPM) has developed and implemented a radiological emergency response plan for the Prairie Island Nuclear Generating Plant (PINGP) and a joint off-site plan for the PINGP and the Monticello Nuclear Generating Plant. As asset owner NSPM, and Xcel Energy, the operating utility, retain all owner obligations.

This Emergency Plan is applicable to Prairie Island Nuclear Generating Plant (PINGP), Units 1 and 2.

In any emergency situation at Prairie Island, the initial response to activate the Emergency Plan is accomplished by the plant staff and, if needed, immediate actions may be required by local support agencies. The plant, during initial stages of the emergency situation, must function independently coordinating both onsite and offsite activities. The augmented response organization will assume those tasks external (offsite) to the plant, thus allowing the plant staff to be responsible for all onsite activities. This plan covers the actions and responsibilities of the PINGP Emergency Organization and the Emergency Operations Facility Organization.

The purpose of the plan is to describe the following:

2.1 Organization and actions within the plant to control and limit the consequences of an accident.

2.2 Organization and actions controlling site and offsite activities in the event of an uncontrolled release of radioactive material. This includes notification of and coordination with required offsite support agencies.

2.3 Identifying and evaluating the consequences of accidents that may occur and affect the public and plant personnel.

2.4 Describing the protective action levels and actions that are required to protect the public and plant personnel in the event of an accident.

2.5 Consideration necessary for the purpose of reentry and short-term recovery.

2.6 Arrangements required for medical support in the event of injury.

2.7 Arrangements required for fire fighting support in the event of major fires requiring outside support.

2.8 The training necessary to assure adequate response to emergencies.

The Emergency Plan is dependent upon various standing plant operating, abnormal operating, emergency operating, plant safety, radiological control and security procedures and the Emergency Plan Implementing Procedures for the implementation of the plant's response to the spectrum of emergency situations.

PINGP has procedures in place that implement on-site protective actions and personnel accountability during security events that are appropriate for plant and environmental conditions.

Coordination between plant, state, local and tribal authorities is defined in the Minnesota and Wisconsin state emergency operations plans, Goodhue, Dakota and Pierce county emergency plans and the Prairie Island Indian Community's emergency plan. Goodhue, Dakota and Pierce Counties have, formulated for their respective areas, individual evacuation plans which are included in the respective state plans.

Monticello & Prairie Island (MT & PI) offsite response is detailed in the Corporate Nuclear Emergency Plan.

3.0 SUMMARY

Abnormal events, both realized and potential, requiring emergency preparedness response are classified into four classes of Emergency Action Levels. The four levels of emergency classes, in increasing order of severity are:

3.1 Notification of Unusual Event (UE)

3.2 Alert

3.3 Site Area Emergency (SAE)

3.4 General Emergency (GE)

Each class requires specific immediate actions on the part of the plant staff in order to protect the public, plant personnel and property. As the severity level of the emergency increases, so does the response of the offsite agencies, in order to protect the public.

The lowest class (least severe) is the Notification of Unusual Event, and will be handled mainly by plant personnel, with only advisory notification to local and state authorities. The Alert Classification requires prompt notification of local and state authorities, which will place their various organizations in a standby mode. In both the Notification of Unusual Event and the Alert Classification, the plant staff is expected to restore the situation to normal without further or minimum involvement of offsite authorities. The two higher severity classes, the Site Area and General Emergency, (the General Emergency being the most severe), requires prompt notification of offsite authorities with immediate involvement of those organizations to assess the emergency situation and to implement the required protective actions for the general public.

During an Alert, Site Area, or General Emergency, Prairie Island Nuclear Generating Plant will automatically activate their site and offsite support emergency response organizations. The normal site organization will staff the Plant Emergency Response Organization and the Emergency Operations Facility (EOF) Organization. The offsite organization will be staffed by members of the MT & PI Offsite Organization and be located at the Minnesota Emergency Operations Center. MT & PI Offsite Organization will communicate to the plant via the EOF Organization. The EOF Organization will support emergency response for the plant and relieve plant personnel of offsite activities who may be needed for plant activities.

When plant conditions stabilize and the potential for future degradation of plant conditions is small, the plant may terminate the emergency classification. If severe equipment or core damage has occurred, a transition to a recovery phase may be warranted. In general terms, an Unusual Event or Alert may be terminated without transition to Recovery while a Site Area Emergency or General Emergency will probably necessitate a planned transition to Recovery and the establishment of a Recovery Organization. The Recovery Organization will manage the overall recovery or post-accident outage plans as work is done to return the plant to a normal operational or shutdown status.

PINGP has and maintains the capability to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an EAL has been exceeded. Upon identification of the appropriate emergency classification level the emergency condition will be promptly declared.

4.0 EMERGENCY CONDITIONS

4.1 Classification System

Four Emergency Classification Levels (ECLs) are established, according to severity, taking into consideration potential as well as actual events in progress.

Initiating Conditions (ICs) are predetermined subset of plant conditions when either the potential exists for a radiological emergency, or such an emergency has occurred.

Emergency Action Levels (EALs) are plant-specific indications, conditions or instrument readings that are utilized to classify emergency conditions.

Annex A contains the Emergency Action Level (EAL) scheme as established by NEI 99-01, Revision 6.

It should be noted that various events could require a graded scale of response. A minor incident could increase in severity and advance to the next class of emergency. This Emergency Plan is constructed to provide for a smooth transition from one class to another.

4.1.1 Notification of Unusual Event (UE)

Notification of Unusual Events are events that are in process or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

The purpose of the Notification of Unusual Event action level is to: (1) have the operating staff come to a state of readiness from the standpoint of emergency response in the event the handling of the initiating condition requires escalation to a more severe action level class; and (2) provide for systematic handling of unusual event information, i.e., to provide early and prompt notification of minor events which could lead to more serious consequences given operator error or equipment failure or which might be indicative of more serious conditions which are not yet fully realized.

4.1.2 Alert

At the Alert action level, events are in process or have occurred which involve actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. It is the lowest level when some necessity for emergency planning and response offsite is necessary. Any radioactive release will be limited to a small fraction of the EPA Protective Action Guideline exposure levels.

The purpose of the Alert action level is to: (1) assure that emergency personnel are readily available to respond if the situation becomes more serious or to perform confirmatory radiation monitoring if required; and (2) provide offsite authorities current status information, i.e., early and prompt notification of minor events which could lead to more serious consequences given operator error or equipment failure or which might be indicative of more serious conditions which are not yet fully realized.

4.1.3 Site Area Emergency

The Site Area Emergency action level describes events that are in process or have occurred which involve actual or likely major failure of plant functions needed for protection of the public or HOSTILE ACTION that result in intentional damage or malicious acts; (1) toward site personnel or equipment that could lead to the likely failure of or; (2) that prevent effective access to equipment needed for the protection of the public. It reflects conditions where significant offsite releases are likely to occur or are occurring but where a core melt situation is not expected although severe fuel damage may have occurred. Any radioactive releases are not expected to exceed the EPA Protective Action Guideline exposure levels except near the site boundary.

The purpose of the Site Area Emergency action level is to: (1) assure that response centers are manned; (2) assure that monitoring teams are dispatched; (3) assure that personnel required for evacuation of near-site areas are at duty stations if the situation becomes more serious; (4) provide current information for and consultation with offsite authorities; and (5) provide updates for the public through offsite authorities.

4.1.4 General Emergency

The General Emergency action level describes events in process or have occurred which involve actual or imminent substantial core degradation or melting with the potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Radioactive releases can be reasonably expected to exceed the EPA Protective Action Guidelines exposure levels offsite for more than the immediate site area.

The purpose of the General Emergency class is to: (1) initiate predetermined protective actions for the public; (2) provide continuous assessment of information from licensee and offsite organization measurements; (3) initiate additional measures as indicated by actual or potential releases; (4) provide consultation with offsite authorities; and (5) provide updates for the public through offsite authorities.

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5.0 ORGANIZATIONAL CONTROL OF EMERGENCIES

5.1 Normal Site Organization

The normal site organization is comprised of the plant organization and several other site support organizations. The normal site organization can be accessed on the Prairie Island web page. Responsibilities and authorities of the various functional groups are delineated in plant Administrative Work Instructions.

5.2 Normal Plant Organization

The normal plant operating crew is staffed and qualified to perform all actions that may be necessary to initiate immediate protective actions and to implement the emergency plan and is designated as the responsible group for such actions. The normal plant organization can be accessed on the Prairie Island web page.

The Plant Manager has overall responsibility for the safe, efficient operation of the plant and for compliance with operating license requirements. The Plant Manager **SHALL** select, train and supervise a qualified staff.

The Shift Manager reports directly to the Assistant Operations Manager who reports directly to the Operations Manager who reports directly to the Plant Manager. The Shift Manager is responsible for the direction and coordination of the Shift Supervisors on his/her shift to perform operations in accordance with the administrative controls and operating procedures. The Shift Manager coordinates activities with other plant groups as required to maintain the safe operation of the plant.

The Shift Supervisor reports to the Shift Manager. The Shift Supervisor is the single focal point for directing and coordinating the operations group, maintenance group and the plant security activities during his/her shift. The Shift Supervisor **SHALL** assume the primary management responsibility for the safe operation of the plant under all conditions during his/her shift. The responsibility and authority of the Shift Supervisor **SHALL** be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty <u>in the Control Room</u>.

5.3 Plant Emergency Organization

A plant emergency organization is designated to augment the normal operating crew. Provisions have been made for rapid assignment of plant personnel to the plant emergency organization during emergency situations. The Prairie Island Plant Emergency Organization is shown in Figure 1.

Various areas of responsibility are assigned to segments of the plant staff during emergency situations as depicted in Table 1. Table 1 shows the personnel available on-shift and the capability for additional personnel within 60 minutes and 90 minutes of event declaration. Table 1 follows the guidance developed in accordance with 10 CFR 50 Appendix E. This staffing analysis is documented in F3-1.1, Emergency Plan On-Shift Staffing.

5.3.1 Direction and Coordination

During the initial stages of an emergency condition at Prairie Island Nuclear Generating Plant, the Emergency Director has overall coordinating authority for Northern States Power Company – Minnesota (NSPM). The Emergency Director alone has the authority and responsibility to immediately initiate any emergency actions, including providing protective action recommendations to offsite authorities responsible for implementing offsite emergency measures.

The Shift Supervisor, of the affected unit, until properly relieved, **SHALL** remain in the Control Room at all times during accident situations, to direct the activities of control room operators. If necessary, the Shift Supervisor of the unaffected unit may function as an alternate Emergency Director backing up the Shift Manager. Twenty-four (24) hour coverage for the Emergency Director position is provided by the Duty Shift Manager who assumes the responsibility of the TSC Emergency Director at the onset of any emergency condition.

When the Technical Support Center (TSC) and Emergency Operations Facility (EOF) Organizations are activated, the Emergency Director (ED) and TSC staff will relieve the Emergency Director on shift of command and control functions as soon as practical and assume the responsibility for the management of NSPM's overall response to the emergency. The Emergency Director on shift can then direct the plant's priorities for event responses. Upon activation of the EOF, responsibility for offsite functions of notification and protective action recommendations transfer from the TSC to the EOF Emergency Manager (EM). The transition of command and control functions is depicted below.

CONTROL ROOM	TSC	EOF
<u>On Shift/Emergency Director</u>	<u>TSC Emergency Director</u>	<u>EOF Emergency Manager</u>
Classification →	Classification	
Notifications →	Notifications →	Notifications
PARs →	PARs →	PARs
Emergency Exposure Controls →	Emergency Exposure Controls	

Figure 1 Prairie Island Plant Emergency Organization

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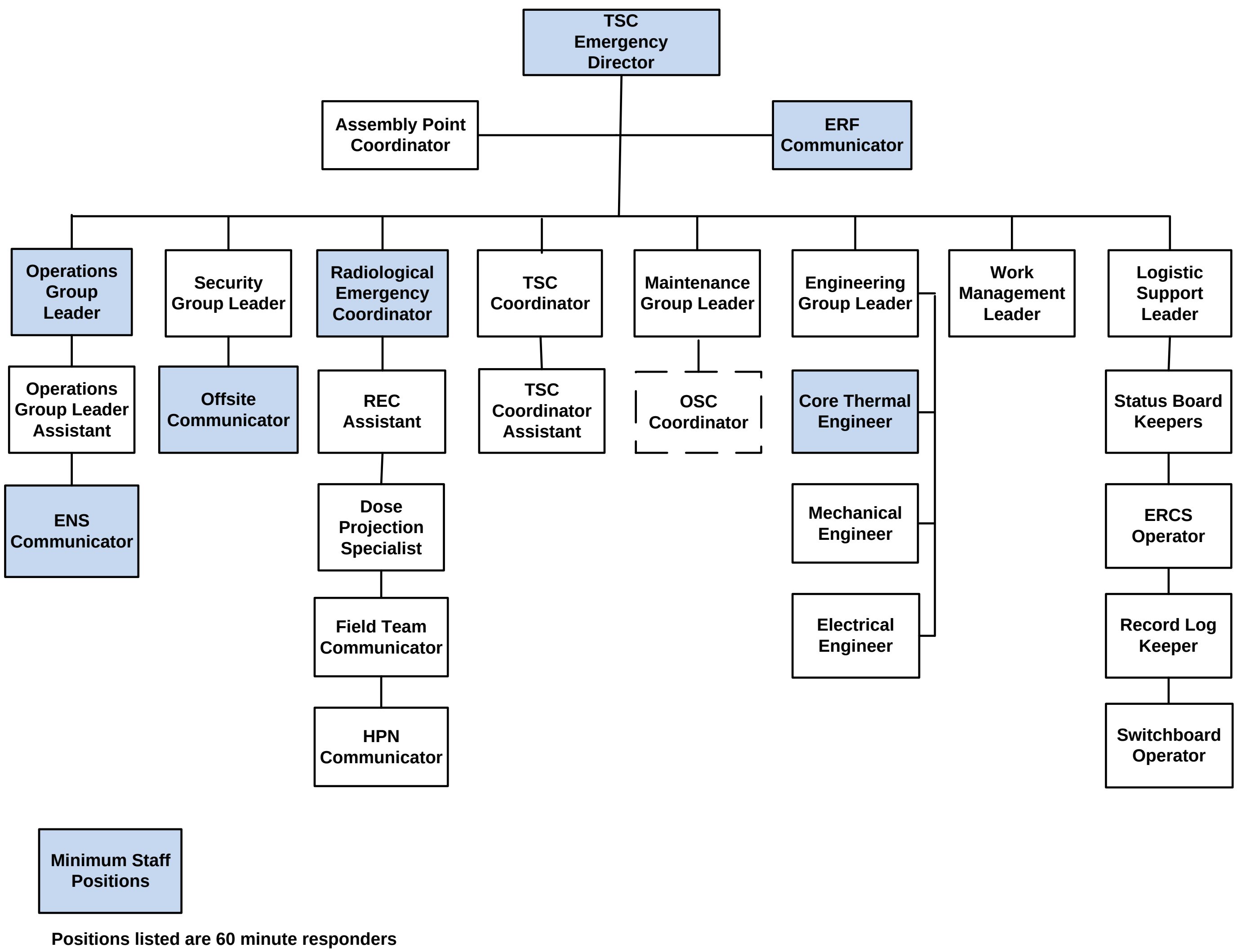

Figure 1 Prairie Island Plant Emergency Organization

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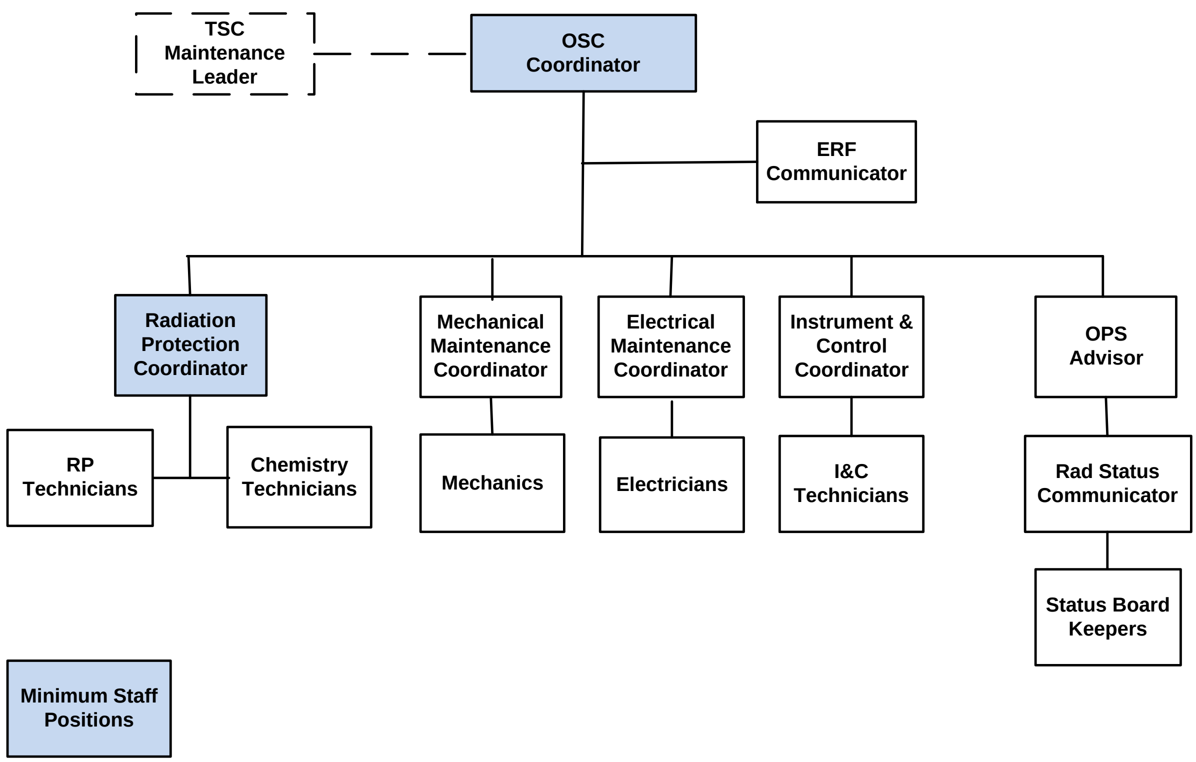

Positions listed are 60 minute responders unless otherwise noted on Table 1

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Table 1 Guidance for Augmentation of Plant Emergency Organization

Major Functional Area	Major Tasks	Position Title or Expertise	On-Shift	Capability for Additions 60 min	90 min
Plant Operations and Shift Supervisor (SRO): Assessment of Control Room Reactor Operational Aspects		Shift Manager/ED	1	-	-
		Unit Supervisors	2	-	-
		Reactor Operators (RO)	4	-	-
		Auxiliary Operators	6	-	-
Notification/ Communication	Notify State, local and Federal personnel & maintain communication	Shift Emergency Communicator	1	-	-
		Offsite Communicator	-	1	1
		ENS Communicator	-	1	1
Radiological Accident Assessment and Support of Operational Accident Assessment	Emergency Operations Facility (EOF) Director	Emergency Manager	-	-	1
		Emergency Director (TSC)	-	1	-
	Offsite Dose Assessment	Radiological Emergency Coordinator	-	1	-
		RP Support Supervisor	-	-	1
	Offsite Surveys	Radiation Protection Specialist/Support	-	2	2
	Onsite Surveys (out-of-plant) In-Plant Surveys	Radiation Protection Specialist	1	1	1

Table 1 Guidance for Augmentation of Plant Emergency Organization

Major Functional Area	Major Tasks	Position Title or Expertise	On-Shift	Capability for Additions 60 min	90 min
	Chemistry/ Radiochemistry	Chemistry Technician	1	1	-
Plant System Engineering	Technical Support	Shift Technical Advisor	1	-	-
		Core/Thermal Engineer (TSC)	-	1	-
		Electrical	-	1	-
		Mechanical	-	1	-
Repair and Corrective Actions	Repairs and Corrective Actions	Mechanical Maintenance	-	1	-
		Electrical Maintenance		1	-
		Instrument Control	-	-	1
Protective Actions (In-Plant)	Radiation Protection: a. Access Control b. HP Coverage for repair, corrective actions, search and rescue, first-aid & firefighting c. Personnel monitoring d. Dosimetry	Radiation Protection Specialist	1	1	1

Table 1 Guidance for Augmentation of Plant Emergency Organization

Major Functional Area	Major Tasks	Position Title or Expertise	On-Shift	Capability for Additions 60 min	90 min
Fire Fighting			Fire Brigade per F5	Local Support	
Rescue Operations and First Aid			2[1]	Local Support	
Site Access Control and Personnel Accountability	Security, firefighting communications, personnel accountability	Security Personnel	As per Security Plan		
		TOTAL	18	14	9

[1] May be provided by shift personnel assigned other functions.

The above table was developed in accordance with 10 CFR 50 Appendix E. This staffing analysis is documented in F3-1.1, Emergency Plan On-Shift Staffing Analysis.

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The Shift Manager **SHALL** be relieved of the Emergency Director responsibilities when the designated Emergency Director arrives onsite. The Plant Manager or Designee **SHALL** be the designated Emergency Director and will be available with a pager on a twenty-four (24) hour basis. When the Plant Manager is unavailable, (e.g., out of town), the designated Emergency Director responsibility will be passed onto another Plant Manager designee who is a member of senior plant management. Specific personnel assignments to the Emergency Director position are found in the MT & PI Nuclear Emergency Preparedness Telephone Directory.

The Shift Manager **SHALL** start the tasks assigned to the Emergency Director, (e.g., notification, activating onsite centers, etc.). These tasks **SHALL** be accomplished promptly and cannot wait for the designated individual to arrive at the plant site.

The Emergency Director's responsibilities are as follows:

A. Activation of onsite emergency organization -

   1. Direct the activation of the onsite emergency response centers and monitor their habitability, and

   2. Coordinate response of the plant onsite emergency organization.

B. Personnel accountability - During a plant evacuation the Emergency Director **SHALL** account for all personnel onsite within thirty minutes of the Site Area or General Emergency requiring the evacuation so that a search for missing personnel can be conducted. A continuous personnel accountability **SHALL** be maintained throughout the emergency. This responsibility may be delegated to a designated individual with assistance from the security force.

C. Radiological monitoring - The Emergency Director **SHALL** direct radiological monitoring of all personnel onsite and at the onsite assembly area, for contamination and/or excessive exposure. This responsibility may be delegated to the Radiation Protection Specialists or to a qualified operations member.

D. Exposure - The Emergency Director **SHALL** be responsible to authorize overexposures in excess of the normal limits (this responsibility may not be delegated).

E. Radiation Survey Teams - The Emergency Director **SHALL** direct the Radiation Survey Teams to obtain the necessary onsite and offsite samples and/or radiation surveys. This responsibility may be delegated to the Radiological Emergency Coordinator.

F. Offsite Dose Projections - The Emergency Director **SHALL** be responsible to project dose rates to the offsite population. This responsibility may be delegated to the Radiological Emergency Coordinator.

G. Protective Action - The Emergency Director **SHALL** be responsible for authorizing offsite Protective Action Recommendations (this responsibility may not be delegated and is relinquished to the Emergency Manager when the EOF is activated).

H. Notification - The Emergency Director **SHALL** be responsible to ensure that the necessary offsite notifications are initiated and completed. This responsibility may be delegated to the Shift Emergency Communicator (SEC). The SEC may designate offsite communications to a qualified Communicator.

1. Immediate (within 15 minutes)

The <u>initial</u> notification message to State, local and tribal authorities, from the plant, **SHALL** contain the following information:

a Class of emergency
b Whether radioactivity is being released and in what form (liquid or gas)
c Potentially affected populace and area, if any
d Necessity of protective measures
e Brief description of the event

Other information, i.e., meteorological data, etc., are available to these authorities via the follow-up notification messages.

2. Subsequent Messages

The plant will continue to provide updating information to offsite authorities. As soon as possible after the initial notification of an Alert, Site Area, or General Emergency, as much of the following information that is known and appropriate will be forwarded to offsite authorities:

a Location of incident
b Name and telephone number of caller
c Date/time of incident
d Class of emergency
e Type of release (airborne, liquid, surface spill) and estimated duration
f Estimate of noble gas, iodine, and particulate release rates
g Prevailing weather conditions (wind speed, wind direction, temperature, atmospheric stability class, precipitation, if any)
h Actual or projected dose rates at site boundary
i Projected dose rate and integrated dose at 2, 5 and 10 miles and the Sectors affected.
j Survey results of offsite dose rates or any surface contamination
k Plant emergency response actions in progress
l Request for onsite support from offsite support organizations
m Prognosis for worsening or termination of event based on plant information

To provide ease in supplying the aforementioned information, a standardized form is used and incorporated into the implementing procedures.

I. Protracted Emergency Shift Coverage - The Emergency Director, with assistance from and coordination with other group Managers and Supervisors, **SHALL** ensure that work force requirements for all subsequent work shifts are determined and the necessary personnel are scheduled for the specific time period.

5.3.2 Plant Emergency Organization Coordinators

A. Technical Support Center Coordinator

The Technical Support Center (TSC) Coordinator **SHALL** be responsible for the general activation, operation and coordination of activities in the Technical Support Center (TSC). Specific personnel assignments to the TSC Coordinator are found in the MT & PI Nuclear Emergency Preparedness Telephone Directory.

The responsibilities of the TSC Coordinator are:

1. Establish and verify radiological monitoring for the TSC;

2. Assist personnel performing the accountability check;

3. Coordinate activities of plant and non-plant personnel located in the TSC;

4. Periodically update personnel located in the TSC with appropriate information;

5. Maintain any necessary status boards;

6. Ensure technical guidance is provided to the Emergency Director and Control Room Operators on plant operations;

7. Establish or ensure that communications are established between all onsite emergency facilities and the EOF.

8. Ensure the Emergency Response Data System data link is established with the NRC's emergency center.

B. Operational Support Center Coordinator

The Operational Support Center Coordinator **SHALL** be responsible for the general activation, operation, and coordination of activities in the Operational Support Center (OSC). Specific personnel assignments to the OSC Coordinator are found in the MT & PI Nuclear Emergency Preparedness Telephone Directory.

The responsibilities of the OSC Coordinator are:

1. Establish and verify radiological monitoring for the OSC and the Control Room;
2. Coordinate activities of plant personnel located in the OSC to support plant operations as requested by the Control Room and TSC.
3. Assist personnel performing the accountability check in the OSC and the Control Room.
4. Maintain the communication systems in the OSC. A person may be designated to act as a communicator.
5. Periodically update personnel located in the OSC with appropriate information.
6. Control the use of equipment located in the emergency locker.

C. Assembly Point Coordinator

The Assembly Point Coordinator **SHALL** be responsible for the general operation of the assembly area. Specific personnel assignments to the Assembly Point Coordinator are found in the MT & PI Nuclear Emergency Preparedness Telephone Directory.

The responsibilities of the Assembly Point Coordinator are:

1. Verify that radiological monitoring has been established for the Assembly Point.
2. Coordinate activities of all personnel (plant and non-plant) located at the Assembly Point.
3. Assist the Emergency Director in performing the accountability check, as necessary.

4. Maintain the communication systems. A person may be designated as the communicator, if necessary.

5. Control the use of equipment located in the Emergency Locker.

6. Update all personnel with appropriate information when directed by the Emergency Director.

7. Provide instructions to personnel when they are released from the assembly point for reentry or transport offsite.

D. Radiological Emergency Coordinator

The Radiological Emergency Coordinator (REC) **SHALL** be responsible for radiological accident assessment, onsite and offsite. The REC should report to the Technical Support Center when the TSC is activated. Upon activation of the EOF, the Radiation Protection Support Supervisor will assume responsibility for the offsite activities. The REC should transfer the responsibility for offsite accident assessment to the Radiation Protection Support Supervisor at the EOF. Specific personnel assignments to the Radiological Emergency Coordinator are found in the MT & PI Nuclear Emergency Preparedness Telephone Directory.

The responsibilities of the REC are:

1. Offsite dose assessment
2. Formulating offsite protective action recommendations
3. Offsite surveys
4. Onsite surveys
5. Chemistry
6. Radiochemistry
7. Onsite Radiation Protection for:
   - a Access Control
   - b Damage control and repair
   - c Search and rescue
   - d First-aid
   - e Personnel monitoring and decontamination
   - f Dosimetry

5.3.3 Plant Shift Organization

The following groups comprise the plant's shift organization. Brief descriptions of their emergency responsibilities are included.

A. Operations Group

The Operations Group consists of the Operations Manager, Asst. Operations Manager, Shift Managers, Shift Technical Advisors, Shift Supervisors, and all operators.

The Operations Group **SHALL** have responsibility for:

1. Plant Operations and assessment of operational aspects of the emergency.

2. Short term damage control and repair for electrical, mechanical, and I&C equipment.

B. Security Group

The Security Group consists of the Security Manager, the Security Staff, and the contract Security Force.

The Security Force **SHALL**:

1. Carry out the plant security and Access Control program.
2. Maintain strict personnel accountability onsite.
3. Assist communications efforts when necessary.
4. Assist in first aid treatment.

C. Shift Manager

The Shift Manager (SM) **SHALL** be onsite continuously. The Shift Manager **SHALL** assume overall coordination and control in the Control Room and provide direction as necessary to the Shift Supervisor.

The Shift Manager **SHALL**:

1. Assume the duties of the Emergency Director until relieved by the TSC Emergency Director. Portions of the E-Plan implementation may be delegated to other members of the plant staff as the condition of the plant dictate.
2. Assess the emergency condition, event evaluation, and safety related aspects of the plant.

D. Shift Technical Advisors

Provide technical and engineering support in the area of accident assessment.

E. Shift Emergency Communicator (SEC)

The Shift Emergency Communicator (SEC) **SHALL** be onsite continuously. The SEC is responsible for initial notification to the offsite agencies and maintaining communications during emergency conditions. The SEC may designate offsite communications to a qualified Communicator.

NOTE:	**1. When the EOF is activated, communications with the offsite agencies and personnel will be maintained by the EOF personnel.** **2. As the emergency organization is activated, additional communicators from TSC support personnel should augment the plant staff to assist in communication efforts.**

F. Fire Brigade

The Fire Brigade should consist of:

1. Brigade Chief - Unit 1 Turbine Building APEO or as designated by the Shift Manager.
2. Assistant Chief - Any Qualified APEO.

NOTE:	Usually the APEO from the affected building SHALL fulfill the duties of the Brigade Chief in his absence.

3. Fire Fighters - BOP Operators.

4. Runner - As designated to accompany fire department, operate equipment, bring additional equipment to fire scene.

The Fire Brigade **SHALL** be responsible for firefighting and primary responders for Search and Rescue, as necessary.

The Red Wing Fire Department should provide emergency assistance and **SHALL** be called immediately on report of fire. Other plant personnel on site may be called on for emergency work or called to plant for emergency service.

G. Radiation Protection Specialist

The Radiation Protection Organization consists of two Radiation Protection Specialists (RPS) onsite at all times. The RPS is responsible for conducting routine and special surveys, maintaining Access Control, writing RWP's and providing job coverage as required.

H. Chemistry Technician

One Chemistry Technician is onsite at all times. The Chemistry Technician is responsible for chemistry, radiochemistry, dose assessments, and offsite dose projections. The Chemistry Technician is also cross-trained to support the Radiation Protection Specialist functions described in Section G above.

5.3.4 Plant Emergency Staff Augmentation Groups

A. Maintenance Group

The Maintenance Group consists of all mechanical maintenance personnel, all plant electricians and I&C Specialists. The onsite Emergency Organization includes the Maintenance Manager, who should report to the Technical Support Center (TSC); and the Maintenance Supervisors (mechanical, electrical and I&C), and designated Electricians who should report to the Operational Support Center (OSC). The mechanical, electrical and I&C maintenance staff in the OSC can be further augmented or decreased as emergency conditions dictate.

The Mechanical, Electrical, and I&C Maintenance Group **SHALL** have responsibility for:

1. Supporting the repair and corrective actions for the mechanical, electrical, and I&C systems in support of emergency response and recovery actions.
2. Supporting the Search and Rescue effort.

B. Radiation Protection Group and Chemistry Group

The Radiation Protection and Chemistry Groups consists of the Radiation Protection Manager & Chemistry Manager and all members of the Radiation Protection and Chemistry Groups. Radiation Protection and Chemistry Managers and other designated group members should report to the Technical Support Center. Other Radiation Protection Specialists and Chemistry Technicians should report to the Operational Support Center.

The responsibilities of the Radiation Protection and Chemistry Groups are:

1. Offsite Dose Assessment
2. Offsite Surveys
3. Onsite Surveys
4. Chemistry

5. Radiochemistry

6. Radiation Protection for:

   a Access Control
   b Damage control and repair
   c Search and rescue
   d First aid
   e Fire fighting
   f Personnel monitoring and decontamination
   g Dosimetry

C. <u>Engineering Group</u>

The Engineering Group consists of Systems, Programs, Design and Equipment Reliability.

Upon activation of the onsite emergency organization, Systems and Programs Engineering Managers and designated engineers assigned to the emergency organization should report to the Technical Support Center. Other designated engineers may be requested to further augment engineering support in the TSC.

The Engineering Group **SHALL** have responsibility for:

1. Providing technical support for plant system engineering on electrical/mechanical systems.

2. Providing technical support for operating radioactive waste control systems.

3. Providing core parameter analysis to determine current core status.

4. Providing plant parameter trending and analysis utilizing the Emergency Response Computer System (ERCS).

5. Projecting possible loss of key equipment and its consequences.

6. Providing technical support for emergency repairs and corrective actions on electrical/mechanical systems.

7. Update TSC staff of potential problems and developments.

D. Logistics Support Group

The Logistics Support Group consists of Business Support Group (Administration Services and Document Control), Plant Services, and Site Materials.

Business Support Group **SHALL** supply logistical support in their area of expertise. Personnel in these areas may be called in to provide support for emergency response on an “as needed” basis.

Site Materials **SHALL** provide assistance in retrieving the parts necessary for an emergency response.

Plant Services **SHALL** support an emergency response by providing necessary assistance by the Nuclear Plant Service Attendants.

5.4 EOF Organization

The EOF (Emergency Operations Facility) Organization consists of a Direction and Control Group and three subordinate groups. The EOF Organization is staffed by personnel from the site's Engineering and Project Management groups and Prairie Island Training Center staff. The Prairie Island EOF Organization is shown in Figure 2.

The EOF will be activated within 90 minutes of when an Alert, Site Area Emergency or General Emergency is declared.

5.4.1 EOF Direction and Control

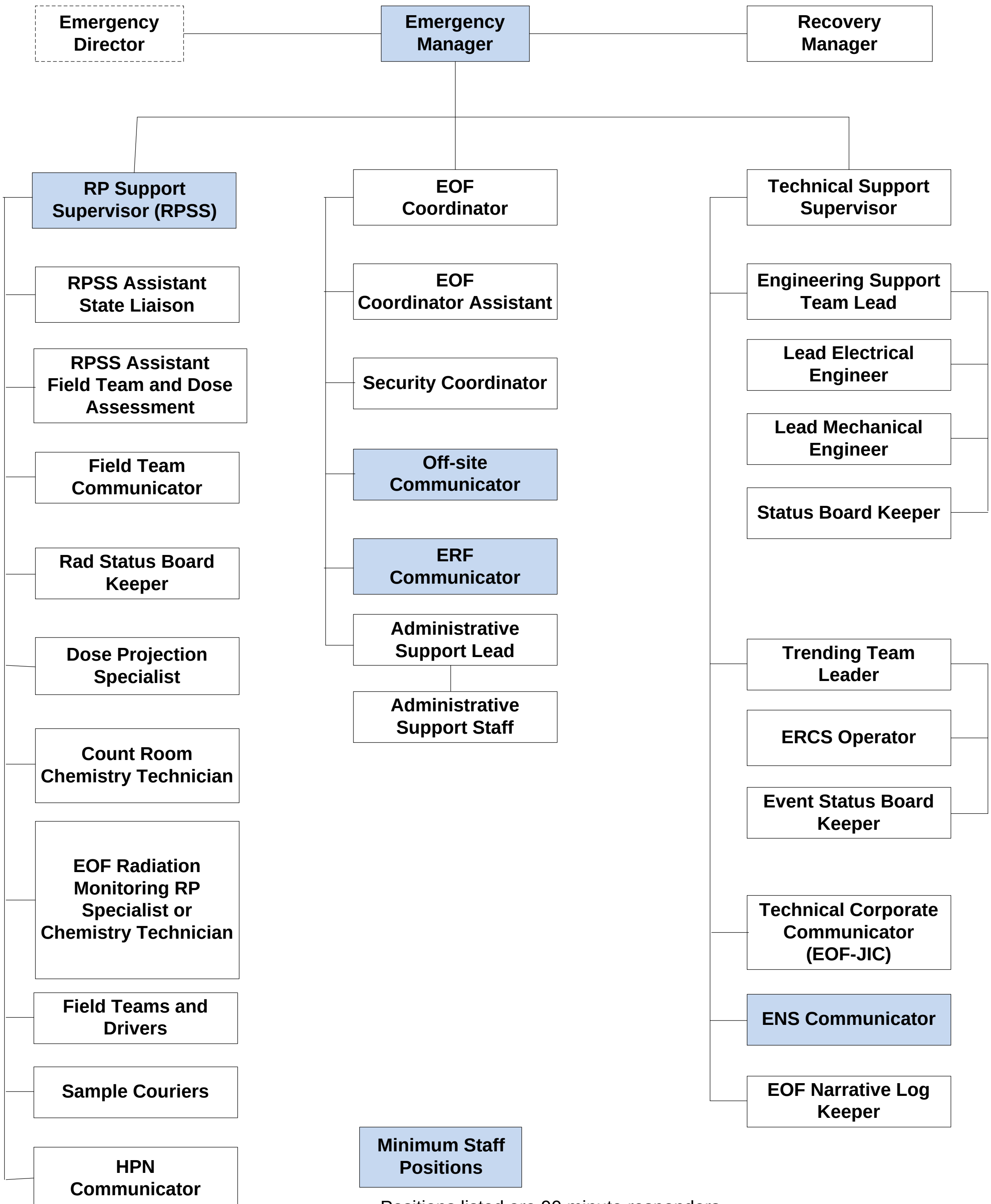
The Emergency Manager is responsible for overall direction and control of NSPM's emergency response effort. Designated members of management staff the Emergency Manager position in the EOF. Specific personnel assignments to the Emergency Manager position are found in the MT & PI Nuclear Emergency Preparedness Telephone Directory. The Emergency Manager relieves the Emergency Director of the following responsibilities:

A. Off-site dose projections and coordination and direction of the utility off-site radiological monitoring teams.

B. Authorization of offsite Protective Action Recommendations.

C. Communications with off-site authorities including Federal, State, Local and Tribal authorities and MT & PI Offsite executive management located at the Minnesota State Emergency Operations Center.

Figure 2 Prairie Island EOF Organization



Positions listed are 90 minute responders

Other responsibilities of the Emergency Manager include:

A. Coordinate the emergency response efforts of other offsite support personnel assisting the plant organization.

B. Obtain and coordinate the services of outside consultants and vendors.

C. Advise utility management on matters related to emergency response efforts and needed resources to support the effort.

5.4.2 EOF Technical Support Group

The EOF Technical Support Group consists of select personnel from the site's Engineering and Project Management groups and Training Center staff. The Technical Support Supervisor is staffed by senior personnel and reports to the Emergency Manager. The Technical Support Group is responsible for trending critical parameters, engineering evaluation in support of the TSC Engineering Group, technical assessment and advising the Emergency Manager on technical matters related to the event.

5.4.3 EOF Radiation Protection Support Group

The EOF Radiation Protection Support Group is staffed by select personnel from the Training Center, plant Radiation Protection and Chemistry Groups and Emergency Plan Group. The Radiation Protection Support Supervisor position is staffed by senior personnel qualified in radiation assessment and reports to the Emergency Manager. The Radiation Protection Support Group includes plant Chemistry personnel for off-site dose projection and EOF Count Room operation and Nuclear Plant Service Attendants who function as sample couriers and drivers for off-site radiological monitoring teams. Radiation Protection Support Group responsibilities include:

A. Direction and coordination of the utility off-site radiological monitoring teams.

B. Off-site dose projection.

C. EOF Count Room activation and operation.

D. EOF habitability, personnel monitoring and decontamination (as necessary).

E. Communications with state assessment groups on matters related to dose projections and off-site protective action recommendations.

F. Staffing the Health Physics Network (HPN) and communications with the NRC (as necessary).

The Radiation Protection Support Supervisor advises the Emergency Manager on matters related to actual or potential radiological impact on the environment, off-site protective action recommendations, and EOF habitability.

5.4.4 EOF General Support Staff

The EOF General Support Staff consists of the EOF Coordinator, emergency communicators, administrative and security support personnel. The EOF Coordinator position is staffed by senior Training Center or site Engineering and Project Management personnel and reports to the Emergency Manager. The EOF Coordinator is responsible for activation and operation of the EOF and assists the Emergency Manager with administrative duties. The emergency communicators, EOF Security Coordinator and Administrative Staff report to the EOF Coordinator. The emergency communicators are responsible for communications with offsite agencies as directed by the Emergency Manager. The Administrative Staff is responsible for emergency document control, recording and document distribution at the EOF. An EOF Coordinator Assistant is responsible for general logistics support and assisting the EOF Coordinator. The EOF Security Coordinator reports to the EOF Coordinator. Responsibilities of EOF Security include EOF access and dosimetry issuance to EOF personnel.

5.5 Recovery Organization

The establishment of the Recovery Organization will be dependent upon the nature and severity of the event or plant conditions. In general terms, an Unusual Event or Alert may be terminated without establishing a special Recovery Organization while a Site Area Emergency or General Emergency will probably necessitate the establishment of a Recovery Organization. The Recovery Organization will manage the overall recovery or post-accident outage plans as work is done to return the plant to a normal operational or shutdown status.

The Recovery Manager is mainly responsible for management of the recovery phase and will perform their initial tasks as directed by the Emergency Manager. The Recovery Manager will report to the Emergency Operations Facility and begin to prepare for the transition to Recovery, as necessary. If Recovery is imminent, the Recovery Manager will establish a recovery or post-accident outage organization following the site’s plant event recovery protocols.

5.6 Augmentation of Plant and EOF Emergency Organizations

5.6.1 Offsite Support Response

The emergency response plan for Prairie Island NGP is designed to be initially implemented independent of any offsite support. However, the onsite effort will be augmented with offsite support resources as described in the MT & PI Offsite Nuclear Emergency Plan.

It is the purpose of the offsite support organization to augment the onsite response effort with offsite support resources as soon as practical and as needed by the Prairie Island Site staff. Such areas of support include: Government Agency Interface, Logistics Support, News Media Interface and Utility Executive Management Interface.

5.6.2 Monticello Radiation Protection Group Support

The Monticello Nuclear Generating Plant is located approximately 100 miles northwest of Prairie Island NGP. The Monticello Radiation Protection and Chemistry Groups are available for supporting the Prairie Island Radiation Protection Group with personnel and equipment during any emergency condition at Prairie Island.

5.6.3 Westinghouse Support

Westinghouse emergency assistance is available on a twenty-four hour per day, seven day per week basis. Westinghouse will activate all appropriate features of the Westinghouse Emergency Response Plan to support the plant needs. When activated, the Westinghouse Emergency Response Plan becomes a functioning organization, comprised of individuals with unique technical, managerial and communication skills and experience, necessary to:

A. Make an early assessment of the situation

B. Provide early assistance to the utility

C. Mobilize appropriate Westinghouse critical skills and functions

D. Initiate timely, accurate communications to involved and interested parties

A Site Response Team may be dispatched to the site to obtain a first hand assessment of actual conditions and establish communications from the site to the Westinghouse response center, as deemed necessary.

5.6.4 Local Support Services

A. Fire Fighting

The Red Wing Fire Department will provide assistance in the event of a fire occurring at the plant. The duties and responsibilities of the Plant Fire Brigade, insuring complete coordination with the Fire Department, are covered in the Operations Manual, Section F5, Fire Fighting.

The Red Wing Fire Department will be the lead fire and Emergency Medical Service (EMS) agency for all emergencies. The Red Wing Fire Department maintains mutual aid agreements with other area ambulance and fire departments as specified in the City of Red Wing/Goodhue County Emergency Response Plan. These agreements provide that the City may call upon other resources to assist in responding to an emergency, including a Hostile Action Based (HAB) event. For a HAB event, Red Wing Fire Department will deploy a representative to the Incident Command Post dependent upon type, location, and scope of the incident, once scene safety is established.

The Red Wing Fire Department has various firefighting apparatus and water pumping equipment available for use. All Red Wing Fire Department apparatus can perform both fire fighting tasks, including rescue, and non-fire fighting tasks, including spraying to contain radiological releases and pumping water into the plant for refilling and cooling purposes. In all cases, such operations can begin once the radiological and security threats are mitigated to insure the safety of both plant personnel and fire fighters.

B. Hospital and Medical Support

Medical support and treatment for non-radiological injuries is provided by the Mayo Clinic Health System, both of which are located in Red Wing, Minnesota. Radiological related injuries are treated at the medical center which is the primary treatment facility. Emergency plans have been prepared, and training of medical center personnel is accomplished on an annual basis.

Regions Hospital in St. Paul, Minnesota is designated as the definitive care center for Prairie Island Nuclear Generating Plant. Regions Hospital may be used for radiation casualties, severe burn casualties, and other non-radiation injuries with use of an appropriate medical air transport service.

C. Ambulance Service

The Red Wing Ambulance Service will provide service to the Prairie Island Nuclear Generating Plant. Training and participation in drills ensures that personnel involved in the transportation of radiation victims are knowledgeable in use of proper procedures and handling methods. Procedures are covered in the Operations Manual, Section F4, Medical Support and Casualty Care.

D. Local Law Enforcement

For a Notification of Unusual Event (NUE) Security Condition and an Alert and Site Area Emergency Hostile Action Based (HAB) event at PINGP, the City of Red Wing Police Department is the lead law enforcement agency. For a HAB event, the Red Wing Police Department will set up an Incident Command Post (ICP) near the site. The pre-designated ICP locations have been identified; however, selection will depend on the incident. The City of Red Wing Police Department maintains the list of potential ICP sites and will be responsible for designating the site during a response and telling the other agencies responding to the location. Unified Command should be established and includes city, county, state, federal and utility expertise. Communication will be established between the Incident Commander and plant security and operations as soon as possible.

The Red Wing Police Department has the ability to request additional response resources from neighboring agencies (i.e. the primary source of additional resources will be the Goodhue County Sheriff's Office with the ability to request assistance from other neighboring agencies as necessary) to assist them in response to any Prairie Island contingency situation, including a HAB event.

The initial hostile action response goals are; maintain vital plant systems to prevent a release of radioactive materials, protection of on-site workforce, neutralizing the adversaries, and restoring plant operating conditions. Tactical operational priorities supported by Law Enforcemment include; securing a perimeter around the site, containment of vital areas, sweep and securing of vital areas, safe movement of critical workers on the site, neutralizing adversaries, protection/evacuation of the on-site workforce, and sweep of protected area and owner controlled area.

The Incident Command Post will support tracking resources and personnel at or near the site and the City of Red Wing/Goodhue County Emergency Operation Center (EOC) will support tracking resources and personnel off-site in accordance with the Radiological Emergency Plan. In the event that NSPM has declared a General Emergency as defined in the City/County Plan, the Goodhue County Sheriff's Office shall assume operational control over all emergency operations.

5.7 Coordination with Governmental Response Organizations

5.7.1 Minnesota Division of Homeland Security and Emergency Management (HSEM)

The Minnesota Division of Homeland Security and Emergency Management has the responsibility for notification and coordination of Minnesota State Agencies in the event of a major emergency at Prairie Island.

The MN HSEM is notified by Prairie Island NGP. In the event of an emergency situation at the plant, the MN duty officer will immediately call the MN Department of Health, the Governor's Authorized Representative and other state agencies with emergency assignments to coordinate the implementation of any emergency procedures. The state agencies responsible for emergency procedures have established a system of twenty-four hour communications.

5.7.2 Minnesota Department of Health (MDH)

The Minnesota Department of Health (MDH) is responsible for providing radiological expertise in the State Emergency Operations Center in conjunction with the MN HSEM.

The Minnesota Department of Health will interpret data and participate in recommending protective actions to the Governor's Authorized Representative.

5.7.3 Wisconsin Emergency Management

The Wisconsin Emergency Management (WEM), has the responsibility for notification and coordination of Wisconsin state agencies in the event of a major emergency at Prairie Island NGP.

In the event of an emergency situation at the plant, Prairie Island NGP will notify the WEM duty officer who will notify the Wisconsin Department of Health Services (Radiation Protection Section) and other state agencies with emergency assignments, to coordinate the implementation of any emergency procedures. The state agencies responsible for emergency procedures have established a system of twenty-four hour communications.

5.7.4 Wisconsin Department of Health Services (DHS)

The Wisconsin Department of Health Services (DHS) is responsible to prevent exposure to ionizing radiation in amounts which are detrimental to health according to nationally accepted standards.

The Wisconsin DHS, Radiation Protection Section, is responsible for coordination of radiation response activities in the State of Wisconsin. In the event of an emergency at Prairie Island NGP, DHS, Radiation Protection Section will be concerned with monitoring the air and water about the plant to assure that the public is not exposed to levels of radioactive pollutants potentially detrimental to public health. DHS’s facilities are located in Madison, Wisconsin.

5.7.5 Goodhue, Dakota and Pierce County Sheriffs

The Sheriff's Departments will notify all necessary local emergency response groups in the event of an accident. The Sheriff is responsible for protection of the general public and can provide personnel and equipment for evacuation, relocation and isolation.

Goodhue County and the Sheriff also has agreements in place to request additional response resources from neighboring agencies, including resources needed to respond to a HAB event. For a HAB event, the Red Wing Police Department will set up an Incident Command Post (ICP) near the site. The Goodhue County Sheriff's Office Tactical Response Team will be the lead tactical response operations group coordinator and coordinate the tactical law enforcement response with Command. Goodhue County Sheriff's Office can request tactical team resources as needed from: Minnesota State Patrol Special Response Team, Dakota County ERT, FBI SWAT and Washington County ERT.

5.7.6 Goodhue, Dakota, Pierce County and City of Red Wing Emergency Management

The Goodhue, Dakota, Pierce County and City of Red Wing Emergency Management Organizations have the responsibility for notification and providing direction to residents in the event of a major emergency that affects their respective area of responsibility.

5.7.7 Prairie Island Indian Community

The Prairie Island Indian Community has an Emergency Operations Plan that includes the description of tribal responsibilities during a nuclear plant declared event. The Prairie Island Nuclear Generation Plant conducts emergency notifications to the Treasure Island Casino security dispatch center who, in turn, notifies appropriate members of the Prairie Island Indian Community and their organization.

5.7.8 Minnesota State Patrol

The Minnesota (MN) State Patrol has the responsibility to protect the general public by providing personnel and equipment to re-route traffic in the event of an emergency situation. Plans have been made for re-routing federal and state highways. Signs and equipment required for re-routing will be stored in the areas where they would be needed to facilitate highway closings. The MN Department of Transportation would be notified by the MN State Patrol to erect the signs.

5.7.9 Minnesota Department of Transportation

The MN Department of Transportation will assist the MN State Patrol in blocking and re-routing traffic around the plant site. In addition to the necessary personnel; vehicles, signals, and barriers for setting up and maintaining detour routes are available.

5.7.10 Canadian Pacific Railway-CP Railway (Soo Line

In an emergency situation, CP Rail will make every reasonable effort to expedite unblocking the road/railroad crossing near Prairie Island NGP. The dispatcher will also provide routing assistance during an emergency at Prairie Island NGP.

5.7.11 Burlington Northern Santa Fe (BNSF) Railway

The dispatcher will provide routing assistance during an emergency at Prairie Island NGP as per the Minnesota State emergency operations plan.

5.7.12 Department of the Army, Corps of Engineers, Lock & Dam #3

The Corps of Engineers at Lock & Dam #3 will be notified by the Minnesota Duty Officer of an emergency at Prairie Island NGP. The Lock and Dam personnel will notify all tows within radio range of impending evacuations and assist in evacuation of personnel at the Lock and Dam.

NOTE:	A complete description of response capabilities, organizational resources, activation plans, designations of emergency operations centers and letters of agreement are available in Minnesota and Wisconsin's state emergency operations plans.

5.7.13 Nuclear Regulatory Commission (NRC)

The basic responsibilities of the NRC are to monitor, assess, and, if necessary, direct the utility to take actions to protect the health and safety of the public. For a radiological incident at a commercial power plant, the NRC is the Lead Federal Agency (LFA). The LFA is responsible for coordinating all Federal onscene actions. The NRC will coordinate Federal assistance to States and local organizations.

A principal role of the LFA is to assist the State in interpretation and analysis of technical information as a basis for making decisions about protective actions. This assistance will begin early in an incident from the NRC Operations Center in Rockville, MD, and later, from the utility's emergency operations facility on scene. The NRC is an independent reviewer of the actions the utility is taking to correct the initiating and related problems. The NRC will assess actual or potential offsite impacts as well, and will make an independent evaluation of Protective Action Recommendations, if necessary. As the LFA, the NRC has the responsibility for coordinating the release of Federal information to the media and others. The NRC will conduct most public information activities from the utility's Joint Information Center (JIC). The NRC also will keep the White House and Congress informed on all aspects of the event.

The NRC is responsible for giving the best possible advice at a given time to the States and will not limit its involvement to presenting a series of options.

The NRC also administers the Price-Anderson Act to ensure that the public that is affected by the event has adequate financial assistance to address most emergency needs.

5.7.14 Department of Energy (DOE)

Among its responsibilities as a support agency, DOE will coordinate the offsite radiological monitoring and assessment for the Lead Federal Agency (LFA) and the State during the initial phases of the emergency. It will maintain a common set of offsite radiological data and provide an appropriate interpretation of the data to the LFA and the State. DOE will manage the Federal Radiological Monitoring and Assessment Center (FRMAC), which is a multi-agency facility. DOE will conduct environmental monitoring, including air, ground, and water.

Their immediate objective is to rapidly dispatch a Radiological Assistant Program (RAP) Team to the scene to assess the hazard to the public and make recommendations to the authorities for the protection of the public. The Planning Chief in the State EOC is the designated Minnesota authority to request RAP assistance, as stated in the Minnesota state plan, and the Wisconsin DH, Radiation Section, is the designated Wisconsin authority to request RAP assistance for Wisconsin, as stated in the Wisconsin state plan.

5.7.15 Institute Of Nuclear Power Operations (INPO)

INPO will coordinate requests from other utility INPO members and participants. They will notify NEI and EPRI of events, maintain an emergency resource capability and information on industry assistance capabilities coordinate the delivery of persons and materials under its Nuclear Power Plant and Transportation Agreements, and provide member communications to facilitate the flow of technical information about the emergency.

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6.0 EMERGENCY MEASURES

This section will describe the activation of the Emergency Organization. Various detailed and specific emergency measures that will be taken by the plant staff are further delineated in the plant's emergency plan implementing procedures.

6.1 Activation of Emergency Organization

6.1.1 Activation of Plant and EOF Organizations

The Shift Manager will be responsible for activating any part of the emergency organization. During the normal work week, the plant and training center public address systems will be used to activate the organizations. During the off-shift hours, activation of the emergency organizations will be accomplished using the ERO (Emergency Response Organization) Pager Network and the ERO Auto Dial System. Personal pagers are carried by the following personnel who are considered members of the emergency organization:

A. Radiation Survey Team Members

B. Plant Operating Review Committee Members

C. Maintenance Supervisors (Mechanical and Electrical)

D. I&C Supervisors

E. Designated Engineers & Technical Personnel

The ERO Pager Network is a personal pager system activated by a phone call. Upon receipt of a notification, it will be the responsibility of the supervisors to contact any additional personnel in their respective groups which may be required to report to the plant site, to staff the Technical Support Center, Operational Support Center and Emergency Operations Facility or to initiate offsite monitoring.

The ERO Auto Dial System is an automatic dialing telephone network with multiple outgoing telephone lines. When activated, it will call and deliver an emergency message to the Plant and EOF Organization's home telephones.

The ERO Auto Dial System and ERO Pager Network are two notification system(s) used to activate the onsite emergency organization. One system is the backup of the other system. Both will be activated for ERO notification. Telephone numbers of all key emergency organization personnel are published in the MT & PI Nuclear Emergency Preparedness Telephone Directory.

If the event involves a credible security threat, EOF staff may be directed to staff the Backup EOF. In this case, the onsite ERO may be directed to the Red Wing Service Center until it is safe to staff the onsite OSC and TSC. The Red Wing Service Center is to be used as the Alternative Facility during a security threat or event. The RWSC has communication links with the Control Room, EOF, and Security.

6.1.2 Notification Scheme

When an abnormal condition is identified by the Operating Staff/Shift Supervisor, the Shift Supervisor will contact the Shift Manager and the Shift Emergency Communicator. An assessment of the safety significance will be performed, and a determination of the emergency classification will be made using the plant’s emergency plan implementing procedures.

Upon declaring an emergency condition, the Shift Manager will activate portions of the Emergency Plan as appropriate to respond to the declared emergency. During a Notification of Unusual Event, the Emergency Director position usually will not be staffed and the Shift Manager **SHALL** designate the Shift Emergency Communicator or other qualified communicator to make the necessary notifications of offsite state and local authorities. The Emergency Director position will be staffed during an Alert, a Site Area Emergency or General Emergency. The Shift Manager will assume the role as Emergency Director until relieved by the individual designated to relieve him.

The Shift Manager/Emergency Director, will designate the Emergency Communicator or qualified designee to make notification calls to the following individuals or agencies, as detailed in the plant's implementing procedures.

A. State of Minnesota HSEM

B. State of Wisconsin Emergency Management

C. Local Authorities (Wisconsin & Minnesota)

1. Dakota County Sheriff

2. Pierce County Sheriff

3. Goodhue County Sheriff

D. Prairie Island Indian Community Representatives via Treasure Island Casino Security Dispatch Center

E. Plant Manager (designated Emergency Director)

F. Emergency Manager

G. Electric Utility System Operations Dispatcher

H. NRC Resident Inspectors

A more detailed call list of agencies and individuals, listing phone numbers, is included in the implementing procedures.

The Shift Manager/Emergency Director will ensure that the NRC Duty Officer is notified of the emergency by a qualified individual within one (1) hour of emergency declaration.

Eventually the Emergency Manager, in the Emergency Operations Facility (EOF), will relieve the Emergency Director of offsite communications and protective action recommendations. At that time offsite notification calls will be initiated by the EOF. The Prairie Island Onsite/Offsite Emergency Organization Interface is shown in Figure 3.

6.1.3 Communicators

It is the responsibility of all individuals in the emergency organization to ensure that any information transmitted or received over any communication channel is formal, clear, and concise so that there will be no misunderstanding.

Dedicated communicators will be assigned at each emergency operations center assuring a uniform transfer of information between segments of the onsite and offsite emergency response organizations. Initially, this responsibility rests with the Shift Emergency Communicator or qualified designee located in the Technical Support Center and subsequently with all backup communicators assigned these responsibilities.

Emergency Response Facilities such as the Technical Support Center, Operational Support Center, Control Room, Assembly Area and EOF will have dedicated communicators. Communicators will be assigned to specific communication duties, for example:

A. ENS Hotline – licensed operator or designee

B. HPN Hotline – Radiation Protection personnel when requested by the NRC following facility activation.

C. NRC Security Bridge – Security personnel when requested by the NRC following facility activation.

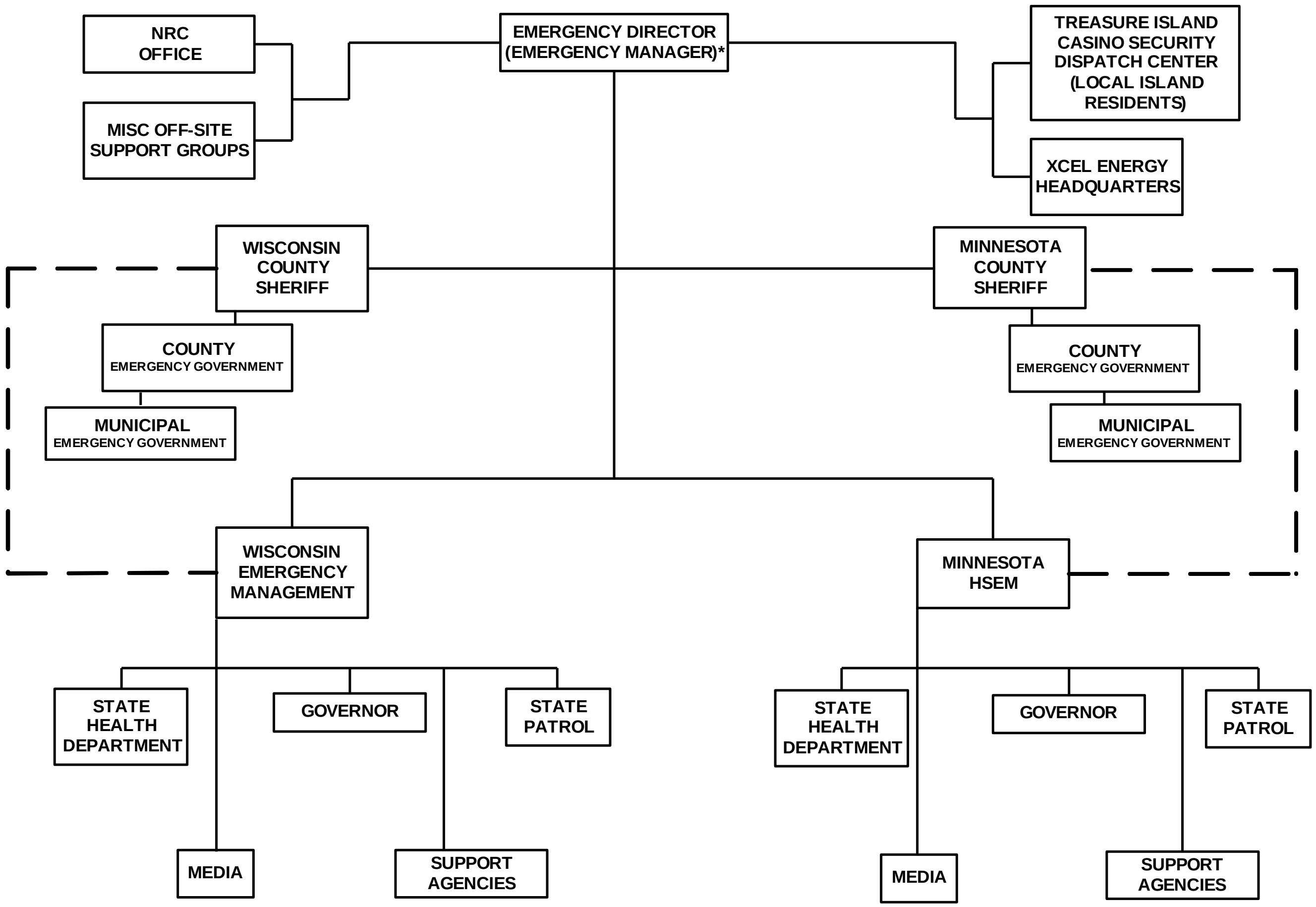
D. Offsite State and Local Agency Notifications – Shift Emergency Communicator and Emergency Communicators

E. Survey Teams – Radiological Emergency Coordinator or Radiation Protection Support Supervisor and/or designee

F. Emergency Operating Centers – Operating Center Coordinators and/or designees

G. Others as deemed necessary

Figure 3 Prairie Island Onsite/Offsite Emergency Organization Interface



* EMERGENCY MANAGER assumes offsite responsibilities when the EOF Organization is activated .

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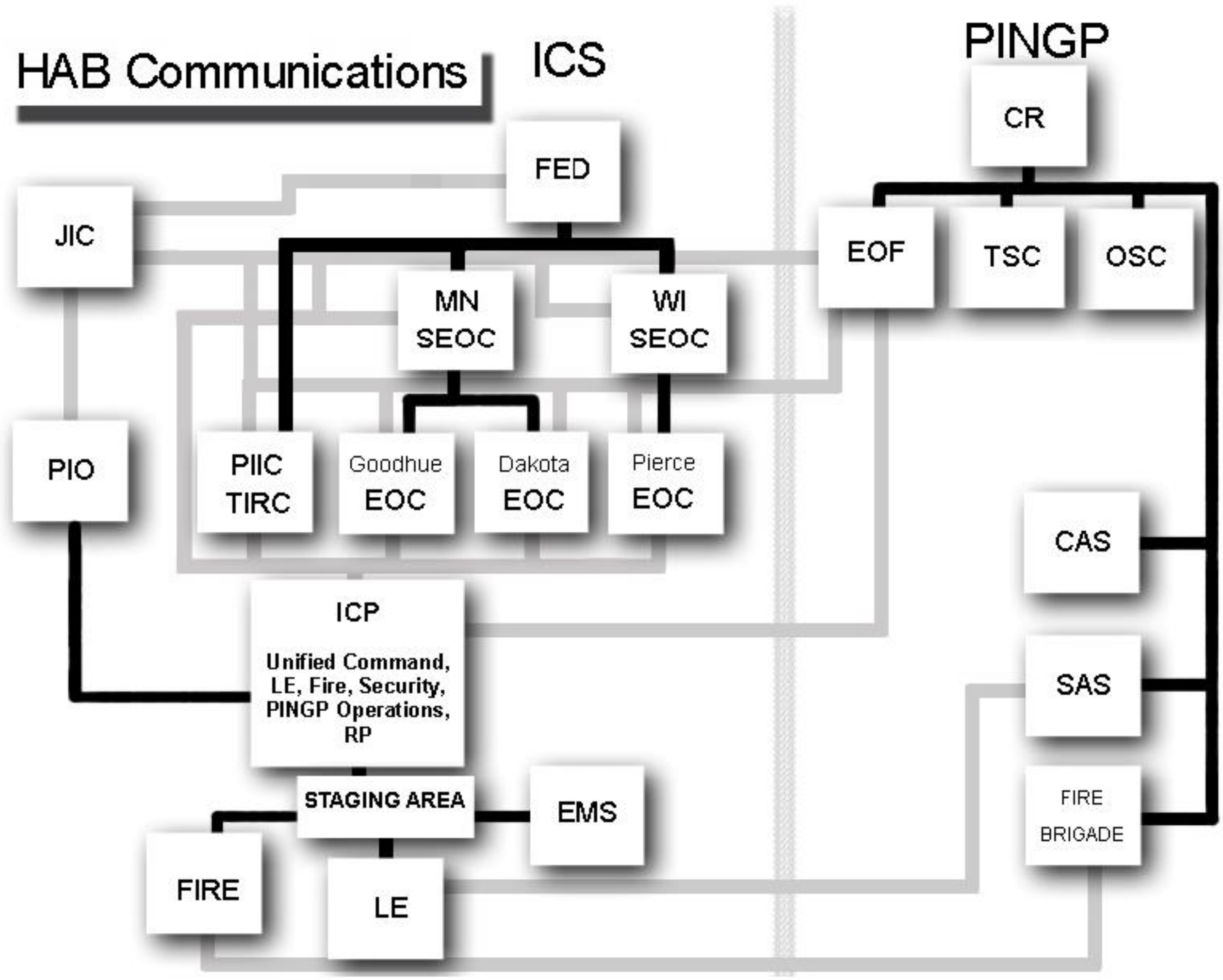
Figure 4 HAB Communications



Primary offsite authorities provide a 24 hour per day manning of communication links, as follows:

A. Wisconsin authorities

1. State of Wisc. (WEM) – State Patrol District 1 Dispatcher

2. Pierce County – Pierce County Sheriff's Dispatcher

B. Minnesota authorities

1. State of Minnesota – Minnesota Duty Officer (MDO)

2. Goodhue County – Goodhue County Sheriff's Dispatcher

3. Dakota County – Dakota County Sheriff's Dispatcher

C. Tribal Authorities – Treasure Island Security Dispatch

6.1.4 Authentication of Emergency Communications

Communication, for the purpose of notifying offsite agencies that an emergency condition exists, **SHALL** be authenticated before offsite agency action is initiated. The authentications will be accomplished in accordance with the offsite agencies specific emergency plans.

6.2 Record Keeping

It is the responsibility of all personnel involved in the emergency organization to ensure that accurate and complete records are maintained throughout the emergency situation. Emergency records may serve the following purposes:

6.2.1 Official documentation used to reconstruct the emergency for critique or analysis;

6.2.2 Check to ensure that necessary actions are completed during the course of an emergency;

6.2.3 Information and data collection during an emergency; and

6.2.4 Documentation of actions for legal purposes.

All activities performed by the operations staff **SHALL** be logged in the applicable reactor log. All other information and activities **SHALL** be maintained by the Emergency Director, Emergency Manager and various coordinators (e.g., individuals in charge of various emergency operating centers, radiation survey teams, etc.) for permanent plant records.

6.3 Summary of Site Response Actions

Summarized below are the actions required by the site staff for each of the four emergency classifications. For each class of emergency, appropriate state, local, and tribal authorities will be notified. Depending on the emergency level classification, they will activate the segment(s) of their emergency organizations, according to their individual plans and based on the information received in the notification.

NOTIFICATION OF UNUSUAL EVENT

1. Promptly inform offsite authorities of unusual event status and the reason for the Unusual Event as soon as discovered.

2. Augment on-shift resources as needed.

3. Assess and respond to Unusual Event.

4. Terminate by contacting offsite authorities

   or

5. Escalate to a more severe class.

ALERT

1. Promptly inform offsite authorities of Alert status and reason for Alert as soon as discovered.

2. Augment resources by activating onsite Technical Support Center (TSC) and onsite Operational Support Center (OSC). The Emergency Operations Facility (EOF) and key offsite emergency organization personnel will be activated.

3. Assess and respond to the Alert condition.

4. Dispatch onsite and offsite survey teams and associated communications.

5. Provide periodic plant status updates to offsite authorities.

6. Provide periodic meteorological assessments to offsite authorities and, if any releases are occurring, dose estimates for actual releases.

7. Terminate by contacting offsite authorities.

   or

8. Escalate to a more severe class.

SITE AREA EMERGENCY

1. Promptly inform offsite authorities of Site Area Emergency status and reason for emergency as soon as discovered.
2. Augment resources by activating onsite Technical Support Center (TSC), onsite Operational Support Center (OSC) and Emergency Operations Facility (EOF).
3. Assess and respond to the Site Area Emergency.
4. If radiological or environmental conditions permit, evacuate onsite, nonessential personnel.
5. Dispatch onsite and offsite survey teams and associated communications.
6. Provide a dedicated individual for plant status updates to offsite authorities.
7. Make senior technical and management staff onsite available for consultation with NRC and State on a periodic basis.
8. Provide meteorological and dose estimates to offsite authorities for actual release via a dedicated individual.
9. Provide release and dose projections based on available plant condition information and foreseeable contingencies.
10. Terminate emergency class by contacting offsite authorities and initiate recovery phase.

    or

11. Escalate to General Emergency class.

GENERAL EMERGENCY

1. Promptly inform offsite authorities of General Emergency status, appropriate offsite protective action recommendations and reason for emergency as soon as discovered.
2. Augment resources by activating onsite Technical Support Center (TSC), onsite Operational Support Center (OSC) and Emergency Operations Facility (EOF).
3. Assess and respond to General Emergency.
4. If radiological or environmental conditions permit, evacuate onsite, nonessential personnel.
5. Dispatch onsite and offsite survey teams and associated communications.
6. Provide a dedicated individual for plant status updates to offsite authorities.
7. Make senior technical and management staff onsite available for consultation with NRC and State on a periodic basis.
8. Provide meteorological and dose estimates to offsite authorities for actual releases via a dedicated individual.
9. Provide release and dose projections based on available plant condition information and foreseeable contingencies.
10. Terminate emergency class by contacting offsite authorities and initiate recovery phase.

6.4 Assessment Actions

6.4.1 Dose Projections

Dose projections may be performed by using the standard dose projection program RASCAL (Radiological Assessment System for Consequence Analysis). Radioactive effluent release and meteorological data is procured from the Emergency Response Computer System (ERCS) and entered into RASCAL for real time dose assessments during inadvertent release of radioactive materials. The RASCAL program may be run from terminals that are located in the Control Room, TSC, EOF, and Backup EOF.

Meteorological data is stored and processed in the ERCS. The onsite 60 meter meteorological tower supplies the following:

A. Wind speed (10 and 60 meters)

B. Wind direction (10 and 60 meters)

C. Ambient Temperature

D. DT (between 10 and 60 meters)

E. Rainfall

A 22 meter backup meteorological tower is located near the EOF, and supplies the following:

A. Wind speed (22 meters)

B. Wind direction (22 meters)

Redundant instrumentation is provided on the onsite 60 meter meteorological tower, and may be designated as primary and secondary sensors. The 22 meter backup tower provides a set of tertiary sensors. The ERCS continuously collects the meteorological data. Meteorological data from all three sets of instruments are displayed simultaneously as well as the calculated stability class (derived from the temperature readings). If all met data is unavailable, manual entry of met data may be made for accident calculations.

Surveillances and quality checks are performed on the meteorological tower equipment and data to ensure emergency responders will have access to representative onsite meteorological data. A daily review of a week's trend of meteorological data is performed. The meteorological tower instruments are functionally tested monthly and calibrated at least annually.

Radiological effluent monitor data is also stored and processed in the ERCS. The effluent monitor reading, the calibration conversion factor and the vent flow rate result in a release rate. Effluent concentrations may also be manually entered into the computer if monitor data is not automatically available to the ERCS.

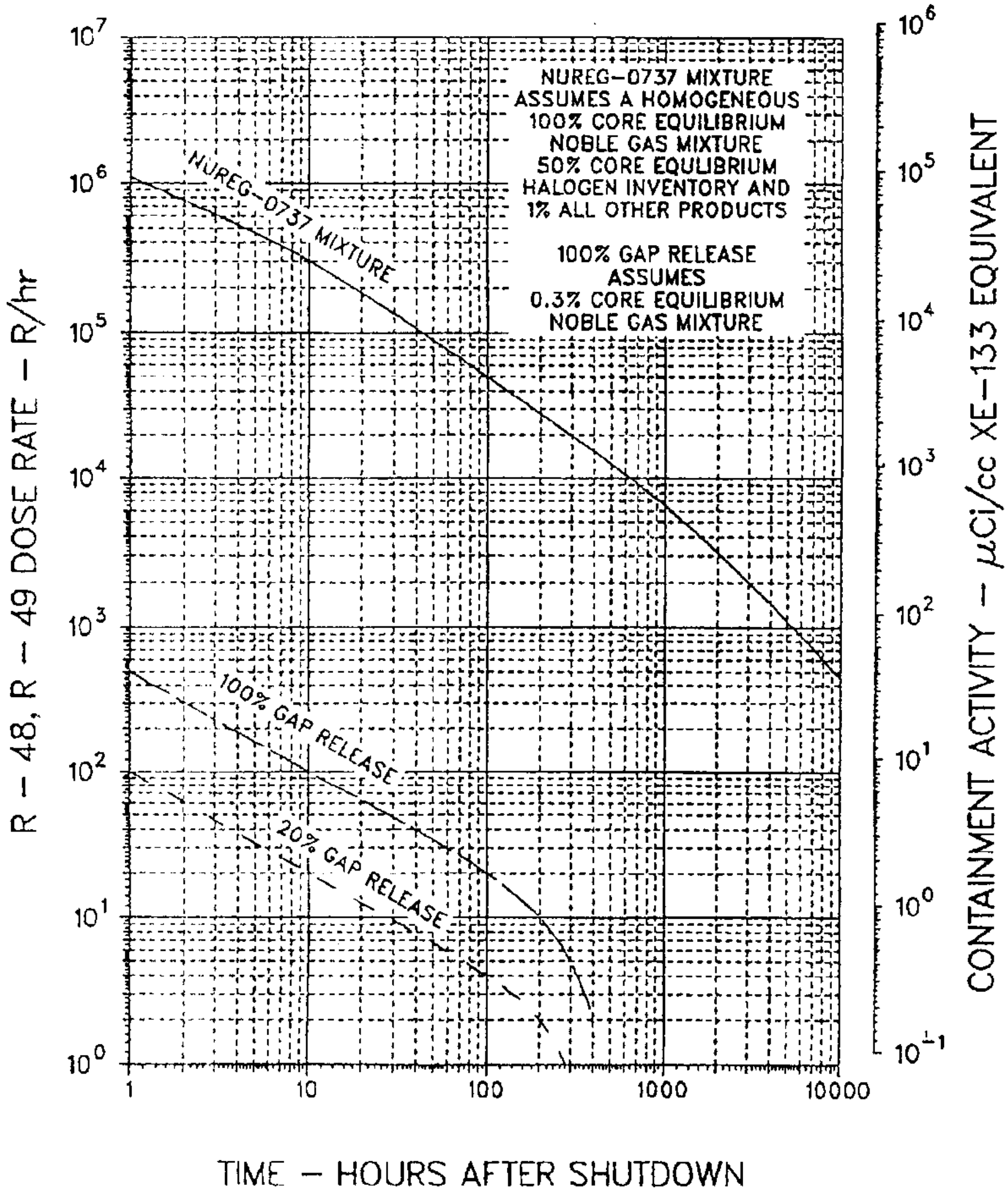
With meteorological and effluent release data available, calculations of offsite radiation dose, air concentration, ground deposition, and external dose rate from the plume can be made. Dose calculations are made for Total Effective Dose Equivalents (TEDE) and Thyroid Committed Dose Equivalents (Thyroid CDE). Results of all calculations can be printed in report format and, in most cases, displayed graphically. Isopleths can be displayed of any or all calculated outputs. Projected calculations take into account values of time of release and duration of release. The isopleth displayed is based on the assumption that the release continues for a predetermined duration time. This gives a display in which the plume overlays the region of potential highest dose.

The dose assessment computer allows quick accident dose calculations to be made, before any results from the Radiation Survey Teams are received. Radiation Survey Team results will be used to verify the dose calculations.

In case of potential release from the containment, the activity available for a release may be obtained from the containment high range dome monitors, as illustrated in Figure 5. The containment dome monitor reading and applicable calibration curve results in an activity available for release, and using an estimated release rate, an offsite dose calculation within the plume exposure pathway may be projected. The activity available in containment may also be obtained directly from sample analysis.

The containment dome monitors are also used as indicators for relative amounts of core damage, as illustrated in Figure 5. The indicated radiation levels in the containment gives an estimate of the gaseous radioactive concentrations in containment. Using the time after shutdown and the radiation levels, an estimate of the relative amount of core damage may be made. This must be used in a confirmatory sense, that is, as backup to other measurements of fission product release and other indicators.

Figure 5 Containment Dose Rate Versus Time



The capability for remote interrogation of the meteorological data will be provided to NRC by either the Emergency Response Data System (ERDS) or direct telephone access to the individual responsible for making offsite dose projections. Implementing procedures will detail this activity.

A hand calculation methodology for offsite dose calculations is available in case of computer system and/or meteorological system failure. Additionally, meteorological data may be obtained from local offsite locations. Atmospheric stability class and weather forecast information is available from the National Weather Service Twin Cities.

The capability to estimate the total offsite population dose (manrem) received during a release is available. The offsite dose assessment computer will supply the projected dose rates or doses at selected distances from the plant. Radiation Survey Team results may also be used to determine the offsite dose rates. Population distribution charts comprised of the geopolitical subareas are available. The Radiological Emergency Coordinator in the TSC or the Radiological Protection Support Supervisor in the EOF may determine the applicable dose rates in the geopolitical subarea and multiply dose rate times the exposure time, times the population in the geopolitical subarea of interest, thereby calculating an estimated total population dose.

The Emergency Director **SHALL** ensure that radiological information (both actual and potential) and recommendations for protective actions are transmitted to the offsite authorities. Upon activation of the EOF Organization, the responsibility for offsite accident assessment is transferred to the EOF. The EOF will serve as a base of operations for all site environmental surveillance, receipt and analysis of all field monitoring data, offsite dose projection and recommendations for offsite protective actions.

6.4.2 Radiological Surveys

The Radiation Protection Group **SHALL** be responsible for all radiological surveys and personnel monitoring both onsite and offsite. The Emergency Director has the responsibility for directing all radiation safety during the emergency.

The Radiation Protection Specialists may be divided into two emergency Radiation Survey Teams. The teams are assigned offsite duties such as radiation surveys, air samples, or liquid sampling. The two offsite survey teams will conduct a search for the plume and obtain dose rates, and iodine, particulate or gaseous samples at pre-designated sample locations. Plume exposure pathway maps with pre-designated sample locations are contained in the emergency survey kits. Additional duties onsite such as radiation surveys, sampling (airborne or liquid) and sample analysis using the equipment available onsite and/or the EOF Count Room facility are completed by other augmented personnel. Silver zeolite adsorbers are used to collect airborne iodine samples, both onsite and offsite. Silver zeolite adsorbers eliminate the problem of entrapped noble gases on the iodine adsorber, allowing a much lower detection sensitivity. Iodine samples may be analyzed in the EOF Counting Room.

The Radiation Survey Teams are activated via the ERO Auto Dial System and/or the ERO Pager Network or the telephone system. If the emergency occurs during normal working hours, the teams will be activated and respond within 10 minutes. If the emergency occurs during off hours, the first team will be activated and respond within sixty (60) minutes and the second team within ninety (90) minutes. Designated Emergency Lockers contain emergency survey kits, which include portable instruments, battery operated air samplers, liquid sampling equipment, and communication equipment.

6.5 Corrective Actions

Certain actions may be taken by the Prairie Island staff during an emergency which may minimize the severity of the accident and lessen the amount of offsite releases. These actions are outlined in the various standing plant abnormal operating, emergency operating, and plant safety procedures.

Repair and Damage Control is the responsibility of the Emergency Director and Shift Supervisors. During the onset of the emergency, plant operators are responsible for minor damage repair and control. Upon activation of the Plant Emergency Organization, equipment repair activities are the responsibility of the Maintenance Group, the I&C Group, the Electrical Group, and the Operations Group depending upon the extent and type of damage. Repair and damage control on radioactive or contaminated systems will be monitored by the Radiation Protection Group.

The Fire Brigade is composed of personnel in accordance with NRC requirements and is directed by a Fire Brigade Chief. Backup support is from the Red Wing Fire Department. All onsite Fire Brigade members are trained in the use of onsite fire fighting equipment and in proper fire fighting procedures. The Fire Brigade will be placed in action under the direction of the Brigade Chief.

6.6 Protective Actions

6.6.1 Evacuation and Sheltering

In the course of an emergency at Prairie Island NGP when there is an actual or potential release of radioactive material to the environs in excess of normal operating levels, the Emergency Director **SHALL** be responsible to ensure that an assessment is made of the projected doses to persons onsite and offsite. Upon activation of the EOF, the Emergency Manager **SHALL** be responsible for ensuring that all assessments are made of the projected doses to the offsite population.

The Protective Action Guides (PAG's), promulgated by the EPA, set dose guides for the offsite population. The Emergency Director also has the responsibility to ensure that protective actions are also taken to maintain exposure to onsite personnel within the PAG's. The Emergency Director or the Emergency Manager when the EOF is activated **SHALL** be responsible to recommend to the state and local authorities any protective actions for the offsite population whether the protective actions be based on predetermined Emergency Action Levels (EALs) or projected offsite dose assessment.

Plant Emergency Organization personnel fall into the category of "Emergency Workers" to which higher PAG's apply. The Emergency Director has the responsibility of maintaining doses within these PAG's.

A. Plant Site

The primary protective measure for non-essential onsite personnel during a Site Area or General Emergency and possibly during an Alert, is evacuation to a suitable assembly area where the personnel can be monitored for contamination. The Emergency Director or Shift Manager, prior to ordering an evacuation **SHALL** determine the habitability of the assembly area (wind direction, magnitude of release, etc.).

If the normal onsite assembly area is determined to be uninhabitable, the Emergency Director will select a location farther from the plant site and designate the route to this location.

The Control Room operator will sound the evacuation alarm and announce the designated assembly area. If a location offsite is selected, the traffic route and area **SHALL** be announced.

Once non-essential personnel are accounted for and monitored for contamination, they may be released from the assembly area.

The evacuation routes from the assembly areas are limited to two directions: County 18 to Etter to Hwy 316 or County 18 to Hwy 61. High water conditions may make the Etter route unusable, leaving only the County 18 to Hwy 61 route available. Prairie Island NGP vehicles and personal cars will be used to transport all personnel.

If conditions (meteorological or radiological) make land routes unavailable, evacuation by alternate means, (e.g., aircraft or watercraft) may be a viable alternative.

All non-essential personnel **SHALL** evacuate to the designated assembly area. The plant security force will assist in the evacuation by directing people to the proper assembly area. The Security Force **SHALL** direct employees to badge out of the Protected Area while exiting the Protected Area. The Security Force will perform an immediate check of the Protected Area to ensure that all personnel did indeed hear the evacuation alarm. The Security Force will perform a check of the Owner Controlled Area and warn all personnel of the evacuation in progress.

Radiation Survey Team Members, extra on-shift operators, group managers, Maintenance Supervisors, I&C Supervisors, Lead Maintenance Personnel and Station Electricians **SHALL** report to the Operational Support Center or the Technical Support Center, as applicable. Plant staff without emergency assignments **SHALL** evacuate to the designated assembly area. NRC Resident Inspector(s) may proceed to the Technical Support Center or Control Room.

Designated individuals **SHALL** complete an accountability check of personnel remaining within the Protected Area by verifying a list of personnel remaining in the Protected Area. The Emergency Director accepts responsibility for solving any discrepancies found during the accountability. The Emergency Director **SHALL** direct the necessary follow-up actions.

The Radiation Protection Group or qualified personnel **SHALL** monitor personnel at the assembly area for contamination, and any exposure determinations **SHALL** be completed, as conditions warrant. The emergency locker contains material necessary for decontamination of personnel under Radiation Protection Group supervision.

The assembly area **SHALL** remain in contact with the Emergency Director or designee via the telephone system or portable radio supplied in the emergency locker. The individual assigned as the Coordinator at the assembly area will be the contact point for all personnel.

The Emergency Director **SHALL** release non-essential personnel for departure from the site when conditions allow or demand this action. The Emergency Director will designate the proper traffic routes to follow during the departure.

Onsite Protective Actions designed for protection of onsite personnel as described above may be inappropriate for a Hostile Action Event. Alternate actions as described in NSIR/DRP-ISG-01 Section IV.F have been developed and proceduralized.

B. Offsite Areas

The primary protective actions for the offsite population are sheltering or evacuation. The Emergency Director **SHALL** recommend the necessary protective actions to offsite authorities based on predetermined protective actions for a General Emergency Classification or results of offsite dose assessment. Upon activation of the EOF, the Emergency Manager **SHALL** be responsible for recommending protective actions for the offsite population. If protective actions are warranted prior to augmentation of state emergency response organizations, the Emergency Director **SHALL** recommend directly to county and tribal authorities the necessary protective actions. In both cases, total responsibility for carrying out the protective actions rests with offsite authorities. Prairie Island NGP **SHALL** make the recommendations and supply the required dose assessments.

C. Protective Action Guides (PAG's)

Table 2 and Table 3 provide guidelines and action levels to be used in the formulation of protective action recommendations for the offsite population and plant personnel.

The specific protective actions carried out by the offsite authorities are contained in their respective emergency plans.

D. Evacuation Time Estimates (ETE) – Plume Exposure EPZ.

Time estimates for evacuation of the plume exposure EPZ are referenced in an appendix to the Off-site Nuclear Emergency Plan and in the Plant Emergency Plan Implementing Procedure for making off-site protective action recommendations. PINGP and the States of Minnesota and Wisconsin use the ETE to develop pre-determined protective action recommendations.

6.6.2 Use of Protective Equipment and Supplies

A. Onsite Respiratory Protection and Protective Clothing

Protective clothing or respiratory protection for onsite personnel **SHALL** be as designated by the Radiation Protection Group or the Emergency Director.

Respiratory Protection will be used as necessary to reduce the inhalation of radioactive material. During emergency conditions, it may become impossible to maintain normal respiratory protection guidelines. An internal exposure program, whole body counting and/or bioassay program, **SHALL** be activated to ensure that all internal exposure is determined as assigned to the individual. Respiratory equipment is stored in the OSC and TSC emergency lockers, Unit 1 695′ Turbine Building Chem. Feed Station Area, Fire Brigade equipment room, and Access Control. Access Control is the main storage area for respiratory equipment. The respiratory equipment available is a combination of Self Contained Breathing Apparatus (SCBAs), and full face canister respirators.

Table 2 Initial Protective Action Recommendation During a General Emergency

The following situations require urgent actions by offsite officials. Conditions are based on Control Room indications with no dose projections required. The following protective action recommendations **SHALL** be made within 15 minutes.

Prerequisite: Plant Staff Detects **GENERAL EMERGENCY**

1. If wind is $\geq$ 5 mph:

(1) **IF HAB Event concurrent with GE –** Issue PAR to SHELTER ALL out to 2 Miles, Evacuate Downwind From 2 Miles to 5 Miles and Circle affected Subareas From 2 Miles to 5 Miles.

(2) **IF Rapidly progressing severe accident with all of the following:**

This PAR is the initial after a GE has been declared

AND

There is LOSS of the containment barrier per the Emergency Action Levels

AND

Either of the following:

a. Greater than or equal to Containment High Range Radiation Monitor Potential Loss Threshold (20% Clad Damage) i.e. 1(2) R-48 or 49 reading > 20,000 R/hr

OR

b. An Offsite Dose Estimate indicates greater than PAGs at the site boundary is occurring or is likely to occur in an hour.

Issue PAR to Evacuate ALL out to 2 Miles, Evacuate Downwind From 2 Miles to 10 Miles and Circle affected Subareas From 2 Miles to 10 Miles.

(3) IF Ongoing Rad release > EPA PAGs expected to be < 1 hour - Issue PAR to SHELTER ALL out to 2 Miles, SHELTER Downwind From 2 Miles to 5 Miles, extend SIP 5-10 Mile if PAGS exceeded at 5 Mile and Circle affected Subareas From 2 Miles to 5 Miles.

(4) Issue PAR to Evacuate ALL out to 2 Miles, Evacuate Downwind From 2 Miles to 5 Miles and Circle affected Subareas From 2 Miles out to 5 Miles.

(5) Advise Remainder of Plume EPZ to Monitor EAS Broadcasts.

(6) Continue with Step 2.

Met Data from 22 meter tower OR wind < 5 mph OR unknown?

(1) **<u>IF HAB Event concurrent with GE</u>** – Issue PAR to SHELTER ALL out to 2 Miles, Evacuate ALL From 2 Miles to 5 Miles and Circle ALL Subareas From 2 Miles to 5 Miles.

(2) **<u>IF Rapidly progressing severe accident with all of the following:</u>**

This PAR is the initial after a GE has been declared

AND

There is LOSS of the containment barrier per the Emergency Action Levels

AND

Either of the following:

a. Greater than or equal to Containment High Range Radiation Monitor Potential Loss Threshold (20% Clad Damage) i.e. 1(2) R-48 or 49 reading > 20,000 R/hr

OR

b. A RASCAL Dose Estimate indicates greater than PAGs at the boundary is occurring or is likely to occur in an hour. Issue PAR to Evacuate ALL out to 10 Miles and Circle ALL Subareas From 2 Miles to 10 Miles.

(3) <u>IF Ongoing Rad release > EPA PAGs expected to be < 1 hour</u> - Issue PAR to SHELTER ALL out to 5 Miles and Circle ALL Subareas From 2 Miles to 5 Miles.

(4) Issue PAR to Evacuate ALL out to 5 Miles and Circle ALL Subareas From 2 Miles out to 5 Miles.

(5) Advise Remainder of Plume EPZ to Monitor EAS Broadcasts.

(6) Continue with Step 2.

2. Continue with dose assessment throughout the emergency and revise initial Protective Action Recommendations in accordance with the protective action guidelines in Table 3.

NOTES:	**The protective action recommendations described above are based on NRC Response Technical Manual, RTM-96, Vol. 1, Rev. 5, October 2002 and EPA 400-R-92-001, May 1992.**

Table 3 Recommended Protective Action to Avoid External and Internal Dose from Exposure to a Gaseous Plume

PAGs for Early Phase Projected Doses

Offsite Projected Doses (mrem)	Recommended Protective Actions	Comments
TEDE < 1000 Thyroid CDE < 5000	No recommended protective actions	The states of MN and WI may choose to implement sheltering or precautionary evacuation for the general public at their discretion.
TEDE ≥ 1000 Thyroid CDE ≥ 5000	Evacuate those sectors and distances where the PAG is exceeded. Use 0-2, 0-5 & 0-10 mile distances. Shelter for known impediments to evacuation and/or controlled puff release.	Evacuation should be recommended in absence of local constraints. MN and WI may choose to shelter if evacuation were not immediately possible due to offsite constraints (severe weather, competing disasters or local traffic constraints).

Notes:
1. **TEDE = Total Effective Dose Equivalent, Thyroid CDE = Thyroid Committed Dose Equivalent**
2. **Based on EPA 400-R-92-001, May 1992**
3. **The Skin CDE PAG for evacuation of the general public is 50,000 mrem**
4. **Offsite projected doses include exposure from radioactive plume (external & internal) and 4 day exposure to ground contamination.**
5. **Known impediments to evacuation are conditions which make evacuation of the public impractical. Conditions include inclement weather (ice/snow storms where driving would be dangerous), and known impacts on the ability to execute public evacuations (severe damage to roads/infrastructures, etc.).**
6. **Controlled puff release exists when there is assurance that the release is short term (puff release) and the area near the plant cannot be evacuated before the plume arrives.**

Table 3 Recommended Protective Action to Avoid External and Internal Dose from Exposure to a Gaseous Plume

PAGs for Emergency Workers

TEDE Dose Limit (mrem)	Activity	Condition
5,000	All emergency activities	This dose limit applies when a lower dose is not practicable through application of ALARA practices.
10,000	Protecting valuable property	Lower dose not practicable
25,000	Life saving or protection of large populations	Lower dose not practicable
>25,000	Life saving or protection of large populations	Doses in excess of this dose limit **SHALL** only be on a voluntary basis to persons fully aware of the risks involved.

Notes:
1. **Based on EPA 400-R-92-001, May 1992**
2. **These are doses to non-pregnant adults from external exposure and intake during an emergency.**
3. **Exposures to the lens of the eye should be limited to 3 times the values listed and doses to the skin and/or extremities and any other organ should be limited to 10 times the values listed.**

B. Radioprotective Drug

The use of a stable iodine thyroid blocking agent, Potassium Iodide (KI), for plant staff and personnel assigned to onsite emergency operating centers is recommended in situations where airborne iodine concentrations have or could increase to unacceptable concentrations resulting in thyroid doses greater than 25 Rem (final recommendation by the Food and Drug Administration).

The Radiological Emergency Coordinator **SHALL** recommend the distribution of Potassium Iodide (KI). The Emergency Director **SHALL** then direct the distribution of Potassium Iodide (KI).

The Potassium Iodide (KI) tablets are stored in the TSC Emergency Locker, EOF Emergency Locker, and in the Field Survey Kits. The tablets will be distributed per the applicable implementing procedure.

C. Shielding

All plant personnel, who are required to occupy the emergency operating centers, (i.e., Tech Support Center and the Control Room), are protected from intense radiation fields and high airborne radioactivity levels by shielding and/or emergency air handling equipment.

All reactor coolant system sampling and radiochemical analysis may be completed using the shielded sampling system with reach rods in the hot sample room and a lead brick shielded work area in the hot cell area.

D. Offsite Areas

There are <u>no</u> plans for the distribution of respiratory protective equipment and/or protective clothing for the general public. The distribution of thyroid blocking agents is the responsibility of the offsite officials. All Protective Actions to be taken for the general public are described in the offsite emergency plans.

6.6.3 Contamination Control Measures

A. Onsite Areas

The Emergency Director **SHALL** designate the Radiation Protection Group responsible for controlling or minimizing direct or subsequent internal exposure from radioactive materials deposited on the ground or other surfaces. The Radiation Protection Group **SHALL** be responsible for determining the extent of contamination in controlled and normally uncontrolled areas. During an emergency, guidelines to follow for contamination limits are shown in Table 4.

The Radiation Protection Group with assistance from the Security Force will establish new secondary access control points at the boundaries of the new controlled areas to ensure that all personnel entering the areas are properly badged and clothed.

The Radiation Protection group **SHALL** advise all personnel that contamination levels in some uncontrolled areas may significantly exceed normal levels. Without protective clothing, personnel will have to take precautions to avoid personal contamination. Limits for personal contamination will remain at the normal limits which will minimize the chance of ingestion of radioactive material.

Table 4 Contamination Limits

	LOOSE SURFACE	LARGE AREA WIPE	FIXED
Authorized Limits (Circle One)	**(Beta-Gamma / Alpha) / 100 sq cm**	**(Beta-Gamma) (GM pancake probe)**	**(Beta-Gamma) (GM pancake probe)**
NORMAL	< 100/10 dpm/100 cm^2	< 100 ccpm	< 100 ccpm
ELEVATED	< 1000/20 dpm/100 cm^2	< 100 ccpm	< 100 ccpm
EMERGENCY	< 5000/100 dpm/100 cm^2	< 500 ccpm	< 500 ccpm
Based on Manual of Protective Action Guides and Protective Actions for Nuclear Accidents, EPA 400-R-92-001, May 1992, Table 7-7. Frisker response: 1 mR/hr ≈ 5000 cpm Cs 137.			

Particular attention will be given to radioiodine contamination of the skin. Oxidizing agents, e.g., Beta dyne or Radiac Wash, are available in the decontamination kits to treat iodine skin contamination.

The Radiation Protection Group **SHALL** have the responsibility of controlling all onsite food and water supplies during the emergency. Whenever a plant evacuation takes place involving radiological hazards onsite, all food and water supplies within the evacuation area may be considered contaminated and not for use.

Material decontamination **SHALL** be performed by the Nuclear Plant Service Attendants or designated personnel under supervision of the Radiation Protection Group. Procedures and equipment for material decontamination are listed in the Decontamination Procedures of the Operations Manual, Sections F-2 and D-13, and in the Radiation Protection Manual RPIP's.

Before any water or food can be consumed, the Radiation Protection Group will check and verify that the food itself and the eating surfaces are below the limits of Section F-2 of the Operations Manual (previously recorded). Random samples of food containers may be analyzed via the GEM Detector for low level contamination not detected by other methods.

During the recovery phase, all areas of the plant will be returned to the original low levels of surface contamination prior to their release for unrestricted entry.

B. Offsite Areas

Contamination control in offsite areas is the responsibility of offsite officials with assistance from Prairie Island NGP. Required protective actions are delineated in Protective Actions guides and criteria are listed in the respective state emergency plans.

6.7 Aid to Affected Personnel

The Emergency Director is responsible for the protection of personnel from exposure to radiation and contamination and arranging for treatment of radiologically induced or contaminated injuries. This responsibility may be delegated to the Radiation Protection Group.

6.7.1 Emergency Personnel Exposure

The Prairie Island Radiation Protection Group has the necessary equipment and personnel required to provide continuous capability to control and determine radiation exposures of emergency organization personnel. The equipment consists of the following:

A. portable radiation detection instruments

B. electronic dosimeters

C. high and low range dosimeters

D. DLR's

E. extra high range dosimeters

F. record keeping equipment

Contractor and vendor representatives may also be present to assist in exposure control and augment the Radiation Protection Group capabilities.

In an emergency situation, all onsite personnel, some offsite support personnel and some local governmental emergency response personnel will be issued DLR's and/or SRD's. Exposure records will be maintained for all emergency response personnel issued dosimetry.

During accident situations, higher radiation exposures may be authorized by the Emergency Director in order to protect life and property. The emergency exposure guidelines established are based on the Environmental Protection Agency's "PAGs for Emergency Workers", as listed in Table 3.

Emergency workers (volunteers) may be allowed to exceed the 10CFR20 limits with specific authorization of the Emergency Director when performing activities to protect life and property.

In certain instances, it may be necessary to exceed 25 Rem exposure during lifesaving operations. All personnel involved **SHALL** be on a volunteer basis and will be advised of the effects of acute exposures and reasonable considerations of the relative risks.

NOTE:	**In all circumstances, every effort SHALL be made to keep exposures within the annual limits of 10CFR20 (5 Rem Total Effective Dose Equivalent).**

6.7.2 Decontamination and First Aid

The Emergency Director **SHALL** delegate the responsibility for personnel decontamination to the Radiation Protection Group. Decontamination procedures and contamination limits are spelled out in the Radiation Protection Manual RPIP's and the Radiation Safety and Medical Support Sections of the Operations Manual, which **SHALL** be followed for both normal and emergency situations involving personnel injury and personnel contamination.

The primary decontamination facility is located at access control. Two showers and a double sink are located there. Special decontamination solutions are also available at access control.

When facilities at access control are not available, the assembly area emergency lockers contain equipment for personnel decontamination and personnel monitoring. Supplies include containers for liquid and solid waste. The decontamination kits contain oxidizing agents for decontamination of the skin due to radioiodines.

Decontamination operations at the assembly area will be confined to minor decontaminations because of limited resources. If necessary, individuals will be furnished with protective clothing and transported to alternate facilities. Contaminated clothing will be disposed of as radioactive waste.

The EOF has a decontamination shower with associated liquid retention system. Equipment for small decontaminations is also available along with personnel monitoring equipment.

Contaminated individuals may be provided whole body counting analysis, as determined by the Radiological Emergency Coordinator. Whole body counting systems are located at PI & MT NGPs and/or mobile units which can be transported on or near the site.

Emergency First Aid will be applied to all injuries including contaminated injuries since contamination will not be life threatening whereas the lack of first aid could be life threatening.

First aid kits are located at the primary emergency centers in the plant.

The First Aid responsibility will be assigned to the Security Officer/EMT when they arrive on the scene. Selected members of the Security Force and plant staff are trained in Advanced First Aid and/or Emergency Medical Training (EMT).

The skill level of the staff is sufficient until offsite medical personnel arrive or until the victim is transported to the local hospital for further medical treatment.

The Operations Manual, Section F4, Medical Support and Casualty care, contains specific procedures for first aid situations complicated by contamination.

6.7.3 Medical and Public Health Support

Medical support and treatment for radiological and non-radiological injuries is provided by the Mayo Clinic Health System located in Red Wing, Minnesota.

Mayo Clinic Health System has a staff of physicians and hospital personnel trained in the proper methods of contamination control. At least one physician has been offered special courses on the treatment of radiological injuries. Prairie Island NGP conducts yearly training sessions with hospital personnel assuring a knowledge of radiation and contamination control procedures.

Regions Hospital in St. Paul, Minnesota is designated as the definitive care center for Prairie Island Nuclear Generating Plant. Regions Hospital may be used for radiation casualties, severe burn casualties, and other non-radiation injuries with use of an appropriate medical air transport service. Medical definitive care centers are offered periodic radiological contamination control training by the Minnesota Division of Homeland Security and Emergency Management (HSEM) according to their plan.

Monitoring instruments and supplies are located at Mayo Clinic Health System to aid in radiation monitoring and contamination control (e.g., DLR's, SRD's, protective clothing, survey meters, etc.).

All casualties on site will be administered emergency First Aid and radiation casualties will be decontaminated to every extent possible prior to departure from the plant site to the hospital. Proper application of first aid will take precedence over decontamination efforts.

Transportation of radiation casualties from Prairie Island NGP will be provided by the Red Wing Ambulance Service. In addition to the Red Wing Ambulance, a plant vehicle could be used as an emergency vehicle for transportation of victims to the hospital.

Procedures to be used at the plant and at the hospital in treating victims of an accident involving radiation exposure and/or personnel contamination are established and delineated in Section F4 of the Plant Operations Manual, Medical Support and Casualty Care.

In addition Sacred Heart Hospital in Eau Claire, WI is prepared and will support request for assistance in response to an emergency at the Prairie Island Nuclear Plant. Sacred Heart Hospital will serve as a radiation accident receiving hospital and has a decontamination room and trauma treatment rooms with isolation capabilities.

6.7.4 Whole Body Counting Facilities

A whole body counter is available at the Prairie Island NGP for determining the uptake of radioactivity. If this area becomes uninhabitable, the person may be transported to Monticello NGP where another whole body counter is available. Additional mobile whole body counters may be brought near or on the site if conditions make it a viable or necessary alternative.

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7.0 EMERGENCY FACILITIES AND EQUIPMENT

7.1 Emergency Control Centers

7.1.1 Technical Support Center (TSC)

The Technical Support Center (TSC) is located across the Turbine Building from Units 1 & 2 Control Room. A plan view of the TSC is shown in Figure 6.

The Technical Support Center (TSC) will serve as a center outside the Control Room from which the plant management, technical, and engineering support personnel will:

A. Support the Control Room command and control functions

B. Assess the plant status and potential offsite impact

C. Coordinate emergency response actions

The Technical Support Center has the following capabilities:

A. Working space for about twenty-five people on the main floor and working space for additional people on the other floor.

B. Shielding and ventilation cleanup system (PAC filter) to provide habitability under accident conditions.

C. An emergency locker containing monitoring equipment (radiation and airborne), respiratory protection equipment and thyroid blocking agent tablets.

D. Communication channels to all onsite and offsite emergency response centers (primary and backup).

E. A complete set of as-built drawings and other records such as plant layout drawings.

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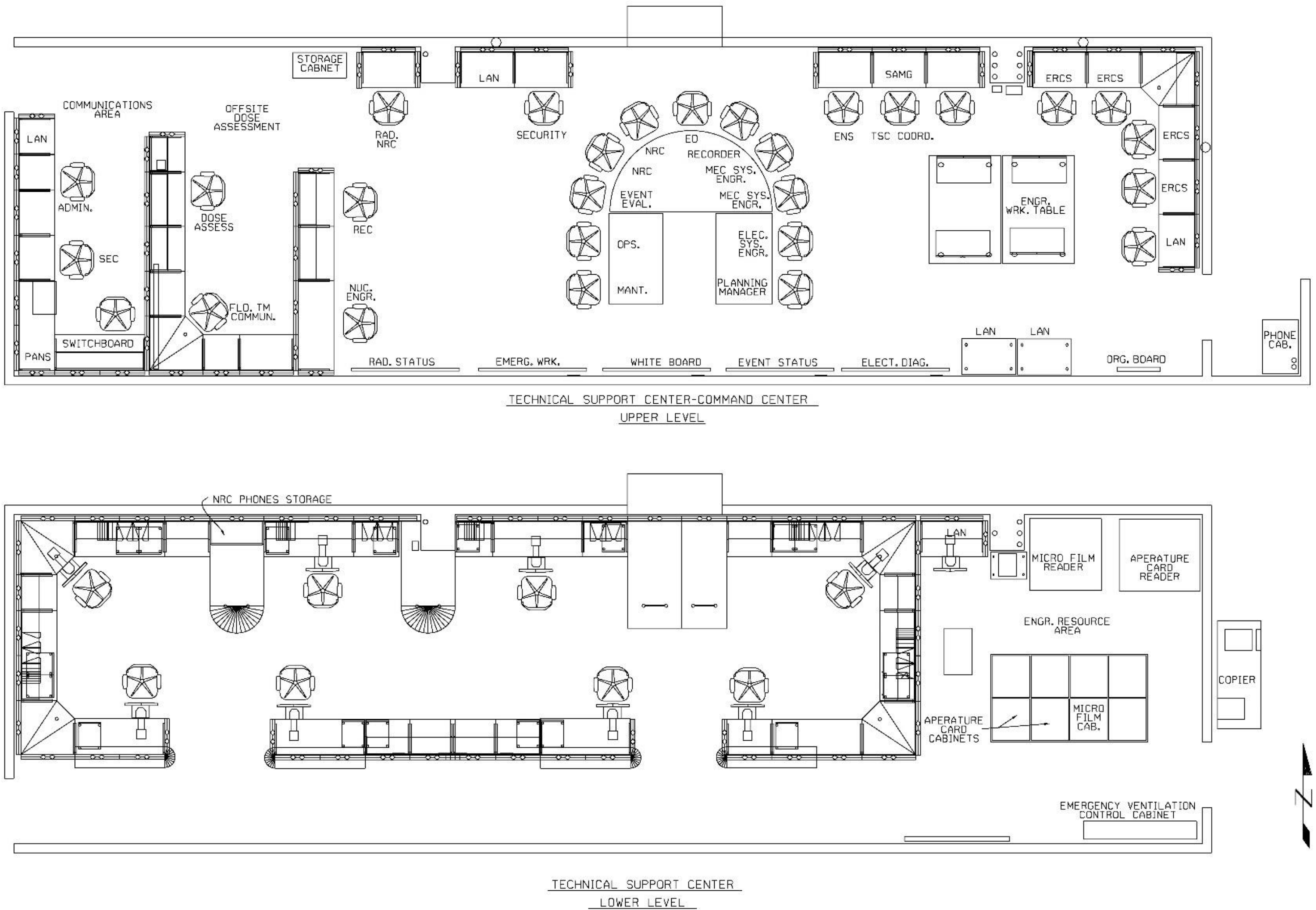
Figure 6 Plan View of TSC



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F. The capability to record and display the following:

1. Plant System Parameters

   a Reactor Coolant System
   b Secondary System
   c ECCS System
   d Containment

2. In-Plant Radiological Parameters

   a Reactor Coolant System
   b Containment
   c Effluent Treatment
   d Release Paths
   e Area Monitors

3. Offsite Radiological Parameters

   a Meteorology
   b Offsite Radiation Levels

The Technical Support Center **SHALL** be activated within 60 minutes when an Alert, Site Area or General Emergency is declared.

The Technical Support Center Coordinator **SHALL** be responsible for coordinating activities in the TSC. This individual **SHALL** be responsible for establishing the monitoring of direct radiation and airborne activity in the Technical Support Center. Communications **SHALL** be established between the TSC, OSC, Control Room and EOF.

If activation of the Technical Support Center occurs during normal work hours, instructions to report to the TSC will be received over the plant public address system.

If activation of the Technical Support Center occurs during the off duty hours, the Shift Manager **SHALL** designate the Shift Emergency Communicator to contact the Emergency Response Organization (ERO) by phone and/or ERO Pager Network and request them to report to the Technical Support Center.

7.1.2 Operational Support Center (OSC)

The Operational Support Center will provide a center to assemble the necessary Operators, Radiation Protection Specialists, Instrument and Control, Electrical, Nuclear Plant Service Attendants, and Maintenance personnel to support the operations of the plant under emergency conditions without causing undue congestion in the Control Room.

The Operational Support Center is located in the New Administration Building.

The Operational Support Center will be activated within 60 minutes when an Alert, Site Area or General Emergency is declared.

The Operational Support Center Coordinator **SHALL** be responsible for the activation and coordination of activities in the OSC. The OSC Coordinator may designate a communicator to establish lines of communications between the Operational Support Center, the Control Room and the Technical Support Center.

If activation of the OSC occurs during a normal working day, instructions to report to the OSC will be received over the plant public address system. Any Operations shift personnel on site that are not assigned to normal shift duty **SHALL** report to the OSC immediately. The following personnel will also report to the OSC if on site (additional personnel will be contacted as necessary):

A. Maintenance Supervisors (Mechanical and Electrical)

B. Designated Lead Station Electricians and Maintenance personnel

C. Instrument and Control Supervisors

D. Radiation Survey Team Members

E. Nuclear Plant Service Attendants

If activation of the Operational Support Center occurs during off duty hours, the Shift Manager **SHALL** designate the Shift Emergency Communicator to activate the onsite emergency organization to establish an initial complement of support personnel to assist in the emergency (additional personnel will be contacted as necessary):

A. Maintenance Supervisors (Mechanical and Electrical)

B. Designated Lead Station Electricians

C. Instrument & Control Supervisors

D. Radiation Survey Team Members

E. Designated Purchasing & Inventory Control Personnel

F. Nuclear Plant Service Attendants

Instrumentation is stored in the emergency locker which provides for monitoring both direct radiation and airborne radioactive contaminants.

An emergency locker located in the OSC contains all equipment necessary for reentry into the plant. This includes but is not limited to both waterproof and paper coveralls, respiratory protection (SCBAs), dosimeters, radiation detection meters, air samplers, decontamination and first aid equipment.

Communication equipment (radio and telephone) is available for contacting designated sections of the emergency response organizations.

7.1.3 Emergency Operations Facility (EOF)

The Emergency Operations Facility (EOF) is a required emergency response facility located near the plant site to provide continuous coordination and evaluation of activities during an emergency having, or potentially having, environmental consequences. A plan view of the EOF Command Center is shown in Figure 7. The EOF will be activated within 90 minutes when an Alert, Site Area or General Emergency is declared.

The functions of the EOF will be:

A. Management of the overall NSPM's offsite emergency response in support of plant activities;

B. Evaluate the magnitude and effects of actual or potential radioactive releases from the plant;

C. Recommend appropriate offsite protective measures, in conjunction with the TSC personnel;

D. Coordinate the offsite radiological monitoring during emergencies and recovery operations;

E. Coordinate emergency response activities with those of local, State, Tribal, and Federal emergency response organizations;

F. Provide current information on conditions potentially affecting the public to the NRC and to offsite emergency response agencies;

G. Act as the post-accident recovery management center for both onsite and offsite activities, if necessary.

The EOF will be staffed by personnel from the Engineering and Projects Management groups and Prairie Island Training Center staff. Activation and various responsibilities within the EOF are described fully in Section F8 of the EOF Emergency Plan Implementing Procedures.

The EOF has been constructed and designed in accordance with the guidance of NUREG 0696. The building has been designed to serve primarily as a Training Center on a regular basis with the capability for prompt conversion to the EOF function when required and, if needed, will serve as the Recovery Center.

The EOF is constructed in a manner which provides habitability in an accident situation. Shielding and ventilation treatment systems have been installed to maintain an acceptable environment. The EOF section of the training building is a concrete structure that contains sufficient shielding to exceed a protection factor of 5. The ventilation system has an emergency mode of operation that will pressurize the building through a High Efficiency Particulate Absolute (HEPA) filtration system.

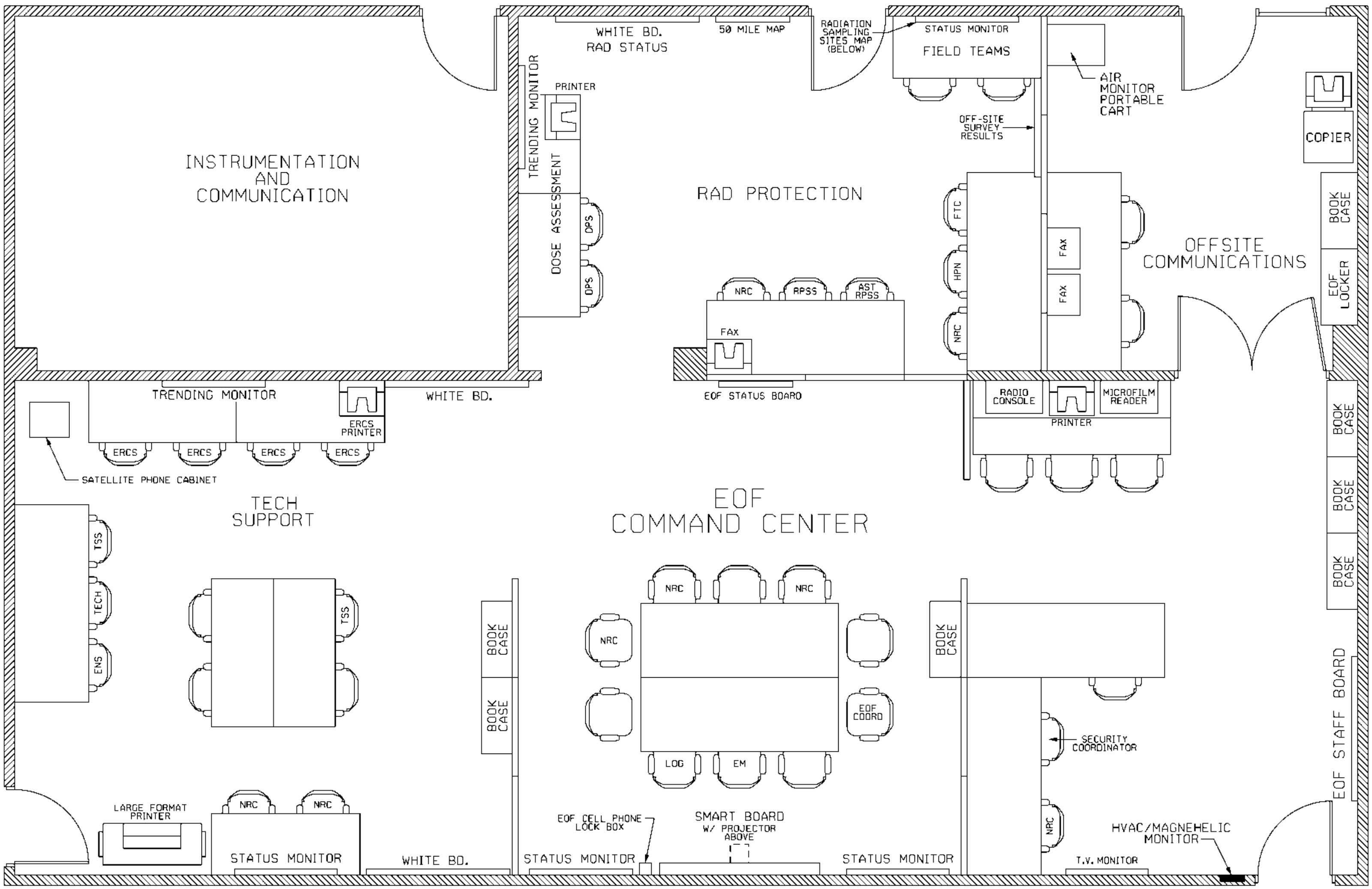
The general layout of the building's entrances and exits have been given consideration for operation of the building in an emergency mode. Radiological monitoring and alarming are provided for the EOF portion of the building. Extensive communication equipment is installed in the building to provide primary and backup means of communication with outside agencies, offsite survey teams, TSC and the Control Room. The EOF portion of the building is served by a dual source power supply for those services necessary to make the EOF functional.

The EOF provides office space for each plant support group, key supervisors, state, local and tribal officials, and the NRC, as well as functioning as a command center. Each space is provided with furnishings necessary to perform routine office functions. The plant support groups and governmental representatives will perform their respective functions in these assigned offices. The command center is intended to function as a work space for the Emergency Manager, Radiation Protection Support Group, Technical Support Group, and for related critical communications. These activities are assigned to this area, due to the high volume of activity and the importance of the information handled. Additionally, this area is the central area for displaying plant status, offsite survey status, conducting accident assessment and directing the activities of the offsite Emergency Response Organization.

The EOF is supplied with the equipment necessary to fulfill its function as an offsite emergency response center. Radiation monitoring and decontamination equipment has been provided to supply offsite monitoring teams. Normal and emergency data acquisition is made available via the Emergency Response Computer System (ERCS). Office equipment such as facsimile machines, copy machines, microfiche readers, computers and printers connected to the Local Area Network are provided to facilitate administrative duties and technical reference work. General office supplies are stocked in adequate numbers. Operating procedures detailing the methods to activate the EOF, conduct routine administrative operations, surveys and accident assessment, analyze offsite survey samples, provide security and deactivate the Emergency Organization are developed and are available in the EOF. Other organization's procedures, plans and reference documents are also available to EOF personnel. If there is a need for expanded support facilities such as trailer space or communication hook-ups for vendors and support contractors, it may be provided at the EOF.

Because the EOF is located within the 10 mile EPZ, a Backup EOF exists in case an evacuation of the EOF is necessary. Equipment and facilities necessary to carry out this function are located at Xcel Energy corporate offices in downtown Minneapolis, Minnesota. A description of the Backup EOF facility is described in the Monticello & Prairie Island Offsite Nuclear Emergency Plan.

Figure 7 Plan View of EOF Command Center



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7.1.4 Control Room

The Control Room **SHALL** be the initial onsite center of emergency control. Control Room personnel must evaluate and effect control over the initial aspects of the emergency and initiate responses necessary for coping with the initial phases of an emergency until such time that the onsite emergency centers can be activated. These activities **SHALL** include:

A. Continuous evaluation of the magnitude and potential consequences of an incident

B. Initial corrective actions

All plant operations are controlled from here by the Shift Manager with direction from the management personnel located either in the Control Room or Technical Support Center.

The Control Room contains the necessary instrumentation (process and radiological) to evaluate all plant conditions. Habitability is maintained by shielding and the special ventilation system (PAC Filter), which is capable of operating in a cleanup or recycle mode.

All emergency equipment is supplied power from the emergency diesel generators with vital instrumentation powered from inverters connected to the storage batteries located in the battery rooms.

7.1.5 The Red Wing Service Center (RWSC) is to be used as an Alternative Facility during a hostile action or security event in the event that response to the site is unsafe. The RWSC will be used by TSC and OSC personnel until it has been determined that it is safe to return to the plant site. This facility is accessible in the event of an onsite Hostile Action and provides the ability to perform the following functions:

- Communication with the Control Room and onsite Security Forces
- Notification of offsite Emergency Response Organizations
- Engineering Assessment Activities including damage control team preparation and planning.

7.2 Communications

Various onsite and offsite communication systems are described in the following sections. Table 5 depicts the various communication links that may be established.

7.2.1 Onsite Communications

All emergency operating facilities have at least two means of communications: (1) portable or installed radio systems; and (2) normal telephone communications.

The normal onsite communications during an emergency will be made via the plant telephone system with a public address system option. The telephone system is powered by noninterruptible power. The public address system includes about 175 loudspeakers located throughout the entire plant area.

A separate paging system has 20 handsets located at strategic plant areas.

At approximately 120 locations in the plant, jackboxes are located for the sound powered system. Each box contains six independent circuits for sound powered headsets. A jackbox is located in the Technical Support Center and Control Room.

The Control Room, Technical Support Center and EOF each have a multi-channel radio system console for communications. At least 50 portable radios are available for use throughout the plant during emergency conditions.

The plant evacuation alarm consists of a 125 VDC operated siren, manually started from the Control Room. This tone consists of a signal starting at approximately 600 cycles per second rising to a peak of approximately 1450 cycles per second, then returning slowly to the low value of 600 cycles per second and repeating. The Control Room operator can remove the siren tone for emergency voice communication over the loudspeaker PA system.

The plant fire alarm consists of a modulating signal interrupted continuously to give a Yip-Yip-Yip sound. This is activated manually from the Control Room.

During an emergency, designated individuals will be responsible for the communications at each of the emergency facilities, as delineated in Section 6.1.3.

7.2.2 Offsite Communications

Both normal and alternate communication links are provided to offsite agencies. Individuals designated to staff the offsite agency communication links are delineated in Section 6.1.3.

The Xcel Energy telephone network provides normal communications to offsite agencies through telephone lines via the Red Wing US West telephone Exchange, or via Xcel Energy fiber optic SONET communications network.

The Control Room, Technical Support Center and EOF have a dedicated Xcel Energy radio channel link to the Xcel Energy System Control Center, the Backup EOF, and the Minnesota HSEM Emergency Operating Center in St. Paul, Minnesota.

The Technical Support Center and EOF have a National Warning System (NAWAS) extension to the Wisconsin Emergency Management EOC at Madison, the Regional Warning Center at Eau Claire and the Pierce County EOC at Ellsworth, Wisconsin.

The Control Room, Technical Support Center and EOF each have a portable cellular phone and satellite phone for emergency communication use, as necessary.

The Technical Support Center has access to a computerized auto dial system used for notification of the site's Emergency Response Organization (ERO). This system consists of a telephone network of several outgoing telephone lines. When activated, it will call and deliver an emergency message to the plant's emergency organization's home telephones.

The plant also has an Emergency Response Organization (ERO) Pager Network. Designated members of the site's emergency organization carry personal pagers which can be activated from the Technical Support Center, Control Room or alternate facility (RWSC). A special emergency code is displayed on the pager.

The Control Room, Technical Support Center and EOF have multi-channel radio system for communication with all Plant Radiation Survey Teams, Plant Operations Personnel, Plant Security Areas, county sheriffs, county EOC's, and Treasure Island Casino (Prairie Island Indian Tribe).

A telecopying network is set up between the TSC, EOF, state & county EOC's and Prairie Island Indian Tribe for the purpose of telecopying update information.

An emailing network is setup between the offsite agencies for the purpose of emailing the emergency notification form.

Auto ring lines link the Technical Support Center to the EOF and the Technical Support Center to the Minnesota State EOC.

Communication links are maintained with medical facilities, both fixed and mobile. The plant can update the hospital via the telephone network of the status of any injuries. Communication channels are provided between the hospital and the ambulance service via the radio system while the victim(s) are enroute.

The plant site also supports the NRC's Emergency Telecommunications System (ETS). The dial tone for the Prairie Island 106G PETS circuits are provided by Xcel Energy's corporate communication network. The ETS provides for reporting emergencies and other significant events to the NRC, Incidence Response Center in Rockville, Maryland. Using the Xcel Energy's private network should avoid the public switched network blockage anticipated during a major emergency.

The following NRC essential emergency communications functions will be provided by the ETS voice service.

A. Emergency Notification System (ENS): Initial notification by the licensee, as well as ongoing information on plant systems, status, and parameters. The ENS (Red Phone) is located in the Control Room, with extensions in the Technical Support Center (TSC) and EOF.

B. Health Physics Network (HPN): Communication with the licensee on radiological conditions (in-plant and off-site) and meteorological conditions, as well as their assessment of trends and need for protective measures on-site and off-site. NRC regional office or NRC Headquarters will announce their decision to establish the HPN link over the ENS. The HPN phones are located in the TSC and EOF.

C. Reactor Safety Counterpart Link (RSCL): Established initially with the base team, and then with the NRC site team representatives once they arrive at the site, to conduct internal NRC discussions on plant and equipment conditions separate from the licensee, and without interfering with the exchange of information between the licensee and NRC. This is the channel by which the NRC Operations Center supports NRC reactor safety personnel at the site. In addition, this link may also be used for discussion between the Reactor Safety Team Director and licensee plant management at the site. The RSCL phones are located in the TSC and EOF.

D. Protective Measures Counterpart Link (PMCL): Established initially with the base team, and then with the NRC site team representatives once they arrive at the site, to conduct internal NRC discussions on radiological releases and meteorological conditions, and the need for protective actions separate from the licensee and without interfering with the exchange of information between the licensee and NRC. This is the channel by which the NRC Operations Center supports NRC protective measures personnel at the site. In addition, this link may also be used for discussion between the Protective Measures Team Director and licensee plant management at the site. The PMCL phones are located in the TSC and EOF.

E. Emergency Response Data System (ERDS) Channel: This dedicated computer network is a direct near real-time electronic data link between the plant's on-site computer system and the NRC Operations Center that provides for the automated transmission of a limited data set of selected parameters. The plant activates the ERDS within one hour after declaring an emergency class of Alert, Site Area, or General Emergency. The ERDS supplements the existing voice transmission over the ENS.

F. Management Counterpart Link (MCL): Established for any internal discussions between the Executive Team Director or Executive Team members and the NRC Director of Site Operations or top level licensee management at the site. The MCL phones are located in the TSC and EOF.

G. Local Area Network (LAN) Access: Established with the base team and the NRC site team for access to any of the products or services provided on the NRC Operations Center's local area network. This includes technical projections, press releases, status reports, E-Mail, and various computerized analytical tools. The LAN access points are located in the TSC and EOF.

Table 5 Prairie Island Site Communications Matrix

	Email	Plant Phone Network	Plant Sound Powered Phone System	Portable Cellular Telephone	Plant Page System	NAWAS	Facsimile	USNRC/ENS	USNRC HPN	Xcel Metro Radio System	Plant Radio System	Auto Ring Direct Dialing Phone	ERO Auto Dial System	Personal Pagers	USNRC MCL RSCL PMCL LAN
Control Room		x	x	x	x			x		x	x	x		x	
Tech Support Center	x	x	x	x	x	x	x	x	x	x	x	x	x	x	x
Ops Support Center		x			x						x			x	
Emerg Op Facility	x	x		x	x	x	x	x	x	x	x	x		x	x
Backup EOF	x	x		x			x	x	x	x		x			
Xcel/ System Ops		x								x	x	x			
Monticello NGP		x													
PI Plant Areas		x	x		x						x			x	
PI Monitoring Teams											x			x	
PI Indian Tribe	x	x					x				x				
MN/HSEM-EOC	x	x					x			x		x			
Goodhue Co. Sheriff	x	x					x				x				
Goodhue-Red Wing EOC	x	x					x				x				
Dakota Co. Sheriff	x	x					x				x				
Dakota EOC	x	x					x				x				
WI/WEM-EOC	x	x				x	x								
WI/WEM-Eau Claire		x				x									
Pierce Co. Sheriff	x	x				x	x				x				
Pierce EOC	x	x				x	x				x				
Red Wing Police/Fire		x													
Red Wing Hospital		x					x								
USNRC/HQ		x					x	x	x						x
USNRC/REG III		x						x	x						x
USNRC/Resident Insp.		x			x									x	
PI Emerg. Personnel		x											x	x	
MN/State Patrol		x													
WI/State Patrol		x				x									
National Weather Service		x													

7.2.3 Alert and Notification System (ANS)

Within the Plume Exposure Emergency Planning Zone (EPZ) there exist provisions for alerting and providing notification to the public. It is the responsibility of state and county governments to activate this system.

The plant maintains a basic fixed siren system for essentially 100% coverage of the offsite population within 5 miles of the plant and population center coverage for the 5-10 mile zone. To reach persons not covered by these population center sirens, Homeland Security Emergency Management or the MN Duty Officer also activates the Integrated Public and Alert Warning System (IPAWS).

A special electronic siren is maintained near the Prairie Island Indian Community Center. The TSC has the capability to activate the siren with a special “stutter” tone at the declaration of a Site Area Emergency for the purpose of quickly notifying Prairie Island’s Indian tribal leaders except during a Hostile Action Based (HAB) event. The siren would also be activated with the normal “Alert” tone by the Goodhue County Sheriff’s Department during a General Emergency as part of the normal Public Alert and Notification System activation.

To supplement PANS, emergency alert radios have been installed in various commercial, institutional, and educational facilities in the 10-mile zone. These locations may harbor large groups of people during all or part of a day, justifying radio alert service, even though many of these facilities are already covered by state and county emergency warning plans. The emergency radios will either be activated by the National Weather Service or by the local county sheriff’s dispatch office.

In the event of an emergency condition, alert and notification information will be relayed through established communication links described in the Minnesota and Wisconsin emergency response plans. Upon receiving notification of the emergency, offsite governments will, if necessary, activate public warning and information procedures which include the State Emergency Alert System (EAS). With this system, essentially 100% of the population in the 10 mile EPZ will be alerted within 15 minutes

In the event a county primary siren activation system fails to operate, each county has a backup siren activation process on a separate activation system utilizing a different tower and controls for activation of the sirens.

In conjunction with the siren system activation, the Integrated Public Alert Warning System (IPAWS) is also activated. This system is also used as a backup when the siren system or individual sirens are out of service.

7.3 Assessment Facilities

The plant instrumentation and monitors perform indicating, recording and protective functions. The Reactor Protection System and associated plant instrumentation provide the ability to maintain plant safety from shutdown to full power operations and to monitor and maintain key variables such as reactor power, flow, temperature, and radioactivity levels within predetermined safe limits at both steady state conditions and during plant transients. Plant instrumentation and control systems also provide means to cope with abnormal operating conditions. The control and display of information of these various systems are centralized in the main Control Room. This instrumentation would provide the basis for initiation of protective actions.

7.3.1 Onsite Systems and Equipment

A. Geophysical Phenomena Monitors

1. Meteorological

Prairie Island has a 60 meter onsite meteorological tower located approximately 0.5 miles northwest of the plant. The tower is equipped with primary and secondary redundant sensors for the 10 and 60 meter temperatures, wind speeds, and wind directions powered by a primary and secondary power source. The following meteorological information is supplied by the tower:

a Wind Direction (10 and 60 meter)
b Wind Speed (10 and 60 meter)
c Ambient Temperature
d ΔT between 10 and 60 meter temperature indications
e Precipitation

A 22 meter backup meteorological tower is located near the EOF. The backup meteorological tower provides the following:

All meteorological data is processed via the ERCS, and may be displayed in the Control Room, TSC, EOF, and Backup EOF.

Barometric pressure is also available in the Control Room.

2. Seismic

The Control Room has an installed earthquake detection system with a three step graded severity level of alarms:

a Seismic Event – 3 percent vertical or horizontal acceleration
b Operational Basis Earthquake – 4 percent vertical or 6 percent horizontal acceleration (No equipment failure)
c Design Basis Earthquake – 8 percent vertical or 12 percent horizontal acceleration (possible equipment failure)

A visual and audible alarm will sound in the Control Room. Upon activation, the accelerometers and accelerographs listed on Table 6 will be automatically recorded for future investigation.

3. Hydrologic

River water level is available from two sources:

a Indicators in Control Room which receive a signal from capacitance level probes located in several locations in the river water canals and the intake screenhouse.
b Lock and Dam #3 (located about 1.6 miles SE) which would give essentially the same indication as at Prairie Island NGP.

B. Radiation Monitoring Equipment

Onsite radiation monitoring equipment at Prairie Island NGP can be categorized into the following groups:

1. Process radiological monitoring system

2. Effluent radiological monitoring system

3. Airborne radioactivity monitoring system

4. Area radiation monitoring system

5. Portable survey and counting room equipment

Table 7 lists all the area, process, and effluent monitors. Table 8 lists the general types of portable survey, count room, airborne monitoring and personnel monitoring equipment.

C. Process Monitors

Adequate instrumentation monitoring capability exists to properly access the plant status during all modes of operation, i.e., instrumentation is available to the operator to determine plant status, aid in emergency classification determination, and aid in post accident assessment. Table 9 lists available instrumentation, ranges and their indicator locations.

D. Fire Detection

The fire detection system consists of various types of detectors/flow devices throughout the main power building and in most of the outbuildings. Ionization, flame and thermal type fire detectors are located throughout safety related structures. Audible alarming is on the Control Room annunciator panel system for actuation or trouble. The Control Room fire panel system will indicate zone location of the alarm. On receipt of the annunciator panel alarm, the fire panel is checked for location and operator assigned to effected area is called for immediate investigation.

Further details of the fire detection system are given in the plant safety procedures, Section F5 Appendix K, “Fire Detection and Protection Systems.”

E. Post Accident Liquid Sampling

A post-accident liquid sampling system is installed at Prairie Island with associated procedures to provide the capability to obtain the following samples:

1. Sample of raw reactor water

2. Diluted samples of reactor water (boron, chloride, isotopic analysis, pH, etc.)

3. Dissolved gas sample for isotopic analysis (noble gases)

4. Dissolved hydrogen sample

The sampling system includes the following exposure reduction equipment:

1. Shielded sample lines and shielded drain lines in the Hot Sample Room.
2. Shielded sample panel which allows collections and analysis of a reactor coolant sample for hydrogen and isotopic analysis.
3. Shielded sample carriers for transporting samples to remote facilities (Hot Cell).
4. Remote analysis lab (Hot Cell) located on 695′ elevation in the Turbine Building.
5. Shielded work area in the Hot Cell with an exhaust hood installed, which discharges through a PAC filter unit.
6. Remote counting labs with geometries for counting extremely high level radioactivity samples.

This system allows sample collection and analysis within the radiation exposure guidelines given in NUREG 0578.

F. Containment Air Sampling

Following an accident, a containment air sample may be obtained, utilizing the gas analyzer to extract a sample via the Hydrogen Post LOCA System for determination of:

1. Hydrogen content
2. Isotopic analysis (noble gas)

All sampling will be completed within the exposure guidelines of NUREG 0578.

G. Shield Building Vent Sample

The Shield Building Stack Hi-Range Monitor (located in the third floor of the turbine building) extracts a sample from the Shield Building stack and pumps it through a large sample chamber which houses the radiation detector. The hi-range detector reading is in mR/hr and is easily converted to mCi/cc via the applicable calibration curves.

Prior to entering the sample chamber, the sample flow is directed through a particulate filter and a silver zeolite adsorber. The particulate filter and silver zeolite adsorber are manually removed and prepared for analysis in the counting labs.

NOTE:	**Silver zeolite adsorbers eliminate the problem of entrapped noble gases on the iodine adsorber allowing a much lower detection sensitivity. In addition, air or N_2 may be used to blow out the adsorber to further eliminate the entrapped noble gases.**

In instances of monitor failure or offscale readings, procedures are available to allow the dose rate on the sample chambers to be measured using portable survey meters. The release concentration can then be calculated by converting the dose rate to concentration utilizing applicable calibration curves.

H. Containment High Range Area Monitors

Two channels of Containment High Range Dome monitors are installed in the containments. Full scale reading on these monitors is 10^8 R/hr. This allows personnel to estimate the amount of activity in containment available for release and the severity of the accident from the applicable calibration curves.

I. In-Plant Iodine Determination

During emergency conditions, it will be necessary for emergency personnel to rapidly and accurately determine or estimate the airborne iodine activity in areas of the plant including all operating centers.

Samples for iodine activity are obtained with portable air samplers (AC and battery operated) and continuous air monitors (CAM's). The iodine is collected on silver zeolite adsorbers.

NOTE:	**The use of silver zeolite adsorbers reduces the amount of noble gases entrapped on the adsorber. This reduces the minimum sensitivity level of iodine on the adsorber. In addition, air or N_2 may be used to blow out the adsorber to further reduce the amount of entrapped noble gases.**

The silver zeolite adsorbers may be analyzed using the GEM system in the onsite counting room or the EOF Counting Room. The adsorbers could also be analyzed with portable instrumentation.

The Control Room, Operational Support Center, Technical Support Center and EOF have continuous air monitors (CAM's) available to monitor the airborne iodine levels. A detector is continuously analyzing the activity (iodine) trapped on the carbon-impregnated filter paper.

This combination of equipment allows iodine determinations under all plant accident conditions.

An Iodine Monitoring program, acceptable to the NRC, was described in letters from L.O. Mayer, NSP, to Director of Nuclear Reactor Regulation, dated December 31, 1979, "Lessons Learned Implementation" and March 13, 1980, "1/1/80 Lessons Learned Implementation Additional Information."

J. Steam Line Monitors

The steam line radiation monitor in conjunction with the ERCS (Emergency Response Computer System) computer will supply a value for noble gas activity released via the steam headers (steam dumps and safeties).

An alternate steam header release calculation procedure exists which allows the determination to be made with portable radiation equipment and applicable calibration curves. This will allow a backup method for release determination during instances of monitor failure.

NOTE:	**Normally the air ejector discharge is routed to the Shield Building Exhaust stacks which are monitored by the low and high range stack radiation monitors.**

K. Air Ejector Noble Gas Release

Releases through the air ejectors are quantified via: (1) the installed air ejector radiation monitor and applicable calibration curves; (2) the Shield Building Exhaust Stack monitors (low and high range) and their applicable calibration curves; or (3) by local sample analysis.

Table 6 Seismographic Monitoring Devices

Triaxial Accelerometers

(1)	Unit 1 Containment Low	32.5/210/697.5
(2)	Unit 1 Containment High	32.5/210/765.5
(3)	Aux Bldg Ground Floor	J.0/9.0/695
(4)	Unit 2 Containment High	29/95/765.5

Triaxial Accelerographs

(1)	Aux Bldg Ground Floor	J.0/9.0/695
(2)	Aux Bldg Spent Fuel Pool	N.8/9.0/755
(3)	Aux Bldg Fan Floor	J.0/9.0/755
(4)	Unit 1 Containment Low	32.5/210/697.5
(5)	Unit 1 Containment High	32.5/210/765.5
(6)	Unit 2 Containment High	29/95/765.5
(7)	Unit 2 Containment Low	29/95/697.5
(8)	Turbine Building Ground Floor	C.9/8.4/695
(9)	Turbine Building Operating Floor	C.6/8.8/735
(10)	Screenhouse Low	C1.0/81.8/670
(11)	Screenhouse High	C1.0/81.8/695
(12)	Screenhouse Cooling Water Piping	B1.9/91.5/680
(13)	Screenhouse Cooling Water Piping	C1.5/91.7/692
(14)	Screenhouse Cooing Water Piping	C1.5/81.3/692
(15)	Aux Bldg Chem & Vol Control Piping	L.1/7.0/741
(16)	Aux Bldg Aux Feedwater	H.7/6.9/709

Table 7 Radiation Monitors

Area Monitors

Radiation Channel No.	Detector/Location	Instrument Range
R-1	GM/Control Room	0.1-10^4 mr/hr
1-R-2	GM/Containment Vessel Unit 1	0.1-10^4 mr/hr
2-R-2	GM/Containment Vessel Unit 2	0.1-10^4 mr/hr
R-3	GM/Radiochem Lab	0.1-10^4 mr/hr
R-4	GM/Charging Pumps Unit 1	0.1-10^4 mr/hr
R-5	GM/Spent Fuel Pool	0.1-10^4 mr/hr
R-6	GM/Hot Sample Room	0.1-10^4 mr/hr
1-R-7	GM/Incore Seal Table Unit 1	0.1-10^4 mr/hr
2-R-7	GM/Incore Seal Table Unit 2	0.1-10^4 mr/hr
R-8	GM/Waste Gas Valve Gallery	0.1-10^4 mr/hr
1-R-9	GM/Letdown HX Unit 1	0.1-10^4 mr/hr
2-R-9	GM/Letdown HX Unit 2	0.1-10^4 mr/hr
R-28	Scint/New Fuel Pit	1.0-10^5 mr/hr
R-29	Scint/Shipping Receiving	0.1-10^4 mr/hr
R-32	Scint/Rad Waste Control Station	0.1-10^4 mr/hr
R-33	Scint/Rad Waste Bldg/2nd Floor	0.1-10^4 mr/hr
R-36	Scint/Charging Pumps Unit 2	0.1-10^4 mr/hr
1-R-48	Ion Chamber/Containment Vessel Unit 1	1-10^8 R/hr
1R-49	Ion Chamber/Containment Vessel Unit 1	1-10^8 R/hr
2R-48	Ion Chamber/Containment Vessel Unit 2	1-10^8 R/hr
2R-49	Ion Chamber/Containment Vessel Unit 2	1-10^8 R/hr

Table 7 Radiation Monitors

Area Monitors

Radiation Channel No.	Detector/Location	Instrument Range
1-R-53	SI Pump Area, Unit 1	0.1-10^7 mr/hr
2-R-53	SI Pump Area, Unit 2	0.1-10^7 mr/hr
1-R-54	CS Pump Area, Unit 1	0.1-10^7 mr/hr
2-R-54	CS Pump Area, Unit 2	0.1-10^7 mr/hr
1-R-55	Aux Bldg 695 East Area	0.1-10^7 mr/hr
2-R-55	Aux Bldg 695 West Area	0.1-10^7 mr/hr
1-R-56	Aux Bldg 695 West Area	0.1-10^7 mr/hr
2-R-56	Aux Bldg 695 East Area	0.1-10^7 mr/hr
1-R-57	Aux Bldg 715 East Area	0.1-10^7 mr/hr
2-R-57	Aux Bldg 715 West Area	0.1-10^7 mr/hr
1-R-58	Aux Bldg 715 West Area	0.1-10^7 mr/hr
2-R-58	Aux Bldg 715 East Area	0.1-10^7 mr/hr
1-R-59	Aux Bldg 715 Pent/Ltdn Area	0.1-10^7 mr/hr
2-R-59	Aux Bldg 715 Pent/Ltdn Area	0.1-10^7 mr/hr
1-R-60	Aux Bldg 735 North Area	0.1-10^7 mr/hr
2-R-60	Aux Bldg 735 North Area	0.1-10^7 mr/hr
1-R-61	A Stm Line Area	0.1-10^7 mr/hr
2-R-61	A Stm Line Area	0.1-10^7 mr/hr
1-R-62	Aux Bldg 755 East Area	0.1-10^7 mr/hr
2-R-62	Aux Bldg 755 West Area	0.1-10^7 mr/hr

NOTE:	**All Area Monitors on this page have Ion Chamber type detectors.**

Table 7 Radiation Monitors

Area Monitors

Radiation Channel No.	Detector/Location	Instrument Range
1-R-63	Aux Bldg 755 West Area	0.1-10^7 mr/hr
2-R-63	Aux Bldg 755 East Area	0.1-10^7 mr/hr
1-R-64	Turb Bldg 735 North Area	0.1-10^7 mr/hr
2-R-64	Turb Bldg 735 North Area	0.1-10^7 mr/hr
R-65	Oper Support Center	0.1-10^7 mr/hr
R-66	D1 Dsl Gen Room	0.1-10^7 mr/hr
R-67	Inst and Control Shop	0.1-10^7 mr/hr
R-68	Tech Support Center Rad	0.1-10^7 mr/hr
R-69	Guardhouse	0.1-10^7 mr/hr
2-R-72	D6 Cable Spreading Room	0.1-10^7 mr/hr
2-R-73	D6 Bus 26 4KV SWGR Room	0.1-10^7 mr/hr
2-R-74	D6 Bus 221 & 222 480V SWGR Room	0.1-10^7 mr/hr

NOTE:	All Area Monitors on this page have Ion Chamber type detectors.

Table 7 Radiation Monitors

Process Monitors

Radiation Channel No.	Detector/Location	Instrument Range
1-R-11	Scint/Containment and Shield Bldg Particulate Unit 1	10^1-10^6 cpm
2-R-11	Scint/Containment and Shield Bldg Particulate Unit 2	10^1-10^6 cpm
1-R-12	GM/Containment and Shield Bldg Gas Unit 1	10^1-10^6 cpm
2-R-12	GM/Containment and Shield Bldg Gas Unit 2	10^1-10^6 cpm
1-R-15	Scint/Condenser Air Ejector Unit 1	10^1-10^6 cpm
2-R-15	Scint/Condenser Air Ejector Unit 2	10^1-10^6 cpm
R-16	Scint/Fan Coils Wtr Disch Unit 1 & Unit 2	10^1-10^6 cpm
R-18	Scint/Waste Disposal Liquid Effluent	10^1-10^6 cpm
1-R-19	Scint/Steam Generator Blowdown Unit 1	10^1-10^6 cpm
2-R-19	Scint/Steam Generator Blowdown Unit 2	10^1-10^6 cpm
R-21	Scint/Circulating Water Dsch	10^1-10^6 cpm
1-R-22	GM/Shield Bldg Vent Gas Unit 1	10^1-10^6 cpm
2-R-22	GM/Shield Bldg Vent Gas Unit 2	10^1-10^6 cpm
R-23	GM/Control Room Vent	10^1-10^6 cpm
R-24	GM/Control Room Vent	10^1-10^6 cpm

Table 7 Radiation Monitors

Process Monitors

Radiation Channel No.	Detector/Location	Instrument Range
R-25	GM/Spent Fuel Pit Vent	10^1-10^6 cpm
R-26	GM/11/21 RHR Cubicle Vent	10^1-10^6 cpm
R-27	GM/12/22 RHR Cubicle Vent	10^1-10^6 cpm
1-R-30	GM/Aux Bldg Vent Gas Unit 1	10^1-10^6 cpm
2-R-30	GM/Aux Bldg Vent Gas Unit 2	10^1-10^6 cpm
R-31	GM/Spent Fuel Pit Vent	10^1-10^6 cpm
R-35	GM/Rad Waste Bldg Vent Gas	10^1-10^6 cpm
1-R-37	GM/Aux Bldg Vent Gas Unit 1	10^1-10^6 cpm
2-R-37	GM/Aux Bldg Vent Gas Unit 2	10^1-10^6 cpm
R-38	Scint/Fan Coils Wtr Dsch Unit 1 & Unit 2	10^1-10^6 cpm
1-R-39	Scint/Component Cooling Liquid Unit 1	10^1-10^6 cpm
2-R-39	Scint/Component Cooling Liquid Unit 2	10^1-10^6 cpm
R-41	GM/Waste Gas High Level Loop	10^1-10^6 cpm
1-R-50	Ion Chamber/Shield Bldg Vent Gas Unit 1	0.1-10^7 mr/hr
2-R-50	Ion Chamber/Shield Bldg Vent Gas Unit 2	0.1-10^7 mr/hr
1-R-51	GM/Steam Line Unit 1, Loop A	0.1-10^5 mr/hr
1-R-52	GM/Steam Line Unit 1, Loop B	0.1-10^5 mr/hr
2-R-51	GM/Steam Line Unit 2, Loop A	0.1-10^5 mr/hr
2-R-52	GM/Steam Line Unit 2, Loop B	0.1-10^5 mr/hr

Table 8 Radiation Monitoring Instruments and Devices

Portable Survey Instruments

Types	Range(s)
GMs	0-70,000 cpm
	0-1000 R/hr
Ion Chambers	0-50 R/hr
Scintillation	0-500,000 cpm
Tissue Equivalent	.001 mR/hr-999 R/hr
Proportional Counter	.001 mR-999 R/hr

Portable Air Sampling Equipment

Types	Range(s)
Continuous Air	50-50,000 cpm
Monitors	10-10^6 cpm
Air Samplers	2.5-20 CFM
	0-80 LPM

Analysis Equipment

Types
Tritium Liquid Scintillation Detection
Gamma Spectroscopy Analysis
Proportional Alpha/Beta Counters
GM Counter

Personnel Monitoring Equipment

Types	Range(s)
Self-Reading Dosimeters (manual)	0-200 mR
	0-1 R
	0-5 R
	0-100 R
Self Reading Dosimeters (electronic)	1 mR – 1000 R
DLR's	All Ranges
Finger Rings	All Ranges
Portal Monitors	0-30,000 cps

NOTE:	Exact quantities and locations are described in the plant's surveillance program procedures.

Table 9 Instruments Available for Monitoring Major Systems

MEASURED PARAMETER	TYPE OF READOUT	INDICATOR RANGE	INDICATOR LOCATION
1. Source Range			
Neutron Level	Log scale indicator, Recorder Computer output	0 to 10^6 cps	System cabinets Main Control Boards ERCS Computer
Startup Rate	Linear scale indicator, Computer output	-0.5 to 5 DPM	Main Control Boards ERCS Computer Hot Shutdown Panel
Neutron Flux Monitor (N51/N52)	Log Scale Linear Scale	10^{-1} to 10^5 cps -1 to 7 DPM	System cabinets ERCS Computer
2. Intermediate Range			
Neutron Level	Log scale indicator, Computer output	10^{-11} to 10^{-3} Amp	System cabinets Main Control Boards ERCS Computer
Neutron Level	Recorder	10^{-11} to 10^{-3} Amp	Main Control Board
Startup Rate	Linear scale indicator	-0.5 to 5 DPM	Main Control Boards ERCS Computer
3. Power Range			
Neutron Level	Linear scale indicator, Computer output	0-120% Full Power	System cabinets Main Control Boards ERCS Computer
Neutron Level	Recorder	0-120% Full Power	Main Control Boards
Neutron Flux Monitor (N51/N52)	Log Scale Linear Scale Linear Scale	10^{-8} to 100% -1 to 7 DPM 10 to 200%	System cabinets ERCS computer

Table 9 Instruments Available for Monitoring Major Systems

MEASURED PARAMETER	TYPE OF READOUT	INDICATOR RANGE	INDICATOR LOCATION
4. <u>RC System</u>			
Hot Leg Temperature	Linear scale recorder, Computer output	50-700°F	Hot Shutdown Panel Main Control Boards ERCS Computer
Cold Leg Temperature	Linear scale recorder ERCS Computer	50-700°F	Hot Shutdown Panel Main Control Boards ERCS Computer
Subcooling Temperature and pressure	Digital Scale	Variable	ERCS Computer Inadequate Core Cooling Monitoring Cabinet ERCS SAS Display
Avg. Temperature	Linear scale indicator recorder, Computer output	520-620°F	Main Control Boards ERCS Computer ERCS SAS Display
Temp. Difference (Delta T)	Linear scale indicator, recorder Computer output	0-150%	Main Control Boards ERCS Computer
Pressure	Linear scale indicator, recorder Computer output	0-3000 psig	Main Control Boards Hot Shutdown Panel ERCS Computer
Low Range Pressure	Linear scale recorder	0-750 psig	Main Control Boards
Flow	Linear scale indicator, Computer output	0-110% rated flow	Main Control Boards ERCS Computer

Table 9 Instruments Available for Monitoring Major Systems

	MEASURED PARAMETER	TYPE OF READOUT	INDICATOR RANGE	INDICATOR LOCATION
5.	Pressurizer			
	Level (cold)	Linear scale indicator, Computer output	0-100%	Main Control Boards Hot Shutdown Panel ERCS Computer
	Level	Linear scale indicator, Recorder, Computer output, Annunciator	0-100%	Main Control Boards ERCS Computer
	Temperature (Vapor temperature and liquid temperature)	Linear scale indicator, Computer output	0-700°F	Main Control Boards ERCS Computer
6.	RWST			
	Level	Linear scale indicator, Computer output, Annunciator	0-100%	Main Control Boards ERCS Computer
	Valve Status	Indicator light	----------	Main Control Boards
7.	Steam Generator			
	Narrow Range Level	Linear scale indicator, Recorder, Annunciator	0-100%	Main Control Boards ERCS Computer
	Wide Range Level	Linear scale indicator, Computer output	0-100%	Main Control Boards Hot Shutdown Panel ERCS Computer
	Pressure	Linear scale indicator, Computer output	0-1400 psig	Main Control Boards Hot Shutdown Panel ERCS Computer
8.	Station Electric			
	Distribution (Safeguards AC and DC)	Linear scale indicator, Indicator light, Computer output	0-5000 Volts 0-600 Volts	Main Control Boards ERCS Computer

Table 9 Instruments Available for Monitoring Major Systems

MEASURED PARAMETER	TYPE OF READOUT	INDICATOR RANGE	INDICATOR LOCATION
9. Aux. FW Status			
Flow	Linear scale indicator, Computer output	0-250 gpm	Main Control Boards Hot Shutdown Panel ERCS Computer
Pressure	Linear scale indicator, Computer output	0-2000 psig	Main Control Boards Hot Shutdown Panel ERCS Computer
10. Containment Vessel			
Pressure	Linear scale indicator, Computer output, Annunciator, Recorder	0-60 psia 0-30 psia	Main Control Boards ERCS Computer
Post-Accident Radiation (R-48) (Containment)	Log scale indicator, Computer output	$1\text{-}10^8$R/hr	System cabinets ERCS Computer
Water Level	Linear scale indicator	(Sump B) 0-100% (Containment) 0-12 ft	Main Control Boards
Isolation Status	Indicator light, Computer output	-------------	Main Control Boards ERCS Computer
Temperature	Computer output,	0-400°F	ERCS Computer
Air Recirc. Fan Status	Indicator light, Computer output	-------------	Main Control Boards ERCS Computer
Air Cooling System Status	Indicator light	-------------	Main Control Boards
Air Cooling Flow Status	Indicator light, Computer output	-------------	Main Control Boards ERCS Computer
Spray Pump & Valve Status	Indicator light, Computer output	-------------	Main Control Boards ERCS Computer

Table 9 Instruments Available for Monitoring Major Systems

MEASURED PARAMETER	TYPE OF READOUT	INDICATOR RANGE	INDICATOR LOCATION
11. <u>Safety Injection</u>			
Flow	Linear scale indicator, Computer output	0-500 gpm 0-1000 gpm	Main Control Boards ERCS Computer
Pump & Valve Status	Indicator light, Computer output	-------------	Main Control Boards ERCS Computer
12. <u>Resident Heat Removal</u>			
Flow (RHR)	Linear scale indicator, Computer output	0-3000 gpm 0-6000 gpm	Main Control Boards ERCS Computer
Pressure	Linear scale indicator	0-750 psig	Main Control Boards
Pump & Valve Status	Indicator light, Computer output	----------	Main Control Boards ERCS Computer
Emerg. Sump Valve Status	Indicator light, Computer output	____________	Main Control Boards ERCS Computer
Decay Heat Pump Suction Temperature	Linear scale recorder, Annunciator	50-400°F	Main Control Boards ERCS Computer
Decay Heat Cooler Outlet Temperature	Linear scale recorder, Annunciator	50-400°F	Main Control Boards ERCS Computer

Table 9 Instruments Available for Monitoring Major Systems

MEASURED PARAMETER	TYPE OF READOUT	INDICATOR RANGE	INDICATOR LOCATION
13. Accumulator			
Accumulator Pressure	Linear scale indicator, Computer output printout Annunciator	0-800 psig	Main Control Boards ERCS Computer
Accumulator Level	Linear scale indicator, Annunciator	0-100%	Main Control Boards ERCS Computer
Valve Status	Indicator light, Annunciator	----------	Main Control Boards
14. Emergency Ventilation System			
Fan & Damper Status	Indicator light, Computer output	-----------	Main Control Boards ERCS Computer
15. Reactor Vessel Level Instrument System	Digital Scale	Variable	Inadequate Core Cooling Monitoring Cabinet Computer

Table 9 Instruments Available For Monitoring Major Systems

HOT SHUTDOWN PANEL AREA INDICATIONS

1.	Steam Generator A Level (wide range)	0-100%
2.	Steam Generator B Level (wide range)	0-100%
3.	Pressurizer (cold) Level	0-100%
4.	Steam Generator A Pressure	0-1400 psig
5.	Steam Generator B Pressure	0-1400 psig
6.	Letdown Valve Status and Control	Indicating Lights
7.	Auxiliary Feedwater Control and Status	Indicating Lights
8.	Charging Pump Control and Status	Indicating Lights
9.	Reactor Coolant System Hot Leg Temperature	50-700°F
10.	Reactor Coolant System Cold Leg Temperature	50-700°F
11.	Wide Range Reactor Coolant System Pressure	0-3000 psig
12.	Pressurizer Heater Control and Status	Indicating Lights
13.	Boric Acid Transfer Pump Control and Status	Indicating Lights
14.	Neutron Flux Level	$0.1\text{-}10^5$ cps
15.	Steam Generator PORV	Controls

7.3.2 Facilities and Equipment for Offsite Monitoring

A. Meteorological

Several locations, exterior to the plant site, can be used to obtain offsite meteorological conditions. Locations and outputs are summarized in Table 10.

B. Assessment Equipment

1. The EOF Count Room contains a GEM detector system and Geiger-Mueller counter to analyze offsite samples.

2. The emergency lockers in the Assembly Points have the equipment necessary to collect and analyze air samples (particulate and iodine) and portable instruments for measuring radiation levels.

3. The hospital emergency kit at Mayo Clinic Health System has instruments for measuring radiation levels and contamination levels of radiation casualties arriving at the medical center for medical treatment.

4. All Monticello Nuclear Plant counting room and portable radiation detection equipment is available for analysis of samples from Prairie Island NGP.

5. There are TLD badges and airborne particulate and iodine sampling stations installed in areas surrounding the plant. The badges and air sampling stations are installed as part of the Radiation Environmental Monitoring Program. During an emergency, these badges and/or air sampling filters or cartridges may be used for dose assessment purposes.

6. All onsite portable equipment and count room equipment at Prairie Island NGP may be used for required offsite radiation surveys or analysis of offsite samples (liquid or airborne).

Table 10 Offsite Meteorological Equipment

1. Lock and Dam #3
   - (a) Temperature
   - (b) Wind Direction
   - (c) Wind Speed

NOTE:	Meteorological information from Lock and Dam #3 is available on a twenty-four hour per day basis.

2. National Weather Service Twin Cities

   Local Area
   - (a) Temperature
   - (b) Wind Direction
   - (c) Wind Speed
   - (d) Stability Class

7.4 Protective Facilities and Equipment

7.4.1 Assembly Points

The primary protective facility for onsite personnel is the evacuation to an assembly point. Either the Distribution Center or the North Warehouse may be used for an assembly point depending on wind direction. The Emergency Director **SHALL** designate which one is to be used.

The assembly area emergency locker contains equipment that will be used for personnel contamination checks, personnel decontamination, radiation detection equipment to assess conditions at the assembly area and communication equipment for contact with the Emergency Director.

7.4.2 Operational Support Center

The Operational Support Center locker contains all the equipment necessary for reentry into the plant. This includes protective clothing, respiratory protection, monitoring devices, and radiation meters. Air sampling and contamination survey equipment is available for onsite surveys. Decontamination and first aid equipment is available for treatment of onsite personnel.

7.4.3 Emergency Operations Facility

The EOF can be designated as an alternate assembly area. Facilities are available for gathering personnel into a specific area. An emergency locker contains equipment necessary for determining personnel contamination and for decontamination of individuals. A decontamination shower and retention system is available for collection of contaminated waste. A spare Field Survey Team Equipment Kit is located at the EOF.

Communication equipment (radio and telephone) is available for contacting emergency personnel both onsite and offsite.

7.4.4 Mayo Clinic Health System

Mayo Clinic Health System has the equipment required to handle medical emergencies complicated by radioactive contamination. Monitoring equipment, decontamination materials and waste storage (solid and liquid) are available.

7.4.5 Red Wing Fire Station

DLR's and Electronic dosimeters (SRD's) will be issued to Emergency responders eg. when they arrive on site before entering the Protected Area.

All dosimeters and DLR's are maintained by plant personnel.

7.4.6 Technical Support Center Emergency Locker

The Technical Support Center emergency locker contains the necessary survey instruments, dosimetry and protective clothing to allow reentry or access into the plant during emergency conditions.

7.5 First Aid and Medical Facilities

First Aid Kits are available at various emergency lockers in the plant. Any injury requiring medical treatment will be treated at the local medical center. All medical support is covered by Section F4 of the Operations Manual, Medical Support and Casualty Care.

7.6 Damage Control Equipment and Supplies

The maintenance area has a completely supplied machine shop with equipment necessary to machine all but the largest pieces of equipment, (e.g., turbine rotors). One shop area, located in the Auxiliary Building, is for contaminated items. The other shop, located in the Service Building, is for non-contaminated items.

8.0 MAINTAINING EMERGENCY PREPAREDNESS

8.1 Organizational Preparedness

8.1.1 Emergency Response Training

To achieve and maintain an acceptable level of emergency preparedness, training **SHALL** be conducted for members of the on-site Emergency Response Organization in accordance with the Prairie Island Nuclear Generating Plant Emergency Plan Training Program.

Training for all on-site Emergency Response Organization members consists of a review of the Emergency Plan in the form of a general overview. In addition to Emergency Plan overview training, personnel assigned key on-site emergency response positions **SHALL** receive training specific to their position.

Key Emergency Response Organization members **SHALL** receive Emergency Plan training on an annual basis.

Monticello & Prairie Island offsite support will make provisions for the training of those off-site organizations who may be called upon to provide assistance in the event of an emergency.

8.1.2 Exercises, Drills, and Tests

The conduct of periodic drills and exercises are conducted in accordance with the guidance provided in FP-EP-WI-14, Emergency Preparedness Drill and Exercise Manual and FP-EP-WI-24, Emergency Preparedness Drill and Exercise Objectives.

A. Exercises

Exercises which test the integrated capability and a major portion of the basic elements existing within the Emergency Plan **SHALL** be conducted at least every 2 years. This exercise may be included in the full participation biennial exercise which tests the offsite emergency plans.

B. Drills

Drills are supervised instructional periods aimed at testing, developing and maintaining skills in a particular operation and are a part of the continuous training program.

In order to ensure that adequate emergency response capabilities are maintained during the interval between biennial exercises, drills **SHALL** be conducted including at least one drill, during the off exercise year, involving a combination of some of the principal functional areas of the onsite emergency response capabilities. The principal functional areas of emergency response include activities such as management and coordination of emergency response, accident assessment, protective action decision making, and plant system repair and corrective actions. During these drills, activation of all of the Emergency Plan’s response facilities (TSC, OSC, and EOF) would not be necessary, opportunities to consider accident management strategies would be given, supervised instruction would be permitted, operating staff would have the opportunity to resolve problems (success paths) rather than have controllers intervene, and the drills could focus on onsite training objectives.

Drills **SHALL** be conducted in the following areas at the designated minimal frequency. Additional drills may be scheduled by plant management if dictated by response of personnel to previous drills.

1. Fire

   Fire drills **SHALL** be conducted in accordance with Prairie Island Administrative Work Instructions (AWIs) and/or the NSPM's Quality Assurance Topical Report.

2. Medical Emergency

   Medical emergency drills involving the transport of a simulated contaminated individual causing the participation of local support agencies **SHALL** be conducted annually.

3. Radiological

   The periodic radiological and health physics drills described below may be conducted as part of the annual Radiation Protection Specialist continuing training program in the form of walkthroughs or job performance measured activities. These drills may also be conducted as part of an annual plant wide full scale drill or facility drill.

   a Health Physics Drills which involve response to, and analysis of, simulated elevated airborne and/or liquid samples and direct radiation measurements in the environment **SHALL** be conducted semi-annually.

   b Radiological monitoring drills which include the collection and analysis of environmental samples for the purpose of ground deposition assessment **SHALL** be conducted annually.

   c Post accident sampling drills which include the analysis of in-plant liquid samples (with simulated elevated radiation levels) including the use of the Post Accident Sampling System (PASS) **SHALL** be conducted annually.

4. Security

   Hostile Action Drills will be conducted to verify readiness to mitigate after a terrorist event. These drills will be conducted in accordance with FP-EP-WI-14 and FP-EP-WI-24.

5. Emergency Organization Augmentation

   Semi-annual Emergency Organization Augmentation Drills are conducted to provide an ongoing verification that the emergency organization can augment the shift organization in a timely fashion.

C. Tests

A test is a functional test of equipment to verify that the equipment is operable.

1. Communications with state, local and tribal governments within the plume exposure pathway **SHALL** be tested monthly.

2. Communications with Federal response organizations and State governments within the ingestion pathway **SHALL** be tested quarterly.

3. Communications between Prairie Island, Minnesota and Wisconsin Emergency Operating Centers and all local Emergency Operations Centers, and radiation monitoring teams **SHALL** be checked annually.

4. Communication from the Control Room, TSC and EOF to the NRC Operations Center **SHALL** be tested monthly.

5. The Emergency Response Data System (ERDS) **SHALL** be tested on a quarterly basis.

6. The fixed siren portion of Public Alert and Notification System (PANS) **SHALL** be tested and verified operational on a weekly and monthly basis.

NOTE:	These communication tests SHALL be used not only to check the equipment operation but also that the various phone numbers and links are correct and 2-way communication can be established.

8.2 Review and Updating of the Plan and Procedures

The Plant Manager has authority and responsibility for the Prairie Island Emergency Plan and the Emergency Plan Implementing Procedures.

The Plant Manager has the responsibility for the development and updating of the Emergency Plan, the Emergency Plan Implementing Procedures and coordination of the plan with offsite response organizations.

The Emergency Plan will be reviewed on an annual basis to ensure it is current according to the plant's controlled procedure program. The update will take into account changes identified during drills and exercises.

Quarterly, all telephone numbers contained in the Emergency Plan Implementing Procedures **SHALL** be verified correct and updated as a result of the required communication tests.

8.2.1 Organization of Plan

The organization of the Emergency Plan is reviewed and updated yearly by the Emergency Preparedness Manager. Reorganization may be necessary as the result of the following:

A. drills or exercises indicating need for changes

B. changes in key personnel

C. changes in the plant's organization structure

D. changes in the organization of offsite response agencies

E. experience gained under actual emergency situations

8.2.2 Maintenance and Inventory of Emergency Equipment and Supplies

Radiation protection equipment at each of the emergency facilities is checked monthly for operability according to surveillance and testing program.

Emergency plan portable radiation instruments **SHALL** receive a Channel Check and Channel Operational Test monthly and a Channel Calibration at least every 24 months. If any emergency plan portable radiation instrument is found inoperable, then immediate actions **SHALL** be initiated to restore operability or replacement.

All supplies are inventoried quarterly and dated equipment and material are periodically replaced according to surveillance and testing program.

9.0 RECOVERY

In general, the plant will be responsible for the short term recovery, that is recovery from an emergency condition in which no core damage or serious release of radioactivity to the environments has occurred.

If it is clear that a high potential exists for core damage and/or a serious release of radioactivity to the environment, a Recovery Phase will be activated to provide for the long-term recovery actions and for establishing support arrangements.

In general, before re-occupying buildings after an emergency, certain recovery criteria must be satisfied: (1) There must be assurance that the problem encountered is solved and that this same incident cannot immediately recur; (2) The general occupancy areas must be free of significant contamination; (3) Radiation areas and High Radiation areas must be properly defined; and (4) Airborne radioactivity must be eliminated or controlled.

9.1 Investigation of Incidents

All incidents **SHALL** be investigated in conjunction with corporate event response procedures.

9.2 Recovery Procedures

All recovery operations **SHALL** be performed in accordance with written procedures. These procedures **SHALL** include the following activities:

A. Investigation of the cause of the incident

B. Investigation of plant conditions following an accident

C. Repair and restoration of facilities

D. Testing and startup of restored facilities

Methods for determining the extent of radioactive contamination and general protective measures to be taken for personnel performing recovery operations are established in Section F2, Radiation Safety, of the Operations Manual, and in the Radiation Protection Manual, RPIP’s.

Written procedures for recovery of the facility from the specific post accident conditions will be prepared by qualified plant staff members and submitted to the Plant Operating Review Committee. Plant Operating Review Committee approval of all such procedures is required prior to their initiation.

9.3 Criteria for Resumption of Operations

If the plant is shutdown as the result of an emergency, it will be restarted only when:

A. The conditions which caused the emergency are corrected.

B. The cause of the emergency is understood.

C. Restoration, repair and testing is completed as required.

D. No unreviewed safety questions exist.

E. All conditions of the license and technical specifications are satisfied.

9.4 Transition to Recovery

If it is clear that extensive plant damage exists and contamination of plant systems have occurred, then a recovery phase may be necessary.

Transition to the recovery phase will take place in an incremental manner as the functions change from operational to engineering/construction. The decision to make the transition from the emergency phase to the recovery phase should be a joint decision by the ED and EM. The Recovery Manager should possess the qualifications of an Emergency Manager. This position should be occupied by personnel representing the company executive level.

Should transition to the recovery phase become necessary, the site engineering/construction staff would provide the nucleus of the organization.

This plant staff would be augmented as required by specialists from the site organization and the offsite support groups. In addition, appropriate assistance would be secured from the Architect-Engineer and the NSSS vendor organizations. This support could be broadened as required by consultant help from the several organizations familiar with Prairie Island NGP's organization. The overall organization envisioned for a substantial Recovery Phase would be a blend of site staff, and appropriate vendor and consultant personnel. On a prior basis it is counterproductive to define in detail the extensive organization that might be involved in a sizable Recovery Phase because of the unlimited variation of conditions that could result from plant emergencies. However, the nucleus organization has been identified together with guidelines on how the organization might be expanded to meet the requirements demanded at the time.

When the Emergency Manager and Emergency Director agree that the onsite emergency condition has been terminated, a complete transfer of the responsibilities for offsite support may be made to the Recovery Organization. The EOF would then become the Recovery Center and function as Command Center for the Recovery Organization and the recovery effort. Details of Recovery Organization activation and implementing criteria are contained in the Emergency Plan Implementing Procedures.

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Attachment A Emergency Plan Implementing Procedures

A.1 PLANT EMERGENCY PLAN IMPLEMENTING PROCEDURES

The following is a listing of F3 procedures (Emergency Plan Implementing Procedures) which **SHALL** be used by plant emergency organization personnel to implement the emergency plan. This may not be a complete detailed procedure list but is meant to serve as a basis for procedure development.

Procedure Number	Procedure Title	Affected Plan Section
F3-1	Onsite Emergency Organization	5.3, 5.4
F3-2	Classifications of Emergencies	4.0
F3-3	Responsibilities During a Notification of Unusual Event	6.1, 6.3, 6.7
F3-4	Responsibilities During an Alert, Site Area or General Emergency	6.1, 6.2, 6.3, 6.4, 6.5, 6.6, 6.7
F3-5	Emergency Notifications	5.3.3(D), 6.1.2, 6.1.3, 6.1.4, 7.2
F3-5.1	Switchboard Operator Duties	6.1.2, 6.1.3, 6.1.4
F3-5.2	Response to False Siren Activation	6.1.2
F3-5.3	Deleted	
F3-6	Activation and Operation of Technical Support Center	5.3, 7.1.1
F3-7	Activation and Operation of Operational Support Center	5.3, 7.1.2
F3-8	Recommendations for Offsite Protective Actions	6.4, 6.6
F3-8.1	Deleted	
F3-9	Emergency Evacuation	6.6.1, 7.4.1

Attachment A Emergency Plan Implementing Procedures

Procedure Number	Procedure Title	Affected Plan Section
F3-10	Personnel Accountability	6.6.1
F3-11	Search and Rescue	6.6.1, 6.7.1
F3-12	Emergency Exposure Control	6.6.1, 6.7.1
F3-13	Offsite Dose Calculations	6.4, 6.6
F3-13.0	Deleted	
F3-13.1	Rad & Met Data for Dose Projections	6.4, 6.6, 7.3
F3-13.2	Deleted	
F3-13.3	Manual Dose Calculations	6.4, 6.6
F3-13.4	Deleted	
F3-13.5	Alternate Meteorological Data	6.4, 6.6, 7.3
F3-13.6	Weather Forecasting Information	6.4, 6.6, 7.3
F3-14.1	Onsite Radiological Monitoring	6.4, 7.3
F3-14.2	Deleted	
F3-15	Responsibilities of the Radiation Survey Teams During a Radioactive Airborne Release	6.4.2
F3-16	Responsibilities of the Radiation Survey Teams During a Radioactive Liquid Release	6.4.2
F3-17	Core Damage Assessment	6.4.1
F3-17.1	Core Damage Determination	
F3-17.2	Long Term Cooling	
F3-18	Thyroid Iodine Blocking Agent (Potassium Iodide)	6.6.2

Attachment A Emergency Plan Implementing Procedures

Procedure Number	Procedure Title	Affected Plan Section
F3-19	Personnel and Equipment Monitoring and Decontamination	6.6.3, 6.7.2
F3-20	Determination of Radioactive Release Concentrations	6.4, 7.3
F3-20.1	Determination of Steam Line Dose Rates	6.4, 7.3
F3-20.2	Determination of Vent Stack Dose Rates	6.4, 7.3
F3-21	Establishment of a Secondary Access Control Point	6.6.3
F3-22	Deleted	
F3-23	Emergency Sampling	7.3
F3-23.1	Emergency Hotcell Procedure	7.3
F3-23.2	Deleted	
F3-23.3	Deleted	
F3-23.4	Deleted	
F3-23.5	Deleted	
F3-23.6	Deleted	
F3-23.7	Deleted	
F3-23.8	Deleted	
F3-24	Record Keeping During an Emergency	6.2
F3-25	Re-Entry	7.3, 9.0
F3-26	Deleted	

Attachment A Emergency Plan Implementing Procedures

Procedure Number	Procedure Title	Affected Plan Section
F3-26.1	Operation of the ERCS Display	7.1, 7.3
F3-26.2	Radiation Monitor Data on ERCS	7.1, 7.3
F3-26.3	ERDS – NRC Data Link	6.4, 7.2.2.E
F3-27	Response to Railroad Grade Crossing Blockage	5.6.4(D)
F3-28	Deleted	
F3-29	Emergency Security Procedures	5.3, 6.1
F3-30	Transition to Recovery	9.0
F3-31	Response to Security Related Threats	5.3, 6.1

Attachment A Emergency Plan Implementing Procedures

A.2 EOF EMERGENCY PLAN IMPLEMENTING PROCEDURES

The following is a listing of F8 procedures (EOF Emergency Plan Implementing Procedures) which **SHALL** be used by EOF emergency organization personnel to implement the emergency. This may not be a complete detailed procedure list but is meant to serve as a basis for procedure development.

Procedure Number	Procedure Title	Affected Plan Section
F8-1	Emergency Operations Facility Organization	5.4, 5.5, 5.6
F8-2	Deleted	
F8-3	Activation and Operation of the EOF	5.4, 7.1.3
F8-4	Emergency Support and Logistics	5.6, 5.7
F8-5	Offsite Dose Assessment and Protective Action Recommendations	6.4, 6.6
F8-6	Radiological Monitoring and Control at the EOF	6.6, 6.7, 7.4
F8-8	Offsite Agency Liaison Activities	5.6, 5.7
F8-9	Event Termination or Recovery	5.5, 9.0
F8-10	Record Keeping in the EOF	6.2
F8-11	Transfer to the Backup EOF	7.1.3
F8-12	Emergency REMP	7.3.2

Attachment B Summary of Emergency Supplies

1. Old Receiving Warehouse Locker and Emergency Vehicle Field Survey Kits

   A. Beta Gamma Survey Meters

   B. Offsite Sample Kits (2)

      1. Airborne Sample Equipment (Particulate, Iodine, Gaseous)

      2. Liquid Sample Equipment

   C. Personnel Dosimetry

   D. Portable Communication Radios

   E. Foul weather gear

   F. Protective clothing

   G. Potassium Iodide Potassium Iodide (KI)

2. North Warehouse and Distribution Center Assembly Points (each location)

   A. Beta Gamma Survey Meters

   B. Portable Communications Radio

   C. 1 Copy of Emergency Plan Implementing Procedures (F-3)

   D. Personnel Decontamination Kit

   E. Airborne Sample Equipment

   F. Small First Aid Kit

   G. Area Radiation Monitor

   H. Protective Clothing

Attachment B Summary of Emergency Supplies

3. Operational Support Center

   A. Beta Gamma Survey Meters
   B. Air Sampling Equipment (Battery and AC Powered)
   C. Personnel Dosimetry
   D. Portable Communications Radios (located in Control Room)
   E. Area Radiation Monitor
   F. Portable Lanterns and Batteries
   G. Copies of Emergency Plan Implementing Procedures (F-3)
   H. Plant Floor Plans
   I. Protective Clothing (Including Waterproof)
   J. Respiratory Protection (SCBA's) and spare bottles
   K. First Aid Kit
   L. Continuous Air Monitor (Control Room and OSC)
   M. Drager Toxic Chemical Air Sampler
   N. Full Face Respirators and Iodine Canisters

4. Technical Support Center

   A. Beta Gamma Survey Meters
   B. Airborne Sampling Equipment
   C. Personnel dosimetry
   D. Portable Communications Radio
   E. Area Radiation Monitor

Attachment B Summary of Emergency Supplies

F. Continuous Air Monitor

G. Copies of Emergency Plan Implementing Procedures (F-3)

H. Protective Clothing

I. Respiratory Protection (SCBA's) and spare bottles

J. Plant Floor Plans

K. Potassium Iodide (KI) Distribution

5. <u>Mayo Clinic Health System</u>

A. Beta Gamma Survey Meters

B. Personnel dosimetry

C. Copy of Operations Manual F-4

D. Supplies (Disposable clothing, solid waste containers, and liquid waste containers)

6. <u>Hot Cell</u>

A. Beta Gamma Survey Meters

B. Protective clothing

C. Alpha Survey Meter

D. Sample team communication gear

E. Copy of Emergency Plan Implementing Procedure (F-3)

Attachment B Summary of Emergency Supplies

7. Fire Brigade Dress Out Area

   A. Self-Contained Breathing Apparatus

8. Emergency Operations Facility

   A. Beta Gamma Survey Meters

   B. Offsite Sample Kit (1)

      1. Airborne Sampling Equipment (Particulate, Iodine, Gaseous)

      2. Liquid Sampling Equipment

   C. Personnel Dosimetry

   D. Airborne Sampling Equipment (Local)

   E. Portable Communication Radios

   F. Personnel Decontamination Kit

   G. Area Radiation Monitor

   H. Continuous Air Monitor

   I. GEM detector for Isotopic Analysis of Samples

   J. Decontamination Shower

   K. Potassium Iodide (KI) Distribution

Attachment C NUREG-0654/PI E-Plan Cross Reference

NUREG-0654 ELEMENT	PI E-PLAN REFERENCE (by Section)
A. Assignment of Responsibility (Organization Control)	
A.1.a	5.6, 5.7
A.1.b	5.6, 5.7
A.1.c	Figure 3
A.1.d	1.5, 1.6, 5.3.1, MT & PI Offsite Plan
A.1.e	5.2, 5.3, 5.3.1, 5.3.3, Table 1
A.2.a	State/Local Plans
A.2.b	State/Local Plans
A.3	MT & PI Offsite Plan
A.4	5.3.1
B. Onsite Emergency Organization	
B.1	5.1, 5.2, 5.3, 5.4
B.2	5.3.1
B.3	5.3.1
B.4	5.3.1
B.5	5.3.2, 5.3.3, 5.3.4, Table 1, 5.4
B.6	5.6, 5.7, Figure 4
B.7	Table 1, 5.4, MT & PI Offsite Plan
B.7.a	5.4, MT & PI Offsite Plan
B.7.b	5.4, 5.5
B.7.c	5.4, MT & PI Offsite Plan
B.7.d	MT & PI Offsite Plan
B.8	5.6.3, MT & PI Offsite Plan
B.9	5.6.4, 5.7, 6.7.3, MT & PI Offsite Plan

Attachment C NUREG-0654/PI E-Plan Cross Reference

NUREG-0654 ELEMENT	PI E-PLAN REFERENCE (by Section)
C. Emergency Response Support and Resources	
C.1.a	5.3.1
C.1.b	MT & PI Offsite Plan
C.1.c	MT & PI Offsite Plan
C.2.a	State/Local Plan
C.2.b	MT & PI Offsite Plan
C.3	Attach A.2, (F8-4), MT & PI Offsite Plan
C.4	Attach A.2 (F8-4), 5.6, MT & PI Offsite Plan
D. Emergency Classification System	
D.1	4.0
D.2	4.0, Annex A
D.3	State/Local Plan
D.4	State/Local Plan
E. Notification Methods and Procedures	
E.1	6.1.2, 6.1.4
E.2	6.1.1
E.3	5.3.1
E.4	5.3.1
E.4.a	5.3.1
E.4.b	5.3.1
E.4.c	5.3.1
E.4.d	5.3.1
E.4.e	5.3.1

Attachment C NUREG-0654/PI E-Plan Cross Reference

NUREG-0654 ELEMENT	PI E-PLAN REFERENCE (by Section)
E. Notification Methods and Procedures (cont'd)	
E.4.f	5.3.1
E.4.g	5.3.1
E.4.h	5.3.1
E.4.i	5.3.1
E.4.j	5.3.1
E.4.k	5.3.1
E.4.l	5.3.1
E.4.m	5.3.1
E.4.n	5.3.1
E.5	7.2.3, State/Local Plan
E.6	5.3.1, 7.2.3, State/Local Plan
E.7	MT & PI Offsite Plan
F. Emergency Communications	
F.1.a	6.1.2, 6.1.3, Table 6
F.1.b	6.1.2, 6.1.3, 7.2.2, Table 6
F.1.c	Table 6
F.1.d	72.2., Table 6
F.1.e	6.1.1, 7.2.2, Table 6
F.1.f	7.2.2, Table 6
F.2	7.2.2
F.3	8.1.2

Attachment C NUREG-0654/PI E-Plan Cross Reference

NUREG-0654 ELEMENT	PI E-PLAN REFERENCE (by Section)
G. Public Education and Information	
G.1	MT & PI Offsite Plan
G.2	MT & PI Offsite Plan
G.3.a	MT & PI Offsite Plan
G.3.b	Attach A (F8-8) MT & PI Offsite Plan
G.4.a	MT & PI Offsite Plan
G.4.b	MT & PI Offsite Plan
G.4.c	MT & PI Offsite Plan
G.5	MT & PI Offsite Plan
H. Emergency Facilities and Equipment	
H.1	7.1.1, 7.1.2
H.2	7.1.3
H.3	State/Local Plan
H.4	6.1.1, MT & PI Offsite Plan
H.5	7.3
H.5.a	7.3.1, Table 7, Table 11
H.5.b	7.3.1, Table 8, Table 9
H.5.c	7.3.1, Table 10
H.5.d	7.3.1
H.6.a	7.3.2, Table 11
H.6.b	7.3.2, MT & PI Offsite Plan
H.6.c	7.3.2, MT & PI Offsite Plan

Attachment C NUREG-0654/PI E-Plan Cross Reference

NUREG-0654 ELEMENT	PI E-PLAN REFERENCE (by Section)
H. Emergency Facilities and Equipment (cont'd)	
H.7	7.3.2, 7.4.3, 7.4.4, 7.4.5, 7.4.6, Attach B.
H.8	6.4, 7 .3.1, 7.3.2, Table 11
H.9	7.1.2, 7.4.2, Attach B
H.10	8.2.2
H.11	Attach B
H.12	7.1.3
I. Accident Assessment	
I.1	Annex A, Table 7 Table 8, Table 10
I.2	6.4, 7.3.1
I.3.a	6.4
I.3.b	6.4.1
I.4	6.4.1
I.5	6. 4.1, 7.3.1, 7.3.2 Table 11
I.6	6.4.1
I.7	6.4.2
I.8	6.4.1, 6.4.2
I.9	6.4.2
I.10	6.4.1
I.11	State Plan

Attachment C NUREG-0654/PI E-Plan Cross Reference

NUREG-0654 ELEMENT	PI E-PLAN REFERENCE (by Section)
J. Protective Response	
J.1.a	6.6.1 (A)
J.1.b	6.6.1 (A)
J.1.c	6.6.1 (A)
J.1.d	6.6.1 (A)
J.2	6.6.1 (A)
J.3	6.6.1 (A)
J.4	6.6.1 (A)
J.5	5.3.1, 6.6.1 (A)
J.6.a	6.6.2 (A)
J.6.b	6.6.2 (A)
J.6.c	6.6.2 (B)
J.7	6.6.2 I Tables 3 and 4
J.8	6.6.1 (D), MT & PI Offsite Plan
J.9	State/Local Plan
J.10.a	6.4.2, MT & PI Offsite Plan
J.10.b	6.6, MT & PI Offsite Plan
J.10.c	7.2.3
J.10.d	State/Local Plan
J.10.e	State/Local Plan
J.10.f	State/Local Plan
J.10.g	State/Local Plan
J.10.h	State/Local Plan
J.10.i	State/Local Plan

Attachment C NUREG-0654/PI E-Plan Cross Reference

NUREG-0654 ELEMENT	PI E-PLAN REFERENCE (by Section)
J. Protective Response [Cont'd]	
J.10.j	State/Local Plan
J.10.k	State/Local Plan
J.10.l	State/Local Plan
J.10.m	6 .6.1 l Tables 3 and 4
J.11	State Plan
J.12	State/Local Plan
K. Radiological Exposure Control	
K.1.a	6.7.1, Table 4
K.1.b	6.7.1, Table 4
K.1.c	6.7.1, Table 4
K.1.d	6.7.1, Table 4
K.1.e	6.7.1, Table 4
K.1.f	6.7.1, Table 4
K.1.g	6.7.1, Table 4
K.2	6.7.1
K.3.a	6.7.1
K.3.b	6.7.1
K.4	State/Local Plan
K.5.a	6.6. 3, Table 5
K.5.b	6.7.2
K.6.a	6.6.3 (A)
K.6.b	6.6.3 (A)
K.6.c	6.6.3 (A)
K.7	6.6.3, 6.7.2

Attachment C NUREG-0654/PI E-Plan Cross Reference

NUREG-0654 ELEMENT	PI E-PLAN REFERENCE (by Section)
L. Medical and Public Health Support	
L.1	5.6.4 (B), 6.7.3
L.2	6.7.2
L.3	State Plan
L.4	5.6.4 I, 6.7.2, 6.7.3
M. Recovery and Re-entry Planning and Post Accident Operations	
M.1	5.5, 9.0
M.2	5.5, 9.4
M.3	5.5,9.4
M.4	6.4.1
N. Exercises and Drills	
N.1.a	8.1.2
N.1.b	8.1.2
N.2.a	8.1.2
N.2.b	8.1.2
N.2.c	8.1.2
N.2.d	8.1.2
N.2.e (1)	8.1.2
N.2.e (2)	8.1.2
N.3.a	Site Drill/Exercise Manual
N.3.b	Site Drill/Exercise Manual
N.3.c	Site Drill/Exercise Manual
N.3.d	Site Drill/Exercise Manual
N.3.e	Site Drill/Exercise Manual

Attachment C NUREG-0654/PI E-Plan Cross Reference

NUREG-0654 ELEMENT	PI E-PLAN REFERENCE (by Section)
N. Exercises and Drills (cont'd)	
N.3.f	Site Drill/Exercise Manual
N.4	Site Drill/Exercise Manual
N.5	Site Drill/Exercise Manual
O. Radiological Emergency Response Training	
O.1	8.1.1
O.1.a	8.1.1
O.1.b	State/Local Plan
O.2	8.1.1, 8.1.2 Site Drill/Exercise Manual
O.3	8.1.1
O.4.a	8.1.1
O.4.b	8.1.1
O.4.c	8.1.1
O.4.d	8.1.1
O.4.e	8.1.1
O.4.f	8.1.1
O.4.g	MT & PI Offsite Plan
O.4.h	8.1.1, MT & PI Offsite Plan
O.4.i	MT & PI Offsite Plan
O.4.j	8.1.1
O.5	8.1.1, MT & PI Offsite Plan

Attachment C NUREG-0654/PI E-Plan Cross Reference

NUREG-0654 ELEMENT	PI E-PLAN REFERENCE (by Section)
P. Responsibility for the Planning Effort: Development, Periodic Review and Distribution of Emergency Plans	
P.1	MT & PI Offsite Plan
P.2	8.2
P.3	8.2
P.4	8.2.1
P.5	MT & PI Offsite Plan
P.6	2.0, MT & PI Offsite Plan
P.7	Attach A
P.8	Table of Contents Attach C
P.9	MT & PI Offsite Plan
P.10	8.2

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EP	ANNEX A	NUMBER: E-PLAN
		REV: 58

EMERGENCY ACTION LEVEL MATRIX

The site-specific Emergency Action Levels (EALs) are presented in the attached Emergency Action Level Matrix. These EALs are based on the NEI 99-01 EAL scheme.

Emergency Plan Implementing Procedure Classification of Emergency also contains the same Emergency Action Level Matrix.

HOT & COLD

Abnormal Rad Release / Rad Effluent

Effluents / Release Rates

GENERAL EMERGENCY

RG1 Release of gaseous radioactivity resulting in offsite dose greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE.

Notes:
- The Emergency Director should declare the General Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RG1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RG1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Reading on **EITHER** of the following radiation monitors greater than the reading shown for 15 minutes or longer:

1R-50 High Range Stack Gas Monitor	45,000 mR/hr
2R-50 High Range Stack Gas Monitor	45,000 mR/hr

RG1.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Dose assessment using actual meteorology indicates doses greater than 1,000 mrem TEDE or 5,000 mrem thyroid CDE at or beyond the site boundary.

RG1.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Field survey results indicate **EITHER** of the following at or beyond the site boundary:
- Closed window dose rates greater than 1,000 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 5,000 mrem for one hour of inhalation.

SITE AREA EMERGENCY

RS1 Release of gaseous radioactivity resulting in offsite dose greater than 100 mrem TEDE or 500 mrem thyroid CDE.

Notes:
- The Emergency Director should declare the Site Area Emergency promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RS1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RS1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Reading on **EITHER** of the following radiation monitors greater than the reading shown for 15 minutes or longer:

1R-50 High Range Stack Gas Monitor	4,500 mR/hr
2R-50 High Range Stack Gas Monitor	4,500 mR/hr

RS1.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Dose assessment using actual meteorology indicates doses greater than 100 mrem TEDE or 500 mrem thyroid CDE at or beyond the site boundary.

RS1.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Field survey results indicate **EITHER** of the following at or beyond the site boundary:
- Closed window dose rates greater than 100 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 500 mrem for one hour of inhalation.

ALERT

RA1 Release of gaseous or liquid radioactivity resulting in offsite dose greater than 10 mrem TEDE or 50 mrem thyroid CDE.

Notes:
- The Emergency Director should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 15 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.
- The pre-calculated effluent monitor values presented in EAL RA1.1 should be used for emergency classification assessments until the results from a dose assessment using actual meteorology are available.

RA1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Reading on **EITHER** of the following radiation monitors greater than the reading shown for 15 minutes or longer.

1R-50 High Range Stack Gas Monitor	450 mR/hr
2R-50 High Range Stack Gas Monitor	450 mR/hr

RA1.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Dose assessment using actual meteorology indicates doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary.

RA1.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Analysis of a liquid effluent sample indicates a concentration or release rate that would result in doses greater than 10 mrem TEDE or 50 mrem thyroid CDE at or beyond the site boundary for one hour of exposure.

RA1.4 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Field survey results indicate **EITHER** of the following at or beyond the site boundary:
- Closed window dose rates greater than 10 mR/hr expected to continue for 60 minutes or longer.
- Analyses of field survey samples indicate thyroid CDE greater than 50 mrem for one hour of inhalation.

NUE

RU1 Release of gaseous or liquid radioactivity greater than 2 times the ODCM limits for 60 minutes or longer.

Notes:
- The Emergency Director should declare the Unusual Event promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.
- If an ongoing release is detected and the release start time is unknown, assume that the release duration has exceeded 60 minutes.
- If the effluent flow past an effluent monitor is known to have stopped due to actions to isolate the release path, then the effluent monitor reading is no longer valid for classification purposes.

RU1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Reading on **ANY** effluent radiation monitor in the table below greater than the listed values for 60 minutes or longer:

Gaseous Effluents		
Unit 1	1R-22 Unit 1 Shield Building Vent Rad Monitor	1×10^4 cpm
	1R-30 Unit 1 Aux Building Vent Rad Monitor	1×10^3 cpm
	1R-37 Unit 1 Aux Building Vent Rad Monitor	1.5×10^3 cpm
Unit 2	2R-22 Unit 2 Shield Building Vent Rad Monitor	1×10^4 cpm
	2R-30 Unit 2 Aux Building Vent Rad Monitor	6×10^3 cpm
	2R-37 Unit 2 Aux Building Vent Rad Monitor	2×10^3 cpm
	R-25 Spent Fuel Pool Vent Rad Monitor	1×10^4 cpm
	R-31 Spent Fuel Pool Vent Rad Monitor	1×10^4 cpm
	R-35 Radwaste Building Vent Rad Monitor	3×10^3 cpm
Liquid Effluents		
Unit 1	1R-19 SG Blowdown Radiation Monitor	2×10^4 cpm
Unit 2	2R-19 SG Blowdown Radiation Monitor	4×10^4 cpm
	R-18 Waste Effluent Liquid Monitor	5×10^4 cpm
	R-21 Circ Water Discharge Monitor	2×10^4 cpm

RU1.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Reading on **ANY** effluent radiation monitor greater than 2 times the calculated limit established by a current radioactivity discharge permit for 60 minutes or longer.

RU1.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Sample analysis for a gaseous or liquid release indicates a concentration or release rate greater than 2 times the ODCM limits for 60 minutes or longer.

Spent Fuel Pool Level / In Plant Rad

GENERAL EMERGENCY

RG2 Spent fuel pool level cannot be restored to at least 729.16' elevation for 60 minutes or longer.

Note: The Emergency Director should declare the General Emergency promptly upon determining that 60 minutes has been exceeded, or will likely be exceeded.

RG2.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Spent fuel pool level cannot be restored to at least 729.16' elevation for 60 minutes or longer.

SITE AREA EMERGENCY

RS2 Spent fuel pool level at 729.16' elevation.

RS2.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Lowering of spent fuel pool level to 729.16' elevation.

ALERT

RA2 Significant lowering of water level above, or damage to, irradiated fuel.

RA2.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Uncovery of irradiated fuel in the REFUELING PATHWAY.

RA2.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Damage to irradiated fuel resulting in a release of radioactivity from the fuel as indicated by **ANY** of the following radiation monitors:

1(2)R-2 Containment Vessel Area Monitor	1×10^{-1} R/hr
1(2)R-12 Containment/SBV Radio Gas Monitor	9.0E+2 cpm
R-5 Fuel Handling Area Monitor	1×10^{-1} R/hr
R-25 Spent Fuel Pool Vent Rad Monitor	2×10^5 cpm
R-29 Shipping and Receiving Area Monitor	1×10^2 mR/hr
R-31 Spent Fuel Pool Vent Rad Monitor	2×10^5 cpm

RA2.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Lowering of spent fuel pool level to 739.16' elevation.

RA3 Radiation levels that impede access to equipment necessary for normal plant operations, cooldown or shutdown.

Note: If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted.

RA3.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Dose rate greater than 15 mR/hr in **EITHER** of the following areas:
- Control Room (R-1)
- Central Alarm Station (R-69)

RA3.2 | 3 | 4 | 5

An UNPLANNED event results in radiation levels that prohibit or impede access to any of the Table H1 plant rooms or areas.

NUE

RU2 UNPLANNED loss of water level above irradiated fuel.

RU2.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

a. UNPLANNED water level drop in the REFUELING PATHWAY as indicated by **ANY** of the following:
- Level less than SFP low water level alarm (752.5 feet elevation)
- Refueling Canal Level (365" on Control Board)
- Visual Observation (752.5 feet elevation)

AND

b. UNPLANNED rise in area radiation levels as indicated by **ANY** of the following radiation monitors:
- R-5 Fuel Handling Area Monitor
- 1(2)R-2 Containment Vessel Area Monitor
- Other Portable Area Radiation Monitoring Instrumentation

GENERAL EMERGENCY | SITE AREA EMERGENCY | ALERT | NUE | HOT & COLD

Hazards

Security

HG1 HOSTILE ACTION resulting in loss of physical control of the facility.

HG1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

a. A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by Security Shift Supervisor.

AND

b. **EITHER** of the following has occurred:

1. **ANY** of the following safety functions cannot be controlled or maintained.
   - Reactivity control
   - Core cooling
   - RCS heat removal

OR

2. Damage to spent fuel has occurred or is IMMINENT.

HS1 HOSTILE ACTION within the PROTECTED AREA.

HS1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

A HOSTILE ACTION is occurring or has occurred within the PROTECTED AREA as reported by Security Shift Supervisor.

HA1 HOSTILE ACTION within the OWNER CONTROLLED AREA or airborne attack threat within 30 minutes.

HA1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

A HOSTILE ACTION is occurring or has occurred within the OWNER CONTROLLED AREA as reported by Security Shift Supervisor.

HA1.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

A validated notification from NRC of an aircraft attack threat within 30 minutes of PINGP.

HU1 Confirmed SECURITY CONDITION or threat.

HU1.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

A SECURITY CONDITION that does not involve a HOSTILE ACTION as reported by Security Shift Supervisor.

HU1.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Notification of a credible security threat directed at PINGP.

HU1.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

A validated notification from the NRC providing information of an aircraft threat.

Natural & Destructive Phenomenon

Table H1		
Mode	**Building**	**Area/Room**
Mode 3	Auxiliary Building	735' General Area
Mode 4	Auxiliary Building	695' General Area
Mode 5	Auxiliary Building	695' General Area

NOTE:
Depending upon the plant mode at the time of the event escalation criteria for HU2 & HU3 would be CA6 or SA9.

HU2 Seismic event greater than OBE levels.

HU2.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Seismic event greater than Operational Basis Earthquake (OBE) as indicated by an "OBE Exceedance" alarm on Seismic Monitoring Panel.

HU3 Hazardous event.

Note: EAL HU3.4 does not apply to routine traffic impediments such as fog, snow, ice, or vehicle breakdowns or accidents.

HU3.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

A tornado strike within the PROTECTED AREA.

HU3.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Internal room or area flooding of a magnitude sufficient to require manual or automatic electrical isolation of a SAFETY SYSTEM component needed for the current operating mode.

HU3.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

Movement of personnel within the PROTECTED AREA is impeded due to an offsite event involving hazardous materials (e.g., an offsite chemical spill or toxic gas release).

HU3.4 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

A hazardous event that results in on-site conditions sufficient to prohibit the plant staff from accessing the site via personal vehicles.

HU3.5 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

High or low river water level occurrences affecting the PROTECTED AREA as indicated by **EITHER** of the following:

- River Intake level greater than 692 ft MSL
- River Intake level less than 669.5 ft MSL

Fire

Table H2	
Building	**Rooms**
Control Building	Control Room
	Control Room Chiller Room
Turbine Building	Battery Rooms
	AFW Rooms
	D1/D2 Diesel Generator Rooms
	Safeguards Switchgear Rooms
	Relay Room
	480 V Switchgear Rooms
	Event Montoring Rooms
D5/D6 Diesel Generator Building	All
Auxiliary Building	All areas within AB Special Vent Zone
Plant Screen House	Safeguards Cooling Water Pump Rooms
	Safeguards Traveling Screen Room
Shield Building/Containment	All

NOTE:
Depending upon the plant mode at the time of the event escalation criteria for HU4 would be CA6 or SA9.

HU4 FIRE potentially degrading the level of safety of the plant.

Note: The Emergency Director should declare the Unusual Event promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

HU4.1 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

a. A FIRE is NOT extinguished within 15-minutes of **ANY** of the following FIRE detection indications:

- Report from the field (i.e., visual observation)
- Receipt of multiple (more than 1) fire alarms or indications
- Field verification of a single fire alarm

AND

b. The FIRE is located within **ANY** of the Table H2 plant rooms or areas.

HU4.2 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

a. Receipt of a single fire alarm (i.e., no other indications of a FIRE).

AND

b. The FIRE is located within **ANY** of the Table H2 plant rooms or areas.

AND

c. The existence of a FIRE is not verified within 30-minutes of alarm receipt.

HU4.3 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

A FIRE within the plant PROTECTED AREA or ISFSI PROTECTED AREA not extinguished within 60-minutes of the initial report, alarm or indication.

HU4.4 | 1 | 2 | 3 | 4 | 5 | 6 | DEF

A FIRE within the plant PROTECTED AREA or ISFSI PROTECTED AREA that requires firefighting support by an offsite fire response agency to extinguish.

Hazards Security Natural & Destructive Phenomenon Fire

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NUE	HOT & COLD
Hazards Continued — Toxic, Corrosive, Asphyxiant or Flammable Gases			**HA5 Gaseous release impeding access to equipment necessary for normal plant operations, cooldown or shutdown.** **Note:** If the equipment in the listed room or area was already inoperable or out-of-service before the event occurred, then no emergency classification is warranted. **HA5.1** (Modes 3, 4, 5) a. Release of a toxic, corrosive, asphyxiant or flammable gas into any of the Table H1 plant rooms or areas. **AND** b. Entry into the room or area is prohibited or impeded.		Toxic, Corrosive, Asphyxiant or Flammable Gases
Hazards Continued — Control Room Evacuation		**HS6 Inability to control a key safety function from outside the Control Room.** **Note:** The Emergency Director should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded. **HS6.1** (Modes 1, 2, 3, 4, 5, 6) a. An event has resulted in plant control being transferred from the Control Room to the remote Hot Shutdown Panels. **AND** b. Control of **ANY** of the following key safety functions is not reestablished within 15 minutes. • Reactivity control (Modes 1, 2, and 3 only) • Core cooling • RCS heat removal	**HA6 Control Room evacuation resulting in transfer of plant control to alternate locations.** **HA6.1** (Modes 1, 2, 3, 4, 5, 6, DEF) An event has resulted in plant control being transferred from the Control Room to the remote Hot Shutdown Panels.		Control Room Evacuation
Hazards Continued — Shift Manager / Emergency Director Judgment	**HG7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a General Emergency.** **HG7.1** (Modes 1, 2, 3, 4, 5, 6, DEF) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or IMMINENT substantial core degradation or melting with potential for loss of containment integrity or HOSTILE ACTION that results in an actual loss of physical control of the facility. Releases can be reasonably expected to exceed EPA Protective Action Guideline exposure levels offsite for more than the immediate site area.	**HS7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a Site Area Emergency.** **HS7.1** (Modes 1, 2, 3, 4, 5, 6, DEF) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which involve actual or likely major failures of plant functions needed for protection of the public or HOSTILE ACTION that results in intentional damage or malicious acts, (1) toward site personnel or equipment that could lead to the likely failure of or, (2) that prevent effective access to equipment needed for the protection of the public. Any releases are not expected to result in exposure levels which exceed EPA Protective Action Guideline exposure levels beyond the site boundary.	**HA7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of an Alert.** **HA7.1** (Modes 1, 2, 3, 4, 5, 6, DEF) Other conditions exist which, in the judgment of the Emergency Director, indicate that events are in progress or have occurred which involve an actual or potential substantial degradation of the level of safety of the plant or a security event that involves probable life threatening risk to site personnel or damage to site equipment because of HOSTILE ACTION. Any releases are expected to be limited to small fractions of the EPA Protective Action Guideline exposure levels.	**HU7 Other conditions exist which in the judgment of the Emergency Director warrant declaration of a NUE.** **HU7.1** (Modes 1, 2, 3, 4, 5, 6, DEF) Other conditions exist which in the judgment of the Emergency Director indicate that events are in progress or have occurred which indicate a potential degradation of the level of safety of the plant or indicate a security threat to facility protection has been initiated. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.	Shift Manager / Emergency Director Judgment

ISFSI ICS/EALS — Cask Confinement Boundary

Table E1

Location	Gamma mrem/hr	Neutron mrem/hr
Center of top protective cover	90	20
Between cask flange and side neutron shield	160	380
Mid-height of side neutron shield	80	70
Between cask bottom and side neutron shield	170	1860

EU1 Damage to a loaded cask CONFINEMENT BOUNDARY.

EU1.1 (Modes 1, 2, 3, 4, 5, 6, DEF)

Damage to a loaded cask CONFINEMENT BOUNDARY as indicated by an on-contact radiation reading on the surface of the spent fuel cask greater than ANY Table E1 value:

Modes:

1	2	3	4	5	6	DEF
Power	Startup	Hot Standby ≥ 350 °F	Hot Shutdown 350 °F > Tavg > 200 °F	Cold Shutdown ≤ 200 °F	Refueling	Defueled

System Malfunctions — HOT

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NUE

Loss of Power

SG1 Prolonged loss of all offsite and all onsite AC power to emergency buses.

Note: The Emergency Director should declare the General Emergency promptly upon determining that 4 hours has been exceeded, or will likely be exceeded.

SG1.1 (1 2 3 4)

a. Loss of **ALL** offsite and **ALL** onsite AC power to both Safeguards Buses 15 and 16 (25 and 26).

AND

b. **EITHER** of the following:
- Restoration of at least one AC emergency bus in less than 4 hours is not likely.
- Core Cooling CSF – RED

SG8 Loss of all AC and Vital DC power sources for 15 minutes or longer.

Note: The Emergency Director should declare the General Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SG8.1 (1 2 3 4)

a. Loss of **ALL** offsite and **ALL** onsite AC power to both Safeguards Buses 15 and 16 (25 and 26) for 15 minutes or longer.

AND

b. Indicated voltage is less than 111 VDC on both 125 VDC Panels 11 and 12 (21 and 22) for 15 minutes or longer.

SS1 Loss of all offsite and all onsite AC power to emergency buses for 15 minutes or longer.

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SS1.1 (1 2 3 4)

Loss of **ALL** offsite and **ALL** onsite AC power to both Safeguards Buses 15 and 16 (25 and 26) for 15 minutes or longer.

SS8 Loss of all Vital DC power for 15 minutes or longer.

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SS8.1 (1 2 3 4)

Indicated voltage is less than 111 VDC on both 125 VDC Panels 11 and 12 (21 and 22) for 15 minutes or longer.

SA1 Loss of all but one AC power source to emergency buses for 15 minutes or longer.

Note: The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SA1.1 (1 2 3 4)

a. AC power capability to both Safeguards Buses 15 and 16 (25 and 26) is reduced to a single power source (Table S1) for 15 minutes or longer.

AND

b. Any additional single power source failure will result in a loss of all AC power to SAFETY SYSTEMS.

Table S1	
Unit 1	**Unit 2**
1R Transformer	2RY Transformer
CT11 Transformer	CT12 Transformer
D1 Diesel Generator	D5 Diesel Generator
D2 Diesel Generator	D6 Diesel Generator
Aligned Available Cross-Ties	

SU1 Loss of all offsite AC power capability to emergency buses for 15 minutes or longer.

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SU1.1 (1 2 3 4)

Loss of **ALL** offsite AC power capability (Table S2) to both Safeguards Buses 15 and 16 (25 and 26) for 15 minutes or longer.

Table S2	
Unit 1	**Unit 2**
1R Transformer	2RY Transformer
CT11 Transformer	CT12 Transformer

Control Room Indications

SA2 UNPLANNED loss of Control Room indications for 15 minutes or longer with a significant transient in progress.

Note: The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SA2.1 (1 2 3 4)

a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.

Reactor Power
Pressurizer Level
RCS Pressure
Core Exit Temperature
Level in at least one steam generator
Auxiliary Feed Water Flow

AND

b. **ANY** of the following transient events in progress.
- Automatic or manual runback greater than 25% thermal reactor power
- Electrical load rejection greater than 25% full electrical load
- Reactor trip
- ECCS (SI) actuation

SU2 UNPLANNED loss of Control Room indications for 15 minutes or longer.

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SU2.1 (1 2 3 4)

a. An UNPLANNED event results in the inability to monitor one or more of the following parameters from within the Control Room for 15 minutes or longer.

Reactor Power
Pressurizer Level
RCS Pressure
Core Exit Temperature
Level in at least one steam generator
Auxiliary Feed Water Flow

Reactor Coolant System

SU3 Reactor coolant activity greater than Technical Specification allowable limits.

SU3.1 (1 2 3 4)

1(2)R-9 reading greater than 1.0 R/hr.

SU3.2 (1 2 3 4)

Sample analysis indicates that a reactor coolant activity value is greater than an allowable limit specified in Technical Specification 3.4.17 Condition C (Dose Equivalent I-131 > 30 μCi/gm).

SU4 RCS leakage for 15 minutes or longer.

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

SU4.1 (1 2 3 4)

RCS unidentified or pressure boundary leakage greater than 10 gpm for 15 minutes or longer.

SU4.2 (1 2 3 4)

RCS identified leakage greater than 25 gpm for 15 minutes or longer.

SU4.3 (1 2 3 4)

Leakage from the RCS to a location outside containment greater than 25 gpm for 15 minutes or longer.

		GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NUE			HOT
System Malfunctions Continued	ATWS		**SS5 Inability to shutdown the reactor causing a challenge to core cooling or RCS heat removal.** **Note:** Heat Sink CSF should not be considered RED if total AFW flow is less than 200 gpm due to operator action. **SS5.1** [1] a. An automatic or manual trip did not reduce reactor power to less than 5%. **AND** b. All manual actions to reduce reactor power to less than 5% have been unsuccessful. **AND** c. **EITHER** of the following conditions exist: • Core Cooling CSF - RED • Heat Sink CSF - RED	**SA5 Automatic or manual trip fails to shutdown the reactor, and subsequent manual actions taken at the main control boards are not successful in shutting down the reactor.** **Note:** A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies. **SA5.1** [1] a. An automatic or manual trip did not reduce reactor power to less than 5%. **AND** b. Manual actions taken at the main control boards are not successful in reducing reactor power to less than 5%.	**SU5 Automatic or manual trip fails to shutdown the reactor.** **Note:** A manual action is any operator action, or set of actions, which causes the control rods to be rapidly inserted into the core, and does not include manually driving in control rods or implementation of boron injection strategies. **SU5.1** [1] a. An initial automatic or manual trip did not reduce reactor power to less than 5%. **AND** b. **EITHER** of the following: • A subsequent manual action taken at the main control boards is successful in shutting down the reactor. o Reactor Trip (Switch) o AMSAC/DSS (Switch) o Turbine Trip (Pushbutton) • A subsequent automatic trip is successful in shutting down the reactor.	System Malfunctions Continued	ATWS	
	Communications / Hazard Events			**SA9 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.** **Note:** • If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted. • If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted. **SA9.1** [1 2 3 4] a. The occurrence of **ANY** of the following hazardous events: • Seismic event (earthquake) • Internal or external flooding event • High winds or tornado strike • FIRE • EXPLOSION • Low River Water Level • Other events with similar hazard characteristics as determined by the Shift Manager. **AND** b. 1. Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode. **AND** 2. **EITHER** of the following: • Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode. • The event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode.	**SU6 Loss of all onsite or offsite communications capabilities.** **SU6.1** [1 2 3 4] Loss of **ALL** of the following onsite communication methods: • Sound Powered Phones • Plant Paging System • Plant Telephone Network • Plant Radio System **SU6.2** [1 2 3 4] Loss of **ALL** of the following Offsite Response Organization (ORO) communications methods: • Plant Telephone Network • Plant Radio System (dedicated offsite channels) **SU6.3** [1 2 3 4] Loss of **ALL** of the following NRC communications methods: • Plant Telephone Network • ENS Network		Communications / Hazard Events	
	Containment				**SU7 Failure to isolate containment or loss of containment pressure control.** **SU7.1** [1 2 3 4] a. Failure of containment to isolate when required by an actuation signal. **AND** b. **ALL** required penetrations are not closed within 15 minutes of the actuation signal. **SU7.2** [1 2 3 4] a. Containment pressure greater than 23 psig. **AND** b. Less than one full train of containment spray and any two containment fan coils is operating per design for 15 minutes or longer.		Containment	

HOT

GE	SAE	ALERT

NOTE: To classify the event: Determine which combination of the three barriers meet the Loss or Potential Loss threshold and check each applicable box.

*Containment Radiation Monitors are sensitive to temperature changes which can cause thermally induced current errors. These monitors should not be used for classification in the first 5 minutes after a Steam Line Break or LOCA in Containment. Once 5 minutes has expired these monitors can be used for EAL determination.

FG1 - Loss of any two barriers and Loss or Potential Loss of the third barrier.	FS1 -Loss or Potential Loss of any two barriers.	FA1 - Any Loss or any Potential Loss of either the Fuel Clad or RCS barrier.
1 2 3 4	1 2 3 4	1 2 3 4

Fission Product Barrier Degradation

	Fuel Clad Barrier LOSS	Fuel Clad Barrier POTENTIAL LOSS	RCS Barrier LOSS	RCS Barrier POTENTIAL LOSS	Containment Barrier LOSS	Containment Barrier POTENTIAL LOSS
1. RCS or SG Tube Leakage		A. Core Cooling CSF – ORANGE entry conditions met.	A. An automatic or manual ECCS (SI) actuation is required by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube RUPTURE.	A. GREATER THAN or EQUAL to 60 gpm leak rate excluding normal reductions in RCS inventory (example, letdown system or RCP seal leak-off) caused by **EITHER** of the following: 1. UNISOLABLE RCS leakage **OR** 2. SG tube leakage. **OR** B. RCS Integrity CSF – RED entry conditions met.	A. A leaking or RUPTURED SG is FAULTED outside of containment.	
2. Inadequate Heat Removal	A. Core Cooling CSF – RED entry conditions met.	A. Core Cooling CSF – ORANGE entry conditions met. **OR** **Note:** Heat Sink CSF should not be considered RED if total AFW flow is less than 200 gpm due to operator action. B. Heat Sink CSF - RED entry conditions met.		**Note:** Heat Sink CSF should not be considered RED if total AFW flow is less than 200 gpm due to operator action. A. Heat Sink CSF - RED entry conditons met.		A. Core Cooling CSF – RED entry conditions met for 15 minutes or longer.
3. RCS Activity/Containment Radiation	* A. Containment radiation monitor reading greater than 5,500 R/hr on 1(2)R-48 or 49. **OR** B. Coolant activity greater than 300 µCi/gm dose equivalent Iodine-131.		* A. Containment radiation monitor reading greater than 40 R/hr on 1(2)R-48 or 49.			* A. Containment radiation monitor reading greater than 20,000 R/hr on 1(2)R-48 or 49.
4. Containment Integrity or Bypass					A. Containment isolation is required. **AND** **EITHER** of the following: 1. Containment integrity has been lost based on Emergency Director judgment. **OR** 2. UNISOLABLE pathway from the containment to the environment exists. **OR** B. Indications of RCS leakage outside of containment.	A. Containment CSF – RED entry conditions met. **OR** B. Containment H_2 concentration greater than or equal to 6%. **OR** C. 1. Containment pressure greater than 23 psig. **AND** 2. Less than one full train of containment spray and any two containment fan coils is operating per design for 15 minutes or longer.
5. Emergency Director Judgment	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Fuel Clad Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the RCS Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Loss of the Containment Barrier.	A. **ANY** condition in the opinion of the Emergency Director that indicates Potential Loss of the Containment Barrier.

Fission Product Barrier Degradation

HOT CONDITIONS

COLD

Cold Shutdown / Refuel System Malfunction

Inventory Control

GENERAL EMERGENCY

CG1 Loss of RPV inventory affecting fuel clad integrity with containment challenged.

Note: The Emergency Director should declare the General Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

CG1.1 (5, 6)
a. RPV level less than 56% RVLIS Full Range (Mode 5) for 30 minutes or longer.

AND

b. **ANY** indication from the Containment Challenge Table C1.

CG1.2 (5, 6)

a. RPV level cannot be monitored for 30 minutes or longer in Mode 5 or 6.
AND
b. Core uncovery is indicated by **ANY** of the following:
- 1(2)R-2, Containment Vessel Area Monitor, reading greater than 1 R/hr (10^0 R/hr).
- Erratic source range monitor indication.
- UNPLANNED increase in Containment Sumps A or C, or Waste Holdup Tank levels of sufficient magnitude to indicate core uncovery.

AND
c. **ANY** indication from the Containment Challenge Table C1.

Containment Challenge Table C1
• CONTAINMENT CLOSURE not established*
• H_2 concentration greater than or equal to 6% inside containment
• UNPLANNED increase in containment pressure
* If CONTAINMENT CLOSURE is re-established prior to exceeding the 30-minute time limit, then declaration of a General Emergency is not required.

SITE AREA EMERGENCY

CS1 Loss of RPV inventory affecting core decay heat removal capability.

CS1.1 (5)
a. CONTAINMENT CLOSURE not established.
AND
b. RPV level less than 65% RVLIS Full Range (Mode 5).

CS1.2 (5)
a. CONTAINMENT CLOSURE established.
AND
b. RPV level less than 56% RVLIS Full Range (Mode 5).

Note: The Emergency Director should declare the Site Area Emergency promptly upon determining that 30 minutes has been exceeded, or will likely be exceeded.

CS1.3 (5, 6)
a. RPV level cannot be monitored for 30 minutes or longer in Mode 5 or 6.

AND

b. Core uncovery is indicated by **ANY** of the following:
- 1(2)R-2, Containment Vessel Area Monitor, reading greater than 1 R/hr (10^0 R/hr).
- Erratic source range monitor indication.
- UNPLANNED increase in Containment Sumps A or C, or Waste Holdup Tank levels of sufficient magnitude to indicate core uncovery.

ALERT

CA1 Loss of RPV inventory.

CA1.1 (5, 6)
Loss of RPV inventory as indicated by level less than **ANY** of the following:
- 10" Refueling Canal
- 10" ERCS DP
- 69% RVLIS Full Range (Mode 5).

Note: The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CA1.2 (5, 6)
a. RPV level cannot be monitored for 15 minutes or longer.
AND
b. UNPLANNED increase in Containment Sumps A or C, or Waste Holdup Tank levels due to a loss of RPV inventory.

NUE

CU1 UNPLANNED loss of RPV inventory for 15 minutes or longer.

Note: The Emergency Director should declare the Unusual Eventt promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CU1.1 (5, 6)
UNPLANNED loss of reactor coolant results in RPV level less than a procedurally required lower limit for 15 minutes or longer.

CU1.2 (5, 6)
a. RPV level cannot be monitored.
AND
b. UNPLANNED increase in Containment Sumps A or C, or Waste Holdup Tank levels.

Loss of Power

ALERT

CA2 Loss of all Offsite and all Onsite AC power to emergency buses for 15 minutes or longer.

Note: The Emergency Director should declare the Alert promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CA2.1 (5, 6, DEF)
Loss of **ALL** Offsite and **ALL** Onsite AC Power to both Safeguards Buses 15 and 16 (25 and 26) for 15 minutes or longer.

NUE

CU2 Loss of all but one AC power source to emergency buses for 15 minutes or longer.

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CU2.1 (5, 6, DEF)
a. AC power capability to both Safeguards Buses 15 and 16 (25 and 26) is reduced to a single power source (Table S1) for 15 minutes or longer.

AND

b. Any additional single power source failure will result in loss of all AC power to SAFETY SYSTEMS.

Table S1	
Unit 1	**Unit 2**
1R Transformer	2RY Transformer
CT11 Transformer	CT12 Transformer
D1 Diesel Generator	D5 Diesel Generator
D2 Diesel Generator	D6 Diesel Generator
Aligned Available Cross-Ties	

CU4 Loss of Vital DC power for 15 minutes or longer.

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CU4.1 (5, 6)
Indicated voltage is less than 111 VDC on required 125 VDC Panels 11 and 12 (21 and 22) for 15 minutes or longer.

COLD

	GENERAL EMERGENCY	SITE AREA EMERGENCY	ALERT	NUE

Cold Shutdown / Refuel System Malfunction Continued

Temperature Control

General Emergency: (none)

Site Area Emergency: (none)

ALERT

CA3 Inability to maintain the plant in cold shutdown.

Note: The Emergency Director should declare the Alert promptly upon determining that the applicable time has been exceeded, or will likely be exceeded.

CA3.1 (Modes: 5, 6)
UNPLANNED increase in RCS temperature to greater than 200 °F for greater than the duration specified in Table C2.

Table C2: RCS Heat-up Duration Thresholds

RCS	CONTAINMENT CLOSURE	Heat-up Duration
Not intact or in RCS Reduced Inventory	Not Established	0 minutes
	Established	20 minutes*
Intact (capable of being pressurized)	N/A	60 minutes*

* If RHR (or at least 1 S/G when RCS intact) is in operation within this time frame and RCS temperature is being reduced, the EAL is not applicable.

CA3.2 (Modes: 5, 6)
UNPLANNED RCS pressure increase greater than 25 psig.
(This Threshold does not apply during water-solid plant conditions.)

NUE

CU3 UNPLANNED increase in RCS temperature.

CU3.1 (Modes: 5, 6)
UNPLANNED increase in RCS temperature to greater than 200 °F.

Note: The Emergency Director should declare the Unusual Event promptly upon determining that 15 minutes has been exceeded, or will likely be exceeded.

CU3.2 (Modes: 5, 6)
Loss of **ALL** RCS temperature and RPV level indication for 15 minutes or longer.

Communications

General Emergency: (none)

Site Area Emergency: (none)

Alert: (none)

NUE

CU5 Loss of all onsite or offsite communications capabilities.

CU5.1 (Modes: 5, 6, DEF)
Loss of **ALL** of the following onsite communication methods:
- Sound Powered Phones
- Plant Paging System
- Plant Telephone Network
- Plant Radio System

CU5.2 (Modes: 5, 6, DEF)
Loss of **ALL** of the following Offsite Response Organizations (ORO) communications methods:
- Plant Telephone Network
- Plant Radio System (dedicated offsite channels)

CU5.3 (Modes: 5, 6, DEF)
Loss of **ALL** of the following NRC communications methods:
- Plant Telephone Network
- ENS Network

Hazardous Events

General Emergency: (none)

Site Area Emergency: (none)

ALERT

CA6 Hazardous event affecting a SAFETY SYSTEM needed for the current operating mode.

Note:
- If the affected SAFETY SYSTEM train was already inoperable or out of service before the hazardous event occurred, then this emergency classification is not warranted.
- If the hazardous event only resulted in VISIBLE DAMAGE, with no indications of degraded performance to at least one train of a SAFETY SYSTEM, then this emergency classification is not warranted.

CA6.1 (Modes: 5, 6)
a. The occurrence of **ANY** of the following hazardous events:
- Seismic event (earthquake)
- Internal or external flooding event
- High winds or tornado strike
- FIRE
- EXPLOSION
- Low River Water Level
- Other events with similar hazard characteristics as determined by the Shift Manager.

AND

b. 1. Event damage has caused indications of degraded performance on one train of a SAFETY SYSTEM needed for the current operating mode.

AND

2. **EITHER** of the following:
- Event damage has caused indications of degraded performance to a second train of the SAFETY SYSTEM needed for the current operating mode.
- Event damage has resulted in VISIBLE DAMAGE to the second train of the SAFETY SYSTEM needed for the current operating mode.

NUE: (none)

COLD CONDITIONS

DEFINITIONS

CONFINEMENT BOUNDARY: The barrier(s) between spent fuel and the environment once the spent fuel is processed for dry storage.

CONTAINMENT CLOSURE: No open containment penetrations exist as identified in C19.9, Containment Boundary Control during Mode 5, Cold Shutdown and Mode 6, Refueling. The definition of an open containment penetration is a penetration that provides direct access from the containment atmosphere to the outside environment with no automatic closure available.

EXPLOSION: A rapid, violent and catastrophic failure of a piece of equipment due to combustion, chemical reaction or overpressurization. A release of steam (from high energy lines or components) or an electrical component failure (caused by short circuits, grounding, arcing, etc.) should not automatically be considered an explosion. Such events may require a post-event inspection to determine if the attributes of an explosion are present.

FAULTED: The term applied to a steam generator that has a steam leak on the secondary side of sufficient size to cause an uncontrolled drop in steam generator pressure or the steam generator to become completely depressurized.

FIRE: Combustion characterized by heat and light. Sources of smoke such as slipping drive belts or overheated electrical equipment do not constitute FIRES. Observation of flame is preferred but is NOT required if large quantities of smoke and heat are observed.

HOSTILE ACTION: An act toward a NPP or its personnel that includes the use of violent force to destroy equipment, take HOSTAGES, and/or intimidate the licensee to achieve an end. This includes attack by air, land, or water using guns, explosives, PROJECTILEs, vehicles, or other devices used to deliver destructive force. Other acts that satisfy the overall intent may be included. HOSTILE ACTION should not be construed to include acts of civil disobedience or felonious acts that are not part of a concerted attack on the NPP. Non-terrorism-based EALs should be used to address such activities (i.e., this may include violent acts between individuals in the owner controlled area).

IMMINENT: The trajectory of events or conditions is such that an EAL will be met within a relatively short period of time regardless of mitigation or corrective actions.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI): A complex that is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage.

ISFSI PROTECTED AREA: The area surrounding the Independent Spent Fuel Storage Installation encompassed by the double chain link fence surrounding the ISFSI as defined in the Security Plan; the ISFSI Protected Area is excluded from the Plant Protected Area.

OWNER CONTROLLED AREA: Land owned or leased by Prairie Island Nuclear Generating Plant. This area is bounded by a wire mesh, owner controlled fence. Unauthroized personnel are not allowed within this area.

PROTECTED AREA: The area encompassing all controlled areas within the security protected area fence as shown in USAR Figure 1.1-3, Site Plan Prairie Island Security Fence. This area does not include the ISFSI.

REFUELING PATHWAY: The reactor refueling cavity, spent fuel pool, or fuel transfer canal.

RUPTURED: The condition of a steam generator in which primary-to-secondary leakage is of sufficient magnitude to require a safety injection.

SAFETY SYSTEM: A system required for safe plant operation, cooling down the plant and/or placing it in the cold shutdown condition, including the ECCS. These are typically systems classified as safety-related.

SECURITY CONDITION: Any Security Event as listed in the approved security contingency plan that constitutes a threat/compromise to site security, threat/risk to site personnel, or a potential degradation to the level of safety of the plant. A SECURITY CONDITION does not involve a HOSTILE ACTION.

UNISOLABLE: An open or breached system line that cannot be isolated, remotely or locally.

UNPLANNED: A parameter change or an event that is not 1) the result of an intended evolution or 2) an expected plant response to a transient. The cause of the parameter change or event may be known or unknown.

VISIBLE DAMAGE: Damage to a SAFETY SYSTEM train that is readily observable without measurements, testing or analysis. The visual impact of the damage is sufficient to cause concern regarding the operability or reliability of the affected SAFETY SYSTEM train.

NOTE: Refer to F3-2.1, Emergency Action Level Technical Bases, as well as this Wall Chart to Classify an Event

F3	CORE DAMAGE DETERMINATION	NUMBER: F3-17.1
		REV: 6

REFERENCE USE
• Procedure should be at the work location. • Procedure segments may be performed from memory. • Use the procedure to verify segments have been completed. • When required, sign or initial appropriate blocks to certify that all segments are complete.

APPROVAL:

PCR #: 602000015508

1.0 PURPOSE

1.1 The purpose of this procedure is to provide a means to determine if reactor core damage has occurred.

2.0 APPLICABILITY

2.1 In the absence of Nuclear Engineering Staff, this procedure should be used during the initial phases of an emergency to determine the likelihood of core damage.

3.0 PRECAUTIONS

C **3.1** The values obtained by the use of this procedure are best estimates, but should be used for decisions until a more detailed evaluation can be completed. The Inadequate Core Cooling Monitor (ICCM) should be used during Emergency Operating Procedure (EOP) usage when checking Reactor Coolant System (RCS) subcooling. Emergency Response Computer System (ERCS) subcooling values should be used for information and comparison only. Any EOP decisions should be based on ICCM indications.

3.2 Iodine spiking may occur after a shutdown or significant power change, usually during the 2 to 6 hour period following the power change. Iodine spiking is a characteristic of the condition where an increase in the normal primary coolant activity is noted, but no damage to the cladding has occurred.

4.0 SPECIAL CONSIDERATION

NONE

5.0 RESPONSIBILITIES

5.1 TSC technical staff are responsible to determine if reactor core damage may or has occurred according to the guidance provided in this procedure.

6.0 DISCUSSION

6.1 The approach utilized in this procedure to determine potential or actual core damage is based on the review of key indicators that measure core exit temperatures, reactor vessel water level, containment radiation monitors, and containment hydrogen concentration. These indicators are readily available at the onset of an event.

6.2 As time and conditions permit, these estimates may be compared to the calculated core damage assessment results that are based on measured fission product distribution in the RCS and containment or post accident sampling.

7.0 PREREQUISITES

7.1 An emergency of an Alert, Site Alert, or General Emergency has been declared.

8.0 PROCEDURE

NOTE:	The plant indicators described in this procedure are available via an ERCS Group Tabular Display for each unit. See ERCS Main Menu – Tabular Display – Load Group – 1_F3-17-1 or 2_F3-17-1.

8.1 Review the following indicators to estimate core damage.

8.1.1 Containment Hydrogen Concentration:

A. Obtain the containment hydrogen concentration (%).

NOTE:	Within the accuracy of this methodology, it is assumed that recombiners will have an insignificant effect on the hydrogen concentration when it is indicated that extensive zirconium-steam reaction could have occurred.

B. From Figure 1, determine the percentage (%) zirconium water reaction.

C. Table 1 can be used to estimate the extent of core damage estimate.

8.1.2 Core Exit Thermocouple Readings:

NOTE:	For core exit thermocouple temperatures, the maximum temperatures achieved during the event should be used in determining core damage.

A. Obtain as many core exit thermocouple readings as possible for evaluation of core temperature conditions.

NOTE:	If a thermocouple reads greater than 1650°F or is reading considerably different than neighboring thermocouples, thermocouple failure should be considered.

B. Compare the thermocouple readings with those in Table 1 to estimate core damage.

NOTE:	Radiation Monitors in containment may experience errors during first 4 hours after a DBA LOCA due to thermally induced errors. See Attachment 1 for more information.

8.1.3 Containment Radiation Monitor:

A. Obtain the containment dome monitor readings, R/Hr, from R-48 and/or R-49.

B. From Figure 2 determine the core damage estimate. The exposure rate in Figure 2 is based on the release of only noble gases to the containment. Halogens and other fission products were not considered to be significant contributors to the containment monitor reading.

8.2 All indicators should confirm any core damage estimates. If some indicators do not agree on core damage estimates, then recheck of indications may be performed, or certain indicators may be discounted, based on engineering judgment.

8.3 As time and conditions permit, the TSC nuclear engineer will be conducting a core damage estimate based on post accident sampling.

9.0 REFERENCES

9.1 CAP 1301548, EP Drill-Source of RCS Subcooling Values, (C)

10.0 ATTACHMENTS

10.1 Table 1 - Characteristics of Categories of Fuel Damage

10.2 Table 2 – Expected Fuel Damage Correlation With Fuel Rod Temperature For Information Only

10.3 Figure 1 – Containment Hydrogen Concentration Based on Zirconium Water Reaction

10.4 Figure 2 – Percent Noble Gases in Containment

10.5 Attachment 1 - Technical Evaluation High Range Containment Radiation Monitors Thermal Induced Current

Table 1 Characteristics of Categories of Fuel Damage

Core Damage Category *	Percent (%) and Type of Fission Products Released	Fission Product Ratio ***	Containment Rad Monitor R/hr 10 hrs after shutdown**	Core Exit Thermocouples Reading (Deg F)	Core Uncovery Indication	Hydrogen Monitor (Vol % H_2)
No clad damage	KR-87 < 1 X 10^{-3} Xe-133 < 1 x 10^{-3} I-131 < 1 X 10^{-3} I-133 < 1 X 10^{-3}	Not Applicable	--	< 750	No uncovery	Negligible
0-50% clad damage	Kr-87 10^{-3} – 0.01 Xe-133 10^{-3} – 0.1 I-131 10^{-3} – 0.3 I-133 10^{-3} – 0. 1	Kr-87 = 0.022 I-133 = 0.71	0 – 50	750 – 1300	Core uncovery	0 - 6
50 – 100% clad damage	Kr-87 0.01 – 0.02 Xe-133 0.1 – 0.2 I-131 0.3 – 0.5 I-133 0.1 – 0.2	Kr-87 = 0.022 I-133 = 0.71	50 to 100	1300 – 1650	Core uncovery	6 - 13
0 – 50% fuel pellet overtemperature	Xe-Kr, Cs, I 1 – 20 Sr-Ba 0 – 0.1	Kr-87 = 0.22 I-133 = 2.1	100 to 1.15E4	> 1650	Core uncovery	6 – 13
50-100% fuel pellet overtemperature	Xe-Kr, Cs, I 20 - 40 Sr-Ba 0.1– 0.2	Kr-87 = 0.22 I-133 = 2.1	1.15E4 to 2.3E4	> 1650	Core uncovery	6 - 13
0 – 50% fuel melt	Xe, Kr, Cs, I 40-70 Sr-Ba 0.2 – 0.8 Pr 0.1 – 0.8	Kr-87 = 0.22 I-133 = 2.1	2.3E4 to 2.7E4	> 1650	Core uncovery	6 - 13
50 – 100% fuel melt	Xe, Kr, Cs, I, Te > 70 Sr, Ba > 24 Pr > 0.8	Kr-87 = 0.22 I-133 = 2.1	> 2.7E4	> 1650	Core uncovery	6 - 13

Characteristics of Categories of Fuel Damage

* This table is intended to indicate whether there is fuel damage.

** These values are from Figure 2 and should be revised for times other than 10 hours.

*** $\frac{\text{Kr-87}}{\text{Xe-133}}, \frac{\text{I-133}}{\text{I-131}}$

Table 2 Expected Fuel Damage Correlation With Fuel Rod Temperature
For Information Only – See Note Below

Fuel Damage	Temperature °F*
No Damage	< 1300
Clad Damage	1300 - 2000
Ballooning of zircaloy cladding	> 1300
Burst of zircaloy cladding	1300 - 2000
Oxidation of cladding and hydrogen generation	> 1600
Fuel Overtemperature	2000 - 3450
Fission product fuel lattice mobility	2000 - 2550
Grain boundary diffusion release of fission products	2450 - 3450
Fuel Melt	> 3450
Dissolution and liquefaction of UO_2 in the Zircaloy - ZrO_2 eutectic	> 3450
Melting of remaining UO_2	5100

* These temperatures are material property characteristics and are non-specific with respect to locations within the fuel and/or fuel cladding.

Figure 1 Containment Hydrogen Concentration Based on Zirconium Water Reaction

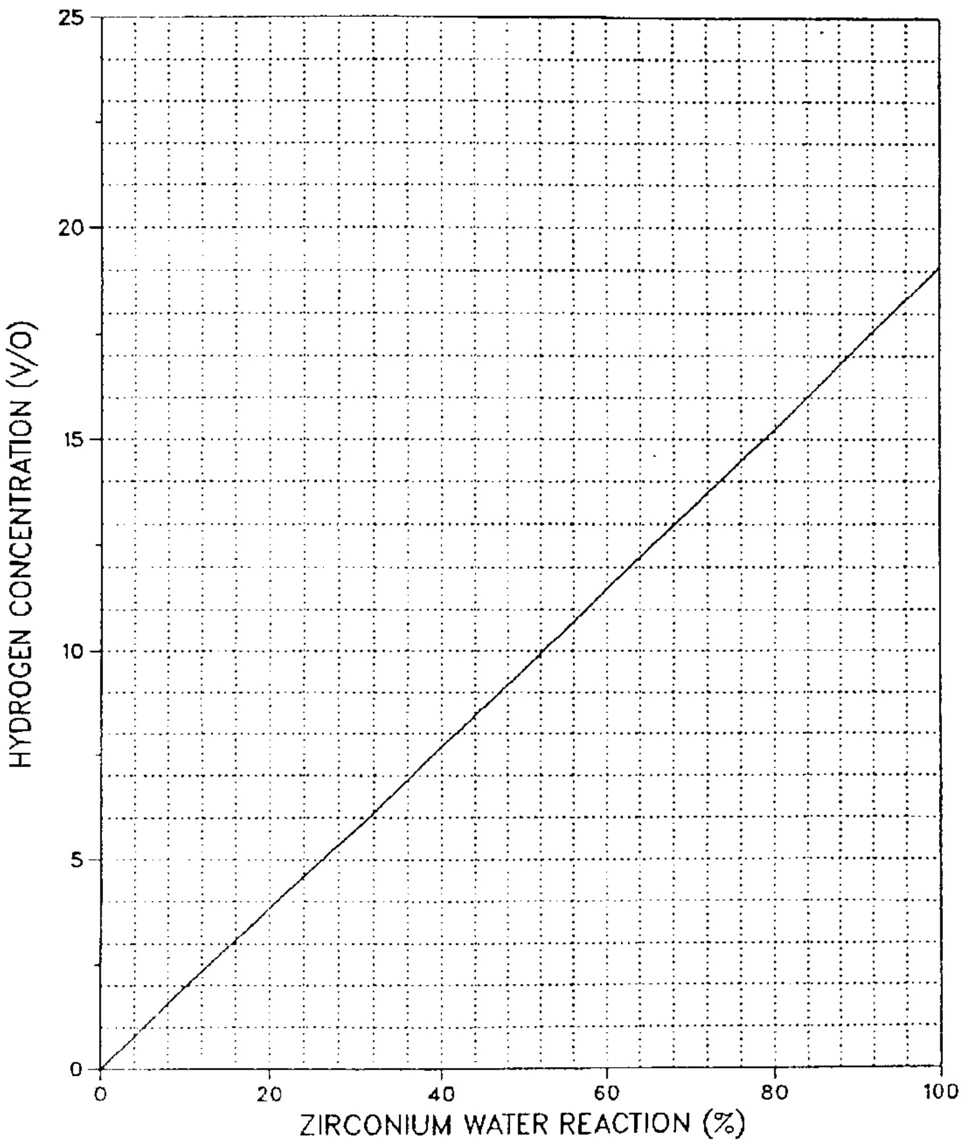
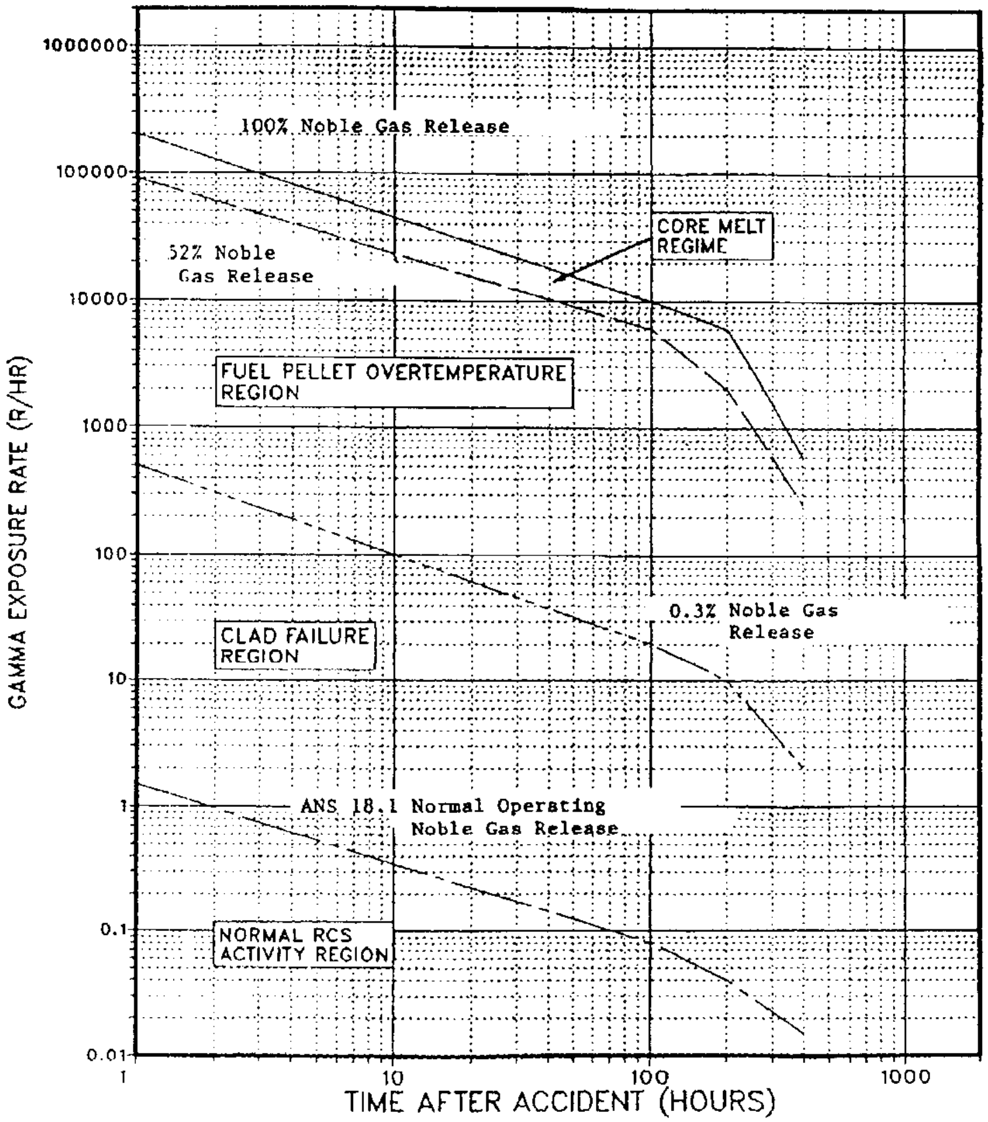


Figure 2 Percent Noble Gases in Containment



Attachment 1 Technical Evaluation High Range Containment Radiation Monitors Thermal Induced Current ECR 608000000176

NOTE:	Sections IV and X discussed in this attachment refer to the sections within the ECR. References listed in brackets can be found in the ECR.

1.0 Background

From recent NRC Environmental Qualification Component Design Bases Inspections (EQ CDBI) several stations were issued green non-cited violations (NCVs) for not adequately documenting operability of their CHRRMs or to restoring the capability to classify emergency action levels [12, 13, 14]. NRC Information Notice 97-45 Supplement 1 had identified a potential transient operational deficiency on the coaxial signal cables associated with CHRRMs from thermally induced currents (TIC). The potential exists for thermally induced currents on the signal cables associated with these monitors to cause erratic and inaccurate readings.

The TIC phenomena appears to be dependent on the temperature change magnitude, rate of change, the type of cabling used for the plant, the service temperature history, and the uniformity of the cable manufacturing run. TR-112582 [19] notes the sources of TIC include trapped space charge in the insulation of coaxial cables and polar properties of a polysulfone layer used in some cables. The thermal resistance of the cable jacket and the high thermal conductivity of the cable braid create a thermal stimulus at the cable dielectric. The thermal wave resulting from an external temperature change expands the dielectric. If free charges are available in the dielectric, a charge is induced on the braid and center conductor of the cable. The charge is dissipated as current into the impedance of the detector electronics. TIC therefore causes small currents to be induced due to temperature differential between the inner and outer conductors of the coax cable. The currents are too small to impact most instrumentation and control cables, are transient in nature, and are only present during significant temperature transients. The current/dose relationship for Sorrento/GA High Range Radiation Monitoring Systems is shown in TR-112582 [19] as:

Cable Current (nA)	Rad Monitor Indicated Dose Rate (R/hr)
10^{-2}	1
10^{-1}	10
1	10^2
10	10^3
10^2	10^4
10^3	10^5
10^4	10^6

Attachment 1 Technical Evaluation High Range Containment Radiation Monitors Thermal Induced Current ECR 608000000176

Testing performed for Southern California Edison confirmed the magnitude and direction of the spurious signal was a function of the temperature gradient across the cable insulation. The NRC Information Notice indicated that the duration of the spurious signal could be as long as 15 minutes, with the worst impact over in approximately 1 minute.

PINGP evaluated the potential impact of TIC on PINGP instrumentation and the only devices impacted at PINGP are the containment high range radiation monitors (CHRRM). From SAP FLOC information, cable lengths inside containment associated with these detectors are 1R-48 (1CMR-1): 80ft, 1R-49 (1CMW-1): 80ft, 2R-48 (2CMR-1): 90ft, and 2R-49 (2CMW-1): 65ft. The PINGP cable is type CBLTP 362 (Rockbestos type RSS-6-104) 2-1/C-22 COAX with a solid #22 AWG conductor. The TIC phenomenon could cause these monitors to go into high alarm during the rapid temperature increases associated with LOCAs and MSLBs inside containment. They could also alarm as failed low due to negative TIC effects during rapid cooling for the same events. CHRRMs are used to estimate post-accident fuel damage which factors into post-accident emergency classifications and in certain EOPs. PINGP's response to this issue was addressed in CHAMPS Condition Reports No. 19980371 [5] and No. 19980809 [6].

A recent review of condition reports no. 19980371 and no. 19980809 determined that no attempt was made to quantify the TIC effects from a PINGP LOCA or MSLB. Also, discussions about whether the CHRRMs meet RG 1.97 requirements with the TIC phenomenon were based on having alternate indications for fuel damage and not expecting to have fuel damage in the first 10-15 minutes of a LOCA. They did not describe how CHRRMs meet RG 1.97 requirements themselves. With recent NRC violations for other sites not adequately addressing the TIC phenomenon in their CHRRMs, a new Passport Action Request 01555551 [7], was written to provide the mechanism to assure a new review of the OE is performed.

Based on the lack of quantitative data of the TIC effect at PINGP for 1R-48/1R-49 & 2R-48/2R-49, Engineering Evaluation ECR 608000000013 was undertaken to determine the site-specific TIC effect of these CHRRMs during a worst-case LOCA and MSLB. Based on the results of this evaluation, CAP 501000001861 was written on 8/22/2017 and 1R-48/1R-49 & 2R-48/2R-49 were declared INOPERABLE by Operations.

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Since this time, the NRC has approved PINGP's LAR L-PI-17-006 [20] in NRC SER on March 6, 2018 [21] revising fission product barrier containment radiation EALs for R-48 & R-49 based on approved calculation GEN-PI-092 [9]. Specifically, the EAL containment radiation levels for a RCS barrier loss increase from 7 to 40 R/hr, a fuel cladding barrier loss increase from 200 to 5,500 R/hr, and a potential containment barrier loss increase from 800 to 23,000 R/hr. This evaluation will consider whether the changes in these EALs along with proposed changes to guidance in associated emergency and operations procedures are adequate to demonstrate that 1R-48/1R-49 & 2R-48/2R-49 can perform their intended functions when required.

2.0 Analysis – Design Basis Accidents and Quantifying Tic Effects

From PINGP accident analyses, the design basis accidents that cause the largest temperature transients in containment that would translate to the largest TIC effects are the Large-Break Loss-Of-Coolant Accident (LOCA) and the Main Steam Line Break (MSLB). From PINGP's LOCA analyses of record [8], the LOCA scenario with the highest peak temperature ramps from 120°F to around 266°F within approximately 15 seconds (See Figure A-1 below). The black line represents a temperature that bounds all analyzed scenarios with margin for EQ Program purposes and is not an actual DBA temperature profile.

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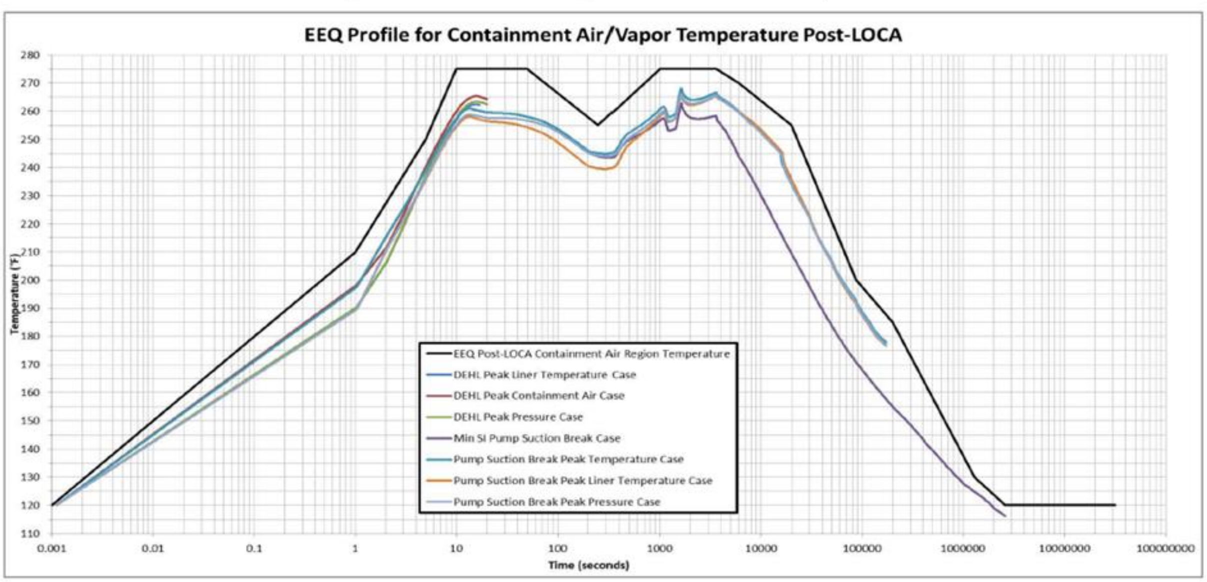

Figure A-1: Containment Air/Vapor Temperature Post-LOCA with Bounding EEQ Profile

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Only four of the seven analyzed LOCA profiles extend to the highest LOCA temperature peak that occurs at around 27 minutes into the event. Since all the LOCA temperature profiles have a very similar shape, it's not expected that TIC effects would be greatly different between the individual LOCA profiles. However, to account for the uncertainty of not obtaining the highest TIC effect from the shorter profiles, all the longer temperature profiles were run on the TIC software and the maximum difference between TIC effect results were added to the maximum TIC result from each of the longer profiles to obtain the maximum TIC effect result. Also, as previously mentioned, cable lengths have a direct effect on the TIC values and cable lengths differ between each radiation monitor. Therefore, TIC values for each radiation monitor were calculated from the TIC software results and cable lengths to show the variation in readings operators or observers would see during a LOCA.

For a MSLB, containment temperature ramps from 120°F to 311°F in 80 seconds followed by a relatively fast temperature reduction to 245 °F after 800 seconds (13 min.) [10] (See Figure B-1 below).

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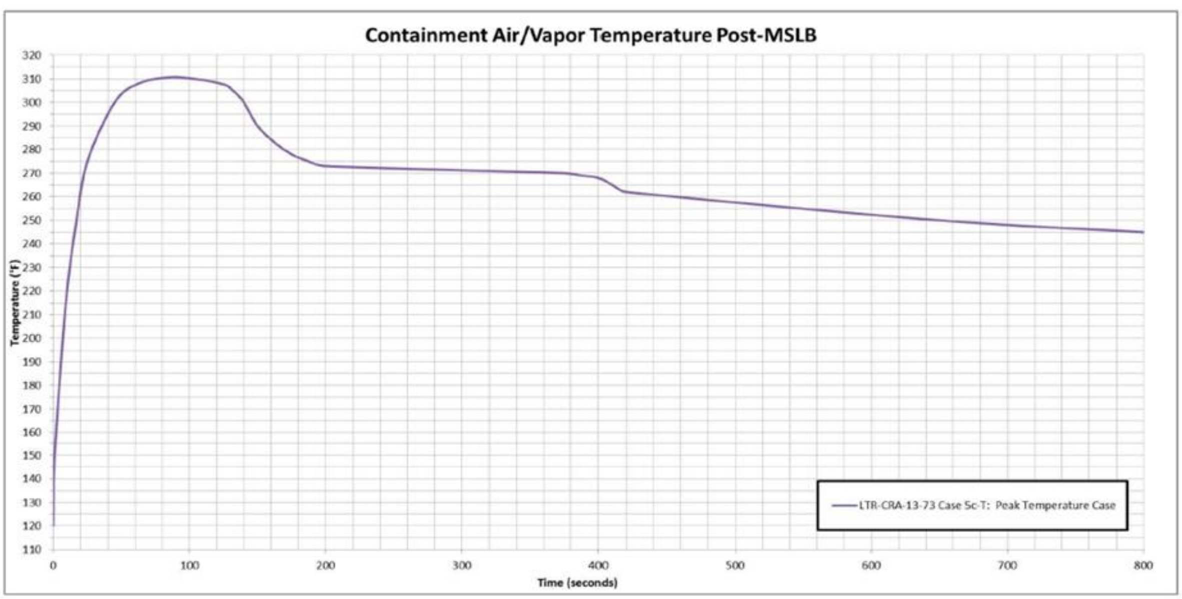

Figure B-1: Containment MSLB Peak Temperature Case

Although the MSLB temperature profile extends to only 800 seconds post-accident, from steam line break descriptions in USAR Section – 14 [2], no additional temperature spikes are expected from the event. So, the trend should continue as a slow decrease in temperature.

3.0 RESULTS - CHRRM POST-ACCIDENT RESPONSE TO A LOCA

TIC software results for the LOCA accident profiles [35] show that during the initial temperature spike that peaks at approximately 15 seconds, the CHRRMs will produce peak TIC errors greater than +2,550 R/hr at around 9 seconds post-LOCA in the radiation monitor most effected by the TIC phenomenon (2R-48) because it has the longest cable length. As containment cools, the TIC errors decrease quickly until they become negative at around 90 seconds. The TIC errors become more negative until about 120 seconds (2 minutes) when they bottom in the -27 R/hr. in 2R-48. TIC errors then trend upward again until they reach zero at around 300 seconds (5 minutes) post-LOCA. Then, as temperatures increase again, a secondary positive TIC error peak of +22 R/hr in 2R-48 occurs around 7 minutes post-LOCA. Then, the TIC errors slowly decrease until around 18 minutes when they quickly drop and turn negative until they bottom out around -11 R/hr. in 2R-48 at approximately 20 minutes post-LOCA. Fluctuation in TIC errors continue in decreasing magnitudes as containment air temperatures change throughout the first 65 minutes of the event. At the end of the 6,000 seconds (100 minutes) post-LOCA simulation, TIC values are slightly negative ranging from -0.49 R/hr on 2R-48 to -0.35 R/hr on 2R-49. The full LOCA TIC response chart for all CHRRMs is shown in Figure D below. Note: The TIC error values used to construct Figure D were calculated from the TIC software output of the LOCA profile that produced the highest peak and lowest trough response in the 5-20 minute timeframe multiplied by 1.26 to account for the 26% maximum variation in the peak magnitudes seen in all four LOCA responses. The reasoning behind using the output with highest TIC error magnitudes from the 5-20 minute timeframe is that it would be the worst case for making an EAL classification decision during an event.

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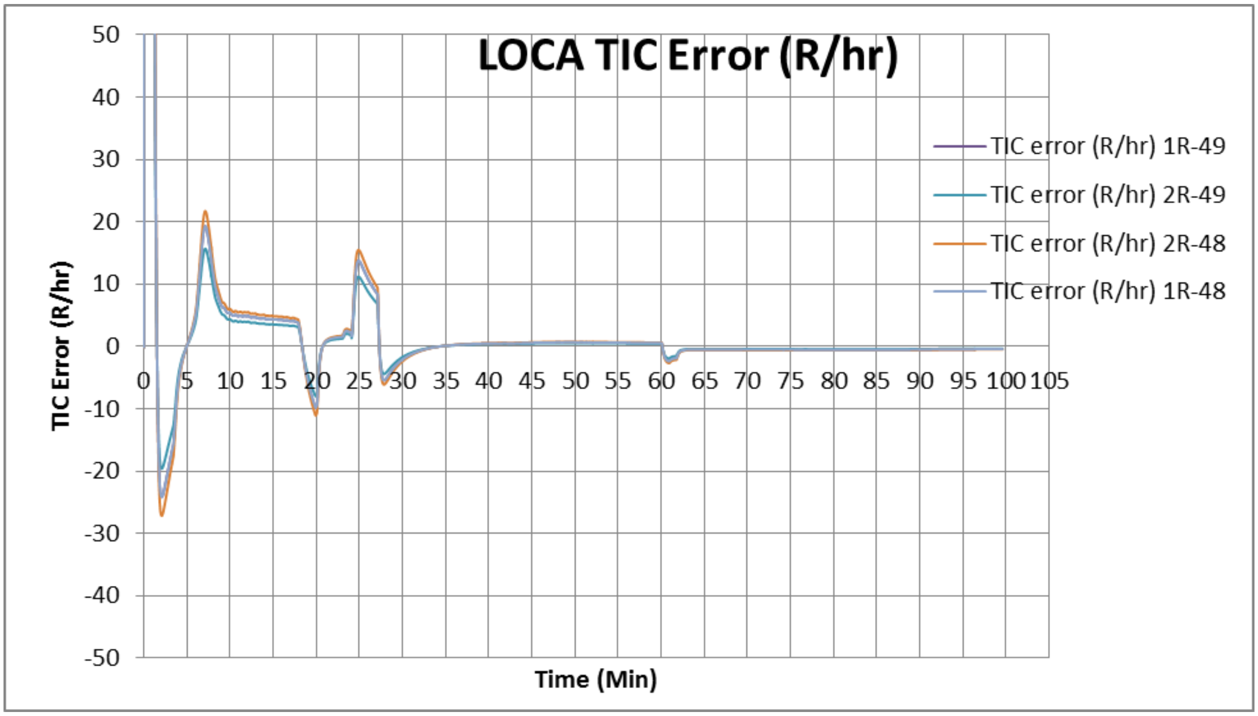

Figure D: Post-LOCA TIC Error for 1R-48, 1R-49, 2R-48, and 2R-49.

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4.0 Results - CHRRM Post-Accident Response to a MSLB

TIC software results for the MSLB accident profile [34] show that during the initial temperature spike that peaks at approximately 85 seconds, the CHRRMs will produce a peak TIC error of +1,922 R/hr at around 27 seconds post-MSLB in the radiation monitor most effected by the TIC phenomenon (2R-48). As containment cools, the TIC error decreases quickly until it becomes negative at around 117 seconds (1.95 minutes). The TIC error becomes more negative until around 158 seconds (2.6 minutes) when it bottoms out at -144 R/hr for 2R-48. TIC errors then trend less negative as temperature in containment slowly cool. However, at the end of the 800 seconds (13.3 minutes) post-MSLB simulation, TIC values are still negative and range from -7 R/hr on 2R-48 to -5 R/hr on 2R-49. See Figure E below.

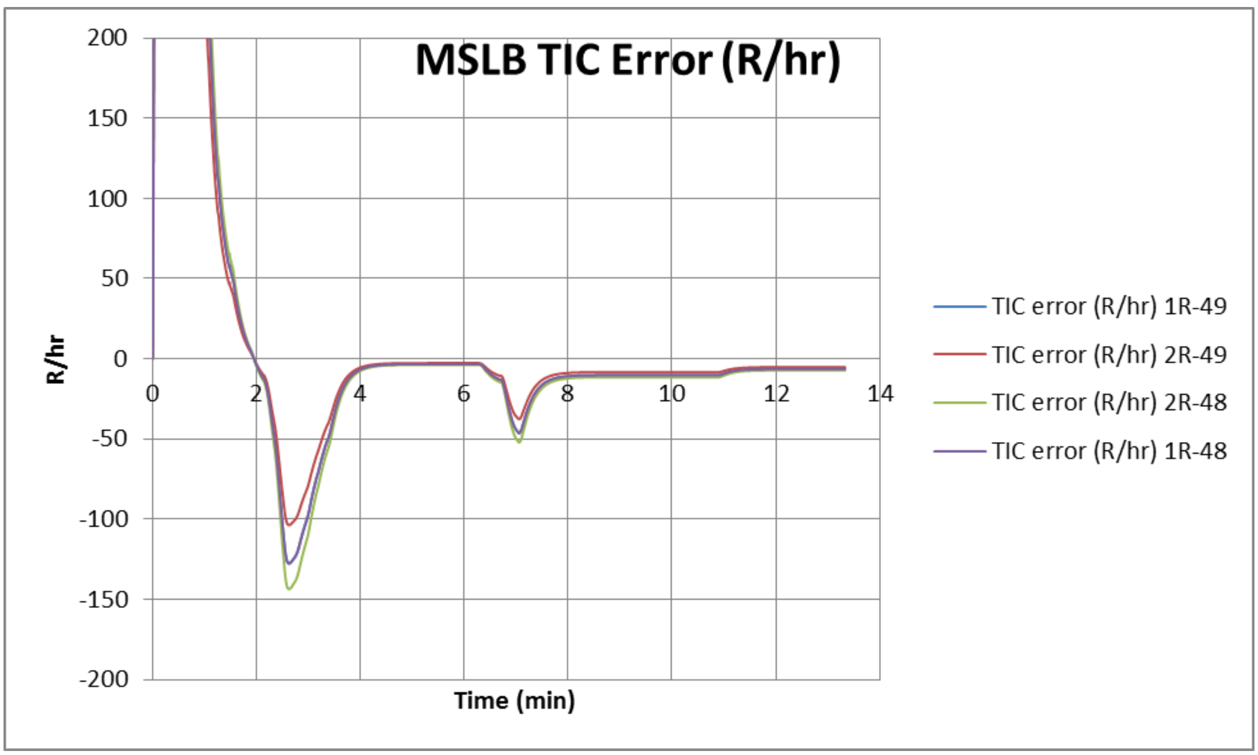

Figure E: Post-MSLB TIC Error for 1R-48, 1R-49, 2R-48, and 2R-49.

5.0 Evaluation - TIC Impact on Design and License Bases Functions

For Emergency Planning, the Containment High-Range Radiation Monitors are used to determine the condition of fission product barriers during an event. Specifically, they are used to identify a RCS Barrier Loss, a Fuel Clad Barrier Loss, and a Containment Barrier Potential Loss from the Emergency Action Level (EAL) Matrix in the Emergency Plan [24]. The recently approved License Amendment Request [20, 21] raises the EAL for an RCS Barrier Loss from 7 R/hr to 40 R/hr on both R-48 and R-49; a Fuel Clad Barrier Loss from 200 R/hr to 5,500 R/hr, and a potential Containment Barrier Loss from 800 R/hr to 20,000 R/hr. So, this evaluation will use and will only apply when these new EALs that are implemented.

In evaluating the ability of the CHRRMs to perform their functions of detection of significant releases and emergency plan actuation, the time constraint of significance is the requirement "to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an EAL has been exceeded.[24]" However, in practice, the individual making the event classification needs the information no later than 13 minutes post-accident to document and verify the correct classification in the EAL tables to meet this deadline. The likelihood of an emergency classification being declared in the first 5 minutes of an event is small due to the high demands on the Main Control Room staff at the beginning of an event. So, for this evaluation, the critical timeframe when information from the CHRRMs will be needed to perform their design and license bases functions is considered to be between 5 and 13 minutes post-accident.

For the CHRRMs to function properly during a LOCA or MSLB, their readings need to provide Operations information to correctly classify the event. Radiation readings must be accurate enough that they don't lead to under- or over-classification of the actual event. With this criteria in mind, CHHRM acceptance criteria for emergency plan actuation and release assessment is determined to be the following: Expected radiation level for fission product barrier loss + TIC error falls in the correct EAL range given for the fission product barrier loss during the 5-13 minute post-event timeframe.

Attachment 1 Technical Evaluation High Range Containment Radiation Monitors Thermal Induced Current ECR 608000000176

By definition, during a LOCA there is a loss of the RCS fission product barrier. Loss of the RCS fission product barrier alone would result in an alert event classification according to the EAL tables. From the current PINGP calculation of record [9], a loss of the RCS barrier without fuel failure would produce radiation levels in containment measured at 43 R/hr on the CHRRMs (not including the TIC error). The next-highest level of classification based on CHRRM readings is site area emergency due to fuel cladding failure. From the calculation of record [9], a 5% fuel cladding damage event (corresponding to a loss of fuel cladding barrier) would produce radiation levels in containment measured at 5,957 R/hr on the CHRRMs (not including the TIC error). Finally, the calculation of record [9] determined that a 20% fuel cladding damage event (corresponding to a potential loss of containment) would produce radiation levels of 23,830 R/hr. These radiation values will be used as the expected radiation levels in containment during loss of fission product barriers when evaluating the ability of CHRRMs to perform their intended functions of emergency plan actuation and release assessment during the 5-13 minute post-event timeframe.

During a MSLB in containment, the breach should be in only the non-radioactive secondary coolant system. Containment radiation should show no significant increase. Thus, based solely on CHRRM readings, acceptance criteria for the CHRRMs during an MSLB is indicated radiation levels should be below the RCS barrier loss EAL in the 5-13 minute post-accident timeframe resulting in no event being declared (The classification would be based on other plant indications). A review of MSLB data from the TIC software output [34] indicates that the range of expected TIC errors during the 5-13 minutes post-MSLB timeframe are -4 R/hr at 5 minutes to -52 R/hr at 7 minutes post-accident. A summary of expected radiation readings from the CHRRMS during MSLBs and LOCAs with the corresponding acceptance criteria is presented in Table 1 below.

One additional item of note is TR-112582 mentions the Sorrento/GA detector has a 10^{-11} amp normal output or “keep alive” signal current. If this signal is lost, the system issues a “fail” alarm to the operator. Therefore, if a negative TIC error signal of 10^{-11} amp or more occurs, the “keep alive” signal will be negated and a “fail” alarm will be generated. The following table compiles all the previously described TIC software results, information, acceptance criteria, and required time frames to summarize the ability of Operations to correctly classify the LOCA and MSLB events based on CHRRMs readings indications:

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Event	Expected Radiation Level (R/hr)	TIC Error 5-13 min. (R/hr)	CHRRMs Readings 5 -13 min. (R/hr) (Expected + TIC)	Acceptance Criteria (R/hr)	Acceptance Criteria Met?
LOCA	43	0 to 22	43 to 65	40 to 5,500	Yes.
LOCA w/Cladding Failure	5,957	0 to 22	5,957 to 5,979	5,500 to 20,000	Yes
LOCA w/potential loss of containment	23,830	0 to 22	23,830 to 23,852	> 20,000	Yes
MSLB	Normal area value <1	-4 to -52	“Failed low”	< 40	Yes, with additional guidance to address “Failed low”

Table 1: Comparison of CHRRM Indication vs. Acceptance Criteria

Table 1 shows that TIC errors during the 5 to 13 minutes period post-LOCA (without fuel cladding damage) are expected to produce readings on all four CHRRMs in the range of 43 to 65 R/hr. This range of radiation readings falls within the new EAL range of >40 and <5,500 R/hr corresponding to a RCS barrier loss. So, this meets the established acceptance criteria. For the case of a LOCA with fuel cladding failure, Table 1 shows CHRRMs would produce readings between 5,957 and 5,979 R/hr. These values also fall in the corresponding EAL acceptance criteria range for fuel cladding failure of 5,501 to 20,000 R/hr. For the case of potential loss of containment barrier, Table 1 shows CHRRMs would produce readings between 23,830 and 23,852 R/hr. These readings are above the EAL level of 20,000 R/hr that would indicate a potential loss of containment barrier. So, CHRRMs readings during all potential loss of fission product barriers due to LOCAs meet the acceptance criteria for emergency plan actuation and release assessment during the 5-13 minute post-accident timeframe.

For the case of an MSLB, Table 1 shows that TIC errors during the 5 to 13 minutes are expected to be of sufficient magnitude to negate the detector signal from normal containment radiation and the “keep alive” signal, producing a “fail low” alarm according to the detector logic diagram [29]. If appropriate guidance were provided in procedures and documents identified in Section IV that “failed low” alarms were a normal response to a cooldown event in containment for these detectors and were not failed detectors, operators and/or the emergency director could correctly conclude the fission product barriers were intact and would not classify the event based on the CHRRM readings. Therefore, CHRRMs acceptance criteria are also met for emergency plan actuation and release assessment during the 5-13 minute post-event timeframe post-MSLB.

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Assessing the ability of CHRRMs to perform its remaining intended function, long-term surveillance, it can be seen in both LOCA and MSLB TIC Error charts the TIC errors decrease as stored energy from the accident has fully released and temperatures stabilize in containment. TIC errors will trend to zero with no additional releases of energy and CHRRMs will provide accurate measurement of actual radiation conditions in containment as time passes. Therefore, it is concluded that CHRRMs are able to perform their design and license bases function of long-term surveillance of radiation levels in containment during accident conditions.

6.0 Conclusions and Recommendations

Based on this evaluation with the new containment radiation EALs in place, enhanced procedural guidance on the TIC effect specified in Section X., and associated training of Operations, RP, and Emergency Response performed, the CHRRMs are able to perform their license and design bases functions. This evaluation shows that five minutes after either a LOCA or a MSLB Operations will be able to use CHRRMs measurements to detect significant radiation releases and actuate the emergency plan (Classify events) in the timeframe necessary (5-13 minutes). If personnel question alarms or data indicating high readings in the first five minutes after an event or failed low readings, there will be appropriate guidance in the alarm response or emergency procedure to assure them that this is expected after a thermal transient. Also, CHRRMs will be accurate enough after five minutes for personnel to monitor containment radiation levels in the long-term. The accuracy of CHRRMs five minutes after LOCAs and MSLBs meets the acceptance criteria for all its intended functions with enhanced procedural guidance and training in place.

One recommendation is to ensure all the conditions of this evaluation are met by creating a tracking action to verify all required procedure changes and training from Section X have been completed before returning 1R-48/1R-49 & 2R-48/2R-49 to operable.

F3	CORE DAMAGE ASSESSMENT	NUMBER: F3-17
		REV: 16

REFERENCE USE
• Procedure should be at the work location.
• Procedure segments may be performed from memory.
• Use the procedure to verify segments have been completed.
• When required, sign or initial appropriate blocks to certify that all segments are complete.

APPROVAL:

PCR #: 602000015507

1.0 PURPOSE

1.1 The purpose of this procedure is to provide a means to best estimate the degree of reactor core damage from the measured fission product concentrations in water and gas samples taken for the primary system and containment under accident conditions.

2.0 APPLICABILITY

2.1 This procedure **SHALL** be used once sample activity data is available through step 8.1; otherwise, use F3-17.1 for core damage indication.

3.0 PRECAUTIONS

3.1 The numbers obtained using this procedure, are at best, estimates only.

3.2 When making core damage calculations as per this procedure, considerations should be given to other plant indicators, for example:

3.2.1 Incore Thermocouples.

3.2.2 Containment Radiation Monitors (R48/49).

3.2.3 Hydrogen Concentration in the Containment Atmosphere

3.3 Spiking may occur after a shutdown or significant power change, usually during the 2 to 6 hour period following the power change. Iodine spiking is a characteristic of the condition where an increase in the normal primary coolant activity is noted, but no damage to the cladding has occurred.

3.4 Keep all generated documents in accordance with F3-24, Record Keeping During an Emergency.

4.0 SPECIAL CONSIDERATIONS

4.1 The fuel pellet inventories listed in Table 2 are based on a 1650 MWth 2-loop Westinghouse plant. The Core Damage Assessment software, CDA, and Section 8.12 of this procedure for a manual calculation, scale these values to the current maximum licensed power level of 1677 MWth.

5.0 RESPONSIBILITIES

5.1 Persons qualified under PI-BEP-ERO-029 are responsible to estimate the degree of reactor core damage according to the guidance provided in this procedure. Persons not qualified **SHALL** assess damage per F3-17.1.

6.0 DISCUSSION

6.1 The approach utilized in this methodology of core damage assessment is measurement of fission product concentrations in the primary coolant system, and containment, when applicable, utilizing the post accident sampling system.

6.2 Certain nuclides have been selected to be associated with each particular core damage state, i.e., clad damage, fuel overheat and fuel melt. These nuclides reach equilibrium quickly within the fuel cycle. Once equilibrium conditions are reached, a fixed inventory of the nuclides is assumed to exist within the fuel pellet. For these nuclides which reach equilibrium, their relative ratios within the fuel pellet can also be considered to be constant. During operation, certain volatile fission products collect in the gap. The relative ratios in the gap can also be considered to be constant, however, the distribution of the nuclides in the gap is not in the same proportion as the fuel pellet inventory since the migration of each nuclide into the gap is dependent on its particular diffusion rate. The relative ratios of the nuclides analyzed during an accident may be compared to the predicted relative ratios existing in the gap and fuel pellet to determine the source of the fission product release, i.e., gap release or fuel pellet.

6.3 Clad damage is characterized by the release of these fission products, i.e., isotopes of the noble gases, iodine, and cesium which have accumulated in the gap and during the operation of the plant. When the cladding ruptures, it is assumed that the fission product gap inventory of the damaged fuel rods is instantaneously released to the primary system. For this methodology it is assumed that the noble gases will escape through the break of the primary system boundary to the containment atmosphere and the iodines will stay in solution and travel with primary system water during the accident.

6.4 Fission product release associated with overtemperature fuel conditions arises initially from the portion of the noble gas, cesium and iodine inventories that was previously accumulated in grain boundaries. In addition, small amounts of the more refractory elements, barium-lanthanum, and strontium are also released.

6.5 Fuel pellet melting leads to rapid release of many noble gases, halides, and cesiums remaining in the fuel after overheat conditions. Significant release of the strontium, barium-lanthanum chemical groups is perhaps the most distinguishing feature of melt release conditions.

6.6 Auxiliary indicators such as core exit thermocouples, reactor vessel water level, containment radiation monitors, and the containment hydrogen concentration are available for estimating core damage. These indications should confirm the core damage estimates which in turn are based on the radionuclide analysis.

7.0 PREREQUISITES

7.1 An emergency of an Alert, Site Alert, or General Emergency has been declared.

8.0 PROCEDURE

NOTE:	**The CDA (Core Damage Assessment) program may be used whenever core damage estimates are desired.**

8.1 Request the Radiation Protection Group to obtain the applicable samples to enable an adequate assessment of core damage. See Table 1 for suggested sampling locations.

NOTE:	**Monitor via LAN ERCS if available; otherwise, obtain printouts of ERCS screens every hour from ERCS terminal in TSC or control room. Monitoring frequency may be increased/decreased per evaluator's discretion. Values for CDA should be taken at the time of the sample.**

8.2 Obtain the following plant data at the approximate sample time:

8.2.1 Incore Thermocouple Map

- ERCS code "TC" and "TC1"
- IF ERCS is unavailable, THEN **request** the information from the control room ICCM panel via the three-way communicator.
- IF ICCM is unavailable, THEN contact I&C to request manual readings.

8.2.2 Containment Pressure

- ERCS point 1U5015A[2U5015A] available on SAS containment panel (code:XT14)
- IF ERCS is unavailable, THEN **request** the information from the control room via the three-way communicator.

8.2.3 Containment Temperature

- ERCS point 1U5013A[2U5013A]
- IF ERCS is unavailable, THEN **request** the information from the control room via the three-way communicator.

8.2.4 Containment Hydrogen Concentration

- ERCS point 1U5021A[2U5021A] available on the SAS containment panel (code:XT14)
- IF ERCS is unavailable, THEN **request** the information from the control room via the three-way communicator.

8.2.5 Containment Radiation Level

- ERCS point 1U5022A[2U5022A] available on the SAS containment panel (code:XT14)
- IF ERCS is unavailable, THEN **request** the information from the control room via the three-way communicator.

8.2.6 Containment Sump Level

- ERCS point 1U5017A[2U5017A] available on the SAS normal operation panel (code: XS11)
- IF ERCS is unavailable, THEN **request** the information from the control room via the three-way communicator.

8.2.7 Containment Sump Temperature

- When considering the containment sump liquid mass, assume containment sump temperature is less than 200 F. In the event of a large break LOCA, USAR Appendix K, Figure K-15 (upper curve) can be used to estimate the sump temperature.

8.2.8 RVLIS Level

- ERCS point 1U5011A[2U5011A] available on the SAS Primary Core Cooling Panel (code: XT12)
- IF ERCS is unavailable, THEN **request** the information from the control room via the three-way communicator.

8.2.9 RCS Temperature

- ERCS panel "RCS temperature" (code: XT21). Use the lowest available water temperature.
- IF ERCS is unavailable, THEN **request** the information from the control room via the three-way communicator.

8.3 Perform a core damage assessment according to the instructions in SWI NE-5 (23). Continue with Step 8.15 of this procedure when the CDA run is complete.

NOTE:	**If the computer is not available, perform the following manual calculations to obtain core damage estimates.**

8.4 Decay correct the specific activities determined by the sample analysis, back to the time of reactor shutdown, as follows:

NOTE:	**The decay correction may have been accomplished by the computer during the spectrum analysis. Therefore, this step may not need to be completed.**

$$A_0 = \frac{A}{e^{-\lambda_i t}}$$

Where:

A = measured specific activity, μCi/gm or μCi/cc

λ_i = decay constant of isotope i, sec^{-1}; where $\lambda_i = 0.693/(\text{half-life})_i$

t = time elapsed from reactor shutdown to time of sampling, sec.

A_0 = decay corrected specific activity μCi/gm or μCi/cc

8.5 If a parent-daughter relationship exists for a specific isotope, the following steps should be followed to calculate the fraction of the measured activity due to the decay of the daughter that was released and then to calculate the activity of the daughter released at shutdown.

8.5.1 Calculate the hypothetical daughter concentration (Q_B) at the time of the sample analysis assuming 100 percent release of the parent and daughter source inventory:

$$Q_B(t) = K_i \frac{\lambda_B}{\lambda_B - \lambda_{Ai}} Q^o_{Ai}\left(e^{-\lambda_{Ai}t} - e^{-\lambda_B t}\right) + Q^0_B\, e^{-\lambda_B t}$$

Where:

Q^0_{Ai}	=	100% source inventory (Ci) of parent i, Table 2 or Table 4.
Q^0_B	=	100% source inventory (Ci) of daughter, Table 2 or Table 4.
Q_B (t)	=	hypothetical daughter activity (Ci) at sample time.
K_i	=	if parent has 2 daughters, K_i is the branching factor, Table 3.
λ_{Ai}	=	decay constant of parent i, sec $^{-1}$; where $\lambda_{Ai} = 0.693/(\text{half-life})_{Ai}$
λ_B	=	daughter decay constant, sec $^{-1}$; where $\lambda_B = 0.693/(\text{half-life})_B$
t	=	time period from shutdown to time sample, sec.

8.5.2 Determine the contribution of only the decay of the initial inventory of the daughter to the hypothetical daughter activity at sample time:

$$Fr = \frac{Q^0_B\, e^{-\lambda_B t}}{Q_B(t)}$$

8.5.3 Calculate the amount of decay corrected sample specific activity associated with just the daughter that was released.

$$M^0_B = Fr \times A_0$$

Where: A_0 = decay corrected specific activity (μCi/gm or μCi/cc) as determined by the analysis.

8.6 Determine the total volume or mass of the medium which was sampled.

8.6.1 Containment Volume:

V = containment free volume (cc's)

= 3.74×10^{10} cc's

NOTE:	**When considering the containment sump liquid mass, assume containment sump temperature is less than 200°F. In the event of a large break LOCA, USAR Appendix K, Figure K-15 (upper curve) can be used to estimate the sump temperature.**

8.6.2 Liquid Mass:

A. Liquid temperature < 200°F

$$\text{Mass (gms)} = \text{volume } (ft^3) \times \rho_{STP} \times \frac{28.3 \times 10^3 cc}{ft^3}$$

Where: ρ_{STP} = water density at STP = 1.0 gm/cc

B. Liquid temperature > 200°F

$$\text{Mass (gms)} = \text{volume } (ft^3) \times \frac{\rho}{\rho_{STP}} (2) \times \rho_{STP} \times \frac{28.3 \times 10^3 cc}{ft^3}$$

Where: $\frac{\rho}{\rho_{STP}}$ (2) = water density ratio at medium temperature, from Figure 1

ρ_{STP} = water density at STP = 1.0 gm/cc

8.7 Determine the total activity of each isotope in each medium.

8.7.1 Containment Atmosphere:

$$\text{Total containment Activity (curies)} = A_0\ (\mu\text{Ci/cc})\ \text{X}\ V\ (\text{cc's})\ \text{X}\ \frac{\text{Curie}}{1 \text{X} 10^6 \mu\text{Ci}}$$

Where: A_0 = Specific activity of containment atmosphere (μCi/cc), decay corrected to time of reactor shutdown and temperature/pressure corrected.

V = containment free volume (cc's)

= $3.74 \text{ X } 10^{10}$ cc's

8.7.2 Liquid Sample:

$$\text{Total Liquid Activity (Curies)} = \text{Liquid MASS (gms)}\ \text{x}\ A_0\ (\mu\text{Ci/cc})\ \text{x}\ \frac{\text{Curie}}{1 \text{X} 10^6 \mu\text{Ci}}$$

Where: A_0 = Specific activity of liquid sample (μCi/gm), decay corrected to time of reactor shutdown.

8.8 The approximate total activity of each isotope in the liquid samples can now be calculated.

Total Water Activity = RCS Activity + Sump Activity + Activity Leaked to Secondary System.

8.9 Now the total activity of each isotope released at the time of the accident can be determined:

Total Activity Released = Total Water Activity + Containment Atmosphere Activity

8.10 Utilizing the total activity of each isotope released, calculate the activity ratios of the released fission products.

8.10.1 Noble Gas Ratio $= \dfrac{\text{Noble Gas Activity}}{\text{Xe} - 133 \text{ Activity}}$

8.10.2 Iodine Ratio $= \dfrac{\text{Iodine Activity}}{\text{I} - 131 \text{ Activity}}$

NOTE:	**Steady state power conditions may be assumed where power does not vary by more than ± 10% from the time averaged value.**

8.11 Determine the power history prior to reactor shutdown.

8.12 Using the power history, determine a power correction factor for each isotope, in accordance with the following guidelines:

NOTE:	**Steady state power condition is assumed where the power does not vary by more than ± 10% from the time averaged value.**

8.12.1 Steady State power prior to shutdown.

A. Half-life of nuclide < 1 day

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for Prior 4 Days}}{\text{Power Level (Mwt) used in Table 2}}$$

B. Half-life of nuclide > 1 day

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for Prior 30 Days}}{\text{Power Level (Mwt) used in Table 2}}$$

C. Half-life of nuclide ~ 1 year

$$\text{Power Correction Factor} = \frac{\text{Average Power Level (Mwt) for Prior 1 year}}{\text{Power Level (Mwt) used in Table 2}}$$

8.12.2 Transient power history in which the power has not remained constant prior to reactor shutdown.

NOTE:	For the majority of the selected nuclides, the 30-day power history prior to shutdown is sufficient to calculate a power correction factor.

A. Power Correction Factor $= \dfrac{\Sigma_j \; P_j \left(1 - e^{-\lambda_i t_j}\right) e^{-\lambda_i t^0_j}}{RP}$

P_j = average power level (Mwt) during operating period t_j

RP = power level (Mwt) used in Table 2

t_j = operating period in days at power P_j where power does not vary more than ±10 percent power from the time averaged value (P_j).

λ_I = decay constant of nuclide I in inverse days. where $\lambda_I = 0.693/(\text{half-life})_I$

T^0_j = time between end of period j and time of reactor shutdown in days.

B. For the few nuclides with half-lives around one year or longer, a power correction factor which ratios effective full power days to total calendar days of cycle operation is applied.

$$\text{Power Correction Factor} = \frac{\text{Actual Operating EFPD of equilibrium cycle}}{\text{Total expected EFPD of equilibrium cycle operation}}$$

Where: Equilibrium Cycle = three (3) cycles of core operation

8.12.3 For Cs-134, Figure 2 is used to determine the power correction factor. To use Figure 2, the average power during the entire operating period is required.

8.13 The total inventory of fission products available for release at reactor shutdown are calculated by applying the power correction factors to the equilibrium, end-of-life core inventories.

$$\text{Corrected Inventory} = \begin{array}{c}\text{Equilibrium Inventory at}\\ \text{end-of-life (Ci)}\\ \text{(Table 2)}\end{array} \quad X \quad \begin{array}{c}\text{Power}\\ \text{Correction}\\ \text{Factor}\end{array}$$

8.14 Determine the percentage of inventory released, for each isotope.

$$\text{Release Percentage (\%)} = \frac{\text{Total Activity Released (Ci)}}{\text{Corrected Inventory (Ci)}} X\,100$$

8.15 The results of radionuclide analysis may now be used to determine an estimate of the extent of core damage.

8.15.1 From Figure 3 thru 15, estimate the extent of core damage by categorizing the percentage of clad damage, fuel over-temperature, and fuel melt.

8.15.2 Compare the calculated activity ratios with those listed in Table 5. Measured relative ratios greater than the gap activity ratios listed in Table 5 are indicative of more severe failures, e.g., fuel overheat.

NOTE:	The plant indicators described in this procedure are available via an ERCS Group Tabular Display for each unit. See ERCS Main Menu – Tabular Display – Load Group – 1_F3-17-1 or 2_F3-17-1.

8.16 To verify the conclusion of the radionuclide analysis, other indicators should now be used to provide verification of the estimate of core damage.

8.16.1 Containment Hydrogen Concentration:

A. Obtain the containment hydrogen concentration (%).

NOTE:	Within the accuracy of this methodology, it is assumed that recombiners will have an insignificant effect on the hydrogen concentration when it is indicated that extensive zirconium-steam reaction could have occurred.

B. From Figure 16, determine the percentage (%) zirconium water reaction.

C. Table 6 can be used to validate the extent of core damage estimate.

8.16.2 Core Exit thermocouple Readings:

NOTE:	For core exit thermocouple temperatures, the maximum temperatures achieved during the event should be used in determining core damage.

A. Obtain as many core exit thermocouple readings as possible for evaluation of core temperature conditions.

NOTE:	If a thermocouple reads greater than 1650°F or is reading considerably different than neighboring thermocouples, thermocouple failure should be considered.

B. Compare the thermocouple readings with those in Table 6 to confirm the core damage estimate.

NOTE:	Radiation Monitors in containment may experience errors during first 4 hours after a DBA LOCA due to thermally induced errors. See Attachment 1 for more information.

8.16.3 Containment Radiation Monitor:

A. Obtain the containment dome monitor readings, in R/Hr, from R-48 and/ or R-49.

B. From Figure 17, verify core damage estimate. The exposure rate in Figure 17 is based on the release of only noble gases to the containment. Halogens and other fission products were not considered to be significant contributors to the containment monitor reading.

8.17 All indicators should confirm any core damage estimates. If radio-nuclide analysis and auxiliary indicators do not agree on core damage estimates, then recheck of indications may be performed, or certain indicators may be discounted, based on engineering judgment.

9.0 REFERENCES

NONE

10.0 ATTACHMENTS

10.1 Table 1 – Suggested Sampling Locations

10.2 Table 2 – Fuel Pellet Inventory

10.3 Table 3 – Parent-Daughter Relationships

10.4 Table 4 – Source Inventory of Related Parent Nuclides

10.5 Table 5 – Isotopic Activity Ratios of Fuel Pellet and Gap

10.6 Table 6 – Characteristics of Categories of Fuel Damage

10.7 Table 7 – Expected Fuel Damage Correlation With Fuel Rod

10.8 Figure 1 – Water Density Ratio (Temperature vs. STP)

10.9 Figure 2 – Power Correction Factor For Cs-134 Based on Average Power During Operation

10.10 Figure 3 – Relationship of % Clad Damage With % Core Inventory Released of Xe-133

10.11 Figure 4 – Relationship of % Clad Damage With % Core Inventory Released of I-131

10.12 Figure 5 – Relationship of % Clad Damage With % Core Inventory Released of I-131 W/Spiking

10.13 Figure 6 – Relationship of % Clad Damage With % Core Inventory Released of Kr-87

10.14 Figure 7 – Relationship of % Clad Damage With % Core Inventory Released of Xe-131M

10.15 Figure 8 – Relationship of % Clad Damage With % Core Inventory Released of I-132

10.16 Figure 9 – Relationship of % Clad Damage With % Core Inventory Released of I-133

10.17 Figure 10 – Relationship of % Clad Damage With % Core Inventory Released of I-135

10.18 Figure 11 – Relationship of % Fuel Over Temperature With % Core Inventory Released of Xe, Kr, I, or Cs

10.19 Figure 12 – Relationship of % Fuel Over Temperature With % Core Inventory Released of Ba or Sr

10.20 Figure 13 – Relationship of % Fuel Melt With % Core Inventory Released of Xe, Kr, I, Cs or Te

10.21 Figure 14 – Relationship of % Fuel Melt With % Core Inventory Released of Ba or Sr

10.22 Figure 15 – Relationship of % Fuel Melt With % Core Inventory Released of Pr

10.23 Figure 16 – Containment Hydrogen Concentration Based on Zirconium Water Reaction

10.24 Figure 17 – Percent Noble Gases in Containment

10.25 Attachment 1 – Thermally Induced Current Errors in Containment Radiation Monitors

Table 1 Suggested Sampling Locations

Scenario	Principal Sampling Locations	Other Sampling Locations
Small Break LOCA Reactor Power > 1%* Reactor Power < 1%*	 RCS Hot Leg, Containment Atmosphere RCS Hot Leg**	
Large Break LOCA Reactor Power > 1%* Reactor Power < 1%*	 Containment Sump, Containment Atmosphere, RCS Hot Leg Containment Sump, Containment Atmosphere	
Steam Line Break	RCS Hot Leg,	Containment Atmosphere
Steam Generator Tube Rupture	RCS Hot Leg, Secondary System	
Indication of Significant Containment Sump Inventory	Containment Sump, Containment Atmosphere	
Containment Building Radiation Monitor Alarm	Containment Atmosphere, Containment Sump	
Safety Injection Actuated	RCS Hot Leg	
Indication of High Radiation Level in RCS	RCS Hot Leg	

* Assume operating at that level for some appreciable time.

** If a RCS hot leg sample is unavailable and the RHR system is operating, obtain a RHR system sample. However, for a RHR system sample to be a good representation of the RCS, the primary water should be circulating through the system.

Table 2 Fuel Pellet Inventory

Fuel Pellet Inventory*

Nuclide	**Half Life**	**Inventory Curies****
Kr 85m	4.4 h	1.0×10^7
Kr 87	76 m	1.85×10^7
Kr 88	2.8 h	2.69×10^7
Xe 131m	11.8 d	2.94×10^5
Xe 133	5.27 d	9.26×10^7
Xe 133m	2.26 d	1.35×10^7
Xe 135	9.14 h	1.77×10^7
I 131	8.05 d	4.54×10^7
I 132	2.26 h	6.65×10^7
I 133	20.3 h	9.26×10^7
I 135	6.68 h	8.33×10^7
Rb 88	17.8 m	2.69×10^7
Cs 134	2 yr	1.09×10^7
Cs 137	30 yr	4.96×10^6
Te 129	68.7 m	1.51×10^7
Te 132	77.7 h	6.65×10^7
Sr 89	52.7 d	3.70×10^7
Sr 90	28 yr	3.36×10^6
Ba 140	12.8 d	7.91×10^7
La 140	40.22 h	8.33×10^7
La 142	92.5 m	7.07×10^7
Pr 144	17.27 m	5.81×10^7

1. Inventory based on ORIGEN run for equilibrium, end-of-life core.
   ** Westingouse, 2-Loop 1650 Mwt Plant

Table 3 Parent-Daughter Relationships

Parent	Parent Half-Life*	Daughter	Daughter Half-Life*	K**
Kr-88	2.8 h	Rb-88	17.8 m	1.00
I-131	8.05 d	Xe-131m	11.8 d	.008
I-133	20.3 h	Xe-133m	2.26 d	.024
I-133	20.3 h	Xe-133	5.27 d	.976
Xe-133m	2.26 d	Xe-133	5.27 d	1.00
I-135	6.68 h	Xe-135	9.14 h	.70
Xe-135m	15.6 m	Xe-135	9.14 h	1.00
I-135	6.68 h	Xe-135m	15.6 m	.30
Te-132	77.7 h	I-132	2.26 h	1.00
Sb-129	4.3 h	Te-129	68.7 m	.827
Te-129m	34.1 d	Te-129	68.7 m	.680
Sb-129	4.3 h	Te-129m	34.1 d	.173
Ba-140	12.8 d	La-140	40.22 h	1.00
Ba-142	11 m	La-142	92.5 m	1.00
Ce-144	284 d	Pr-144	17.27 m	1.00

* Table of Isotopes, Lederer, Hollander, and Perlman, Sixth Edition

** Branching decay factor

Table 4 Source Inventory of Related Parent Nuclides

Nuclide	Half-Life	Inventory, Curies
Xe-135m	15.6 m	1.97×10^7
Sb-129	4.3 h	1.49×10^7
Te-129m	34.1 d	3.74×10^6
Ba-142	11 m	7.65×10^7
Ce-144	284 d	4.83×10^7

Table 5 Isotopic Activity Ratios of Fuel Pellet and Gap

Isotopic Activity Ratios of Fuel Pellet and Gap*

Nuclide	Fuel Pellet Activity Ratio	Gap Activity Ratio
Kr-85m	0.11	0.022
Kr-87	0.22	0.022
Kr-88	0.29	0.045
Xe-131m	0.004	0.004
Xe-133	1.0	1.0
Xe-133m	0.14	0.096
Xe-135	0.19	0.051
I-131	1.0	1.0
I-132	1.5	0.17
I-133	2.1	0.71
I-135	1.9	0.39

$$\text{Noble Gas Ratio} = \frac{\text{Noble Gas Isotope Inventory}}{\text{Xe-133 Inventory}}$$

$$\text{Iodine Ratio} = \frac{\text{Iodine Isotope Inventory}}{\text{I-131 Inventory}}$$

* The measured ratios of various nuclides found in reactor coolant during normal operation is a function of the amount of "tramp" uranium on fuel rod cladding, the number and size of "defects" (i.e., "pin holes"), and the location of the fuel rods containing the defects in the core. The ratios derived in this report are based on calculated values of relative concentrations in the fuel or in the gap. The use of these present ratios for post accident damage assessment is restricted to an attempt to differentiate between fuel overtemperature conditions and fuel cladding <u>failure</u> conditions. Thus the ratios derived here are <u>not</u> related to fuel <u>defect</u> levels incurred during normal operation.

Table 6 Characteristics of Categories of Fuel Damage

Core Damage Category	Percent and Type of Fission Products Released	Fission Product Ratio***	Containment Radiogas Monitor R/hr 10 hrs after shutdown**	Core Exit Thermocouples Reading (Deg F)	Core Uncovery Indication	Hydrogen Monitor (Vol % H_2)
No clad damage	Kr-87 $< 1 \times 10^{-3}$ Xe-133 $< 1 \times 10^{-3}$ I-131 $< 1 \times 10^{-3}$ I-133 $< 1 \times 10^{-3}$	No Applicable	--	< 750	No uncovery	Negligible
0-50% clad damage	Kr-87 10^{-3} – 0.01 Xe-133 10^{-3} – 0.1 I-131 10^{-3} – 0.3 I-133 10^{-3} – 0. 1	Kr-87 = 0.022 I-133 = 0.71	0 – 50	750 – 1300	Core uncovery	0 – 6
50 – 100% clad damage	Kr-87 0.01 – 0.02 Xe-133 0.1 – 0.2 I-131 0.3 – 0.5 I-133 0.1 – 0.2	Kr-87 = 0.022 I-133 = 0.71	50 to 100	1300 – 1650	Core uncovery	6 – 13
0 – 50% fuel pellet overtemperature	Xe-Kr, Cs, I 1 – 20 Sr-Ba 0 – 0.1	Kr-87 = 0.22 I-133 = 2.1	100 to 1.15E4	> 1650	Core uncovery	6 – 13
50-100% fuel pellet overtemperature	Xe-Kr, Cs, I 20 – 40 Sr-Ba 0.1– 0.2	Kr-87 = 0.22 I-133 = 2.1	1.15E4 to 2.3E4	> 1650	Core uncovery	6 – 13
0 – 50% fuel melt	Xe, Kr, Cs, I 40-70 Sr-Ba 0.2 – 0.8 Pr 0.1 – 0.8	Kr-87 = 0.22 I-133 = 2.1	2.3E4 to 2.7E4	> 1650	Core uncovery	6 – 13
50 – 100% fuel melt	Xe, Kr, Cs, I, Te > 70 Sr, Ba > 24 Pr > 0.8	Kr-87 = 0.22 I-133 = 2.1	> 2.7E4	> 1650	Core uncovery	6 – 13

Characteristics of Categories of Fuel Damage*

* This table is intended to indicate whether there is fuel damage.

** These values are from Figure 17 and should be revised for times other than 10 hours.

*** $\frac{Kr\text{-}87}{xe\text{-}133}, \frac{I\text{-}133}{I\text{-}131}$

Table 7 Expected Fuel Damage Correlation With Fuel Rod Temperature

For Information Only – See Note Below

Fuel Damage	Temperature °F*
No Damage	< 1300
Clad Damage	1300 - 2000
Ballooning of zircaloy cladding	> 1300
Burst of zircaloy cladding	1300 - 2000
Oxidation of cladding and hydrogen generation	> 1600
Fuel Overtemperature	2000 - 3450
Fission product fuel lattice mobility	2000 - 2550
Grain boundary diffusion release of fission products	2450 - 3450
Fuel Melt	> 3450
Dissolution and liquefaction of UO_2 in the Zircaloy - ZrO_2 eutectic	> 3450
Melting of remaining UO_2	5100

* These temperatures are material property characteristics and are non-specific with respect to locations within the fuel and/or fuel cladding.

Figure 1 Water Density Ratio (Temperature vs. STP)

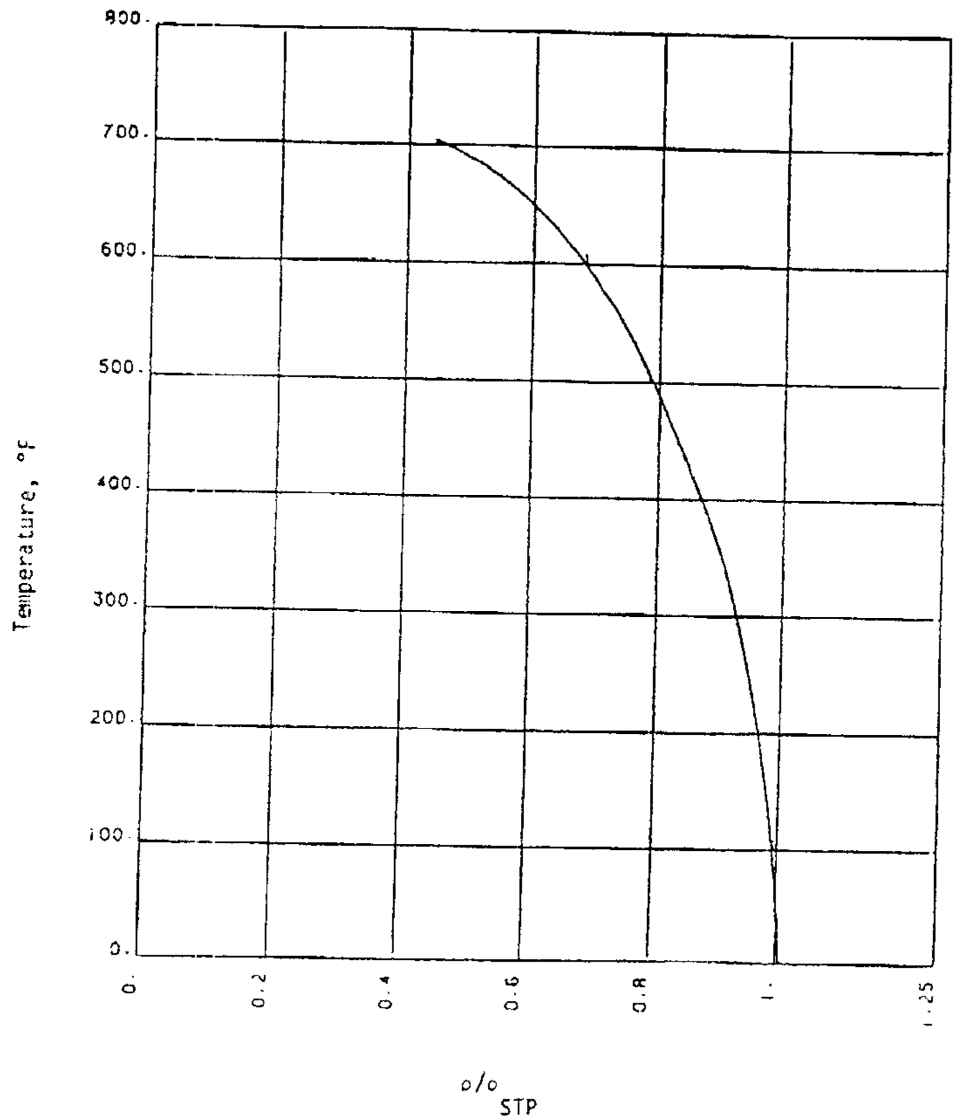


Figure 2 Power Correction Factor For Cs-134 Based on Average Power During Operation

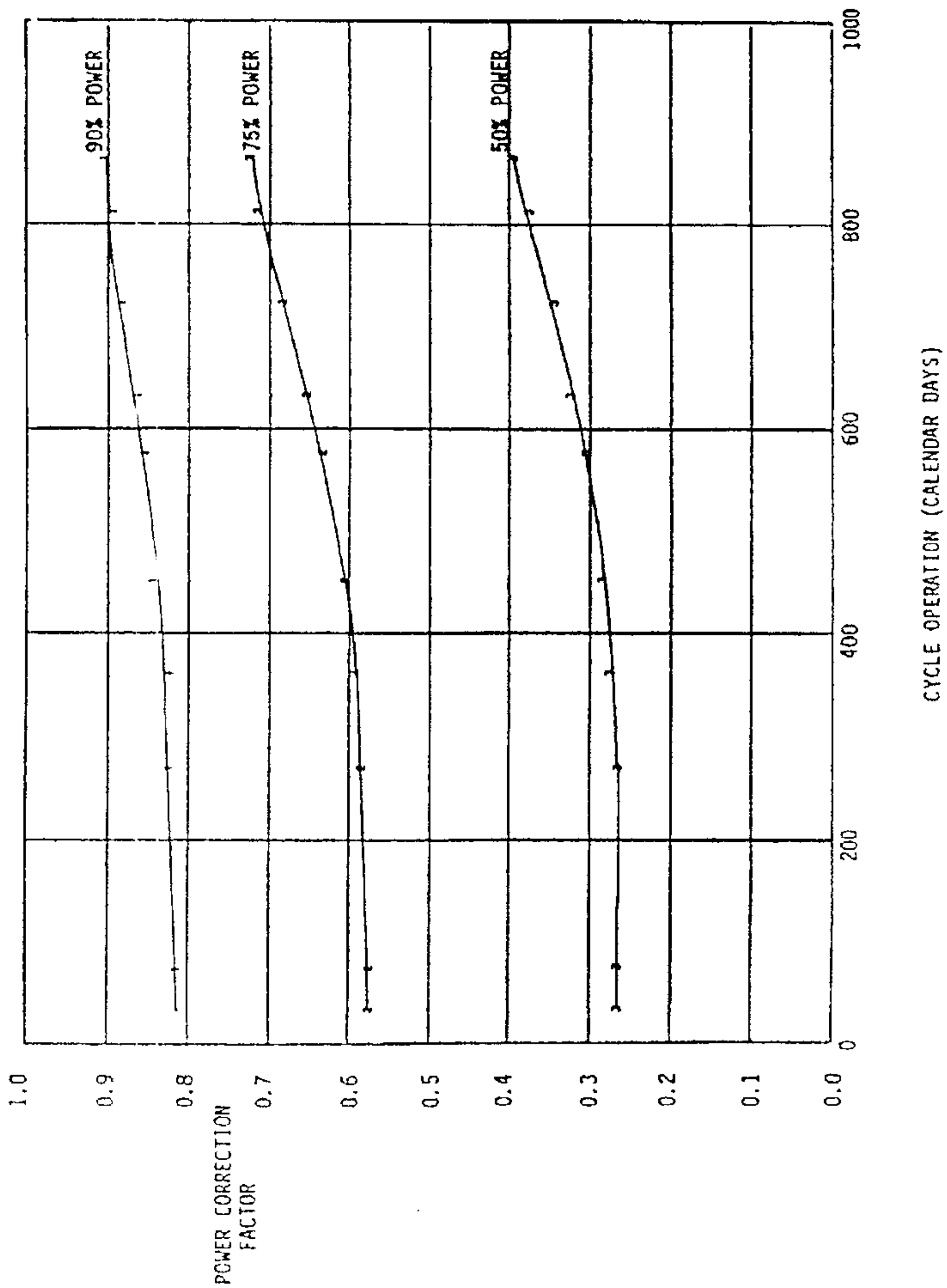


Figure 3 Relationship of % Clad Damage With % Core Inventory Released of Xe-133

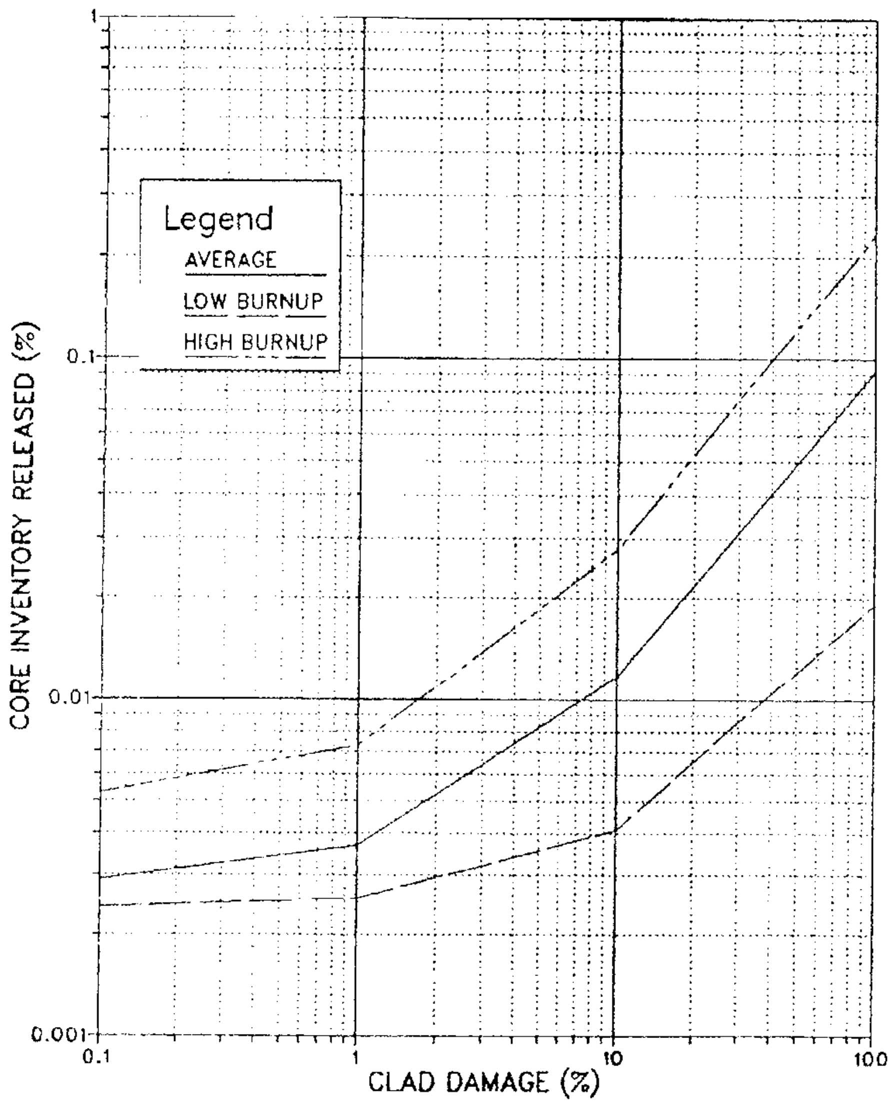


Figure 4 Relationship of % Clad Damage With % Core Inventory Released of I-131

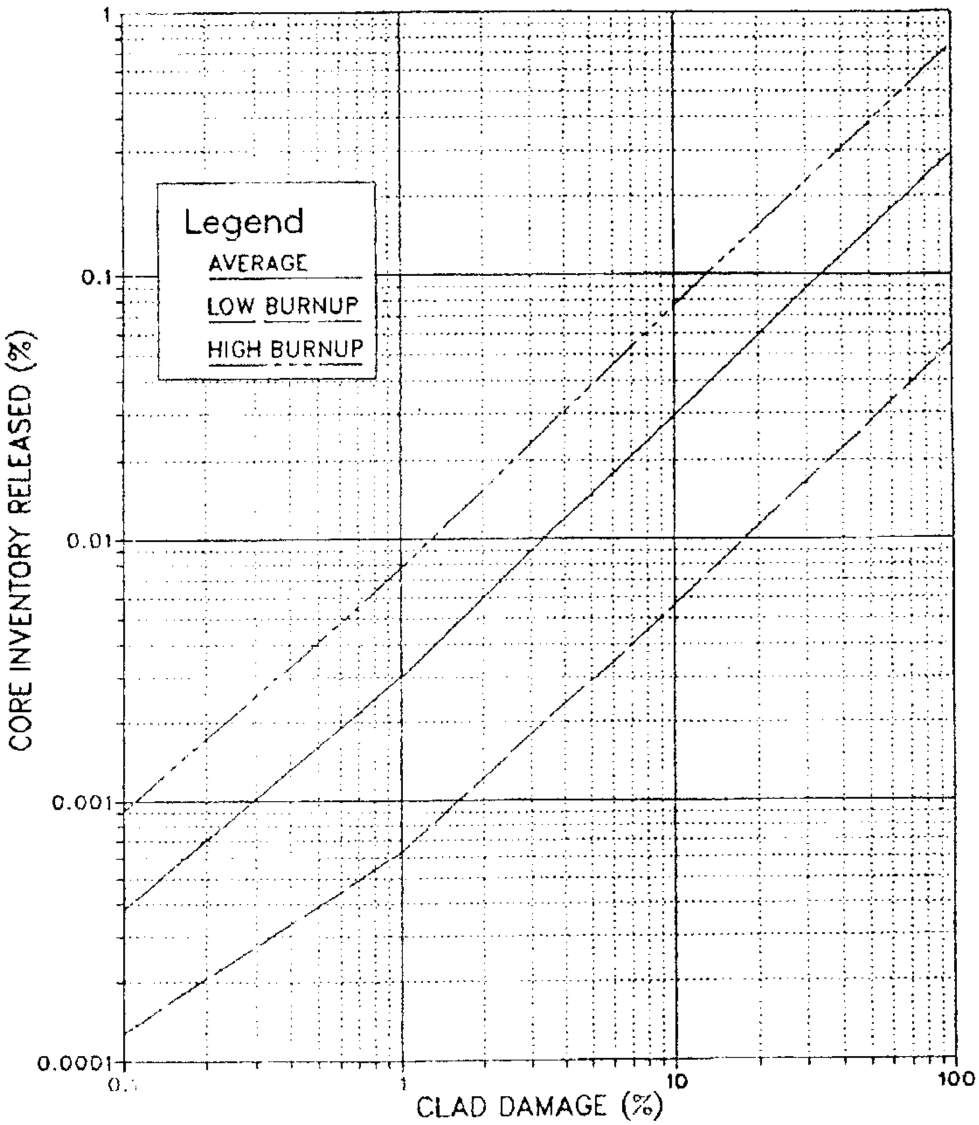


Figure 5 Relationship of % Clad Damage With % Core Inventory Released of I-131 W/Spiking

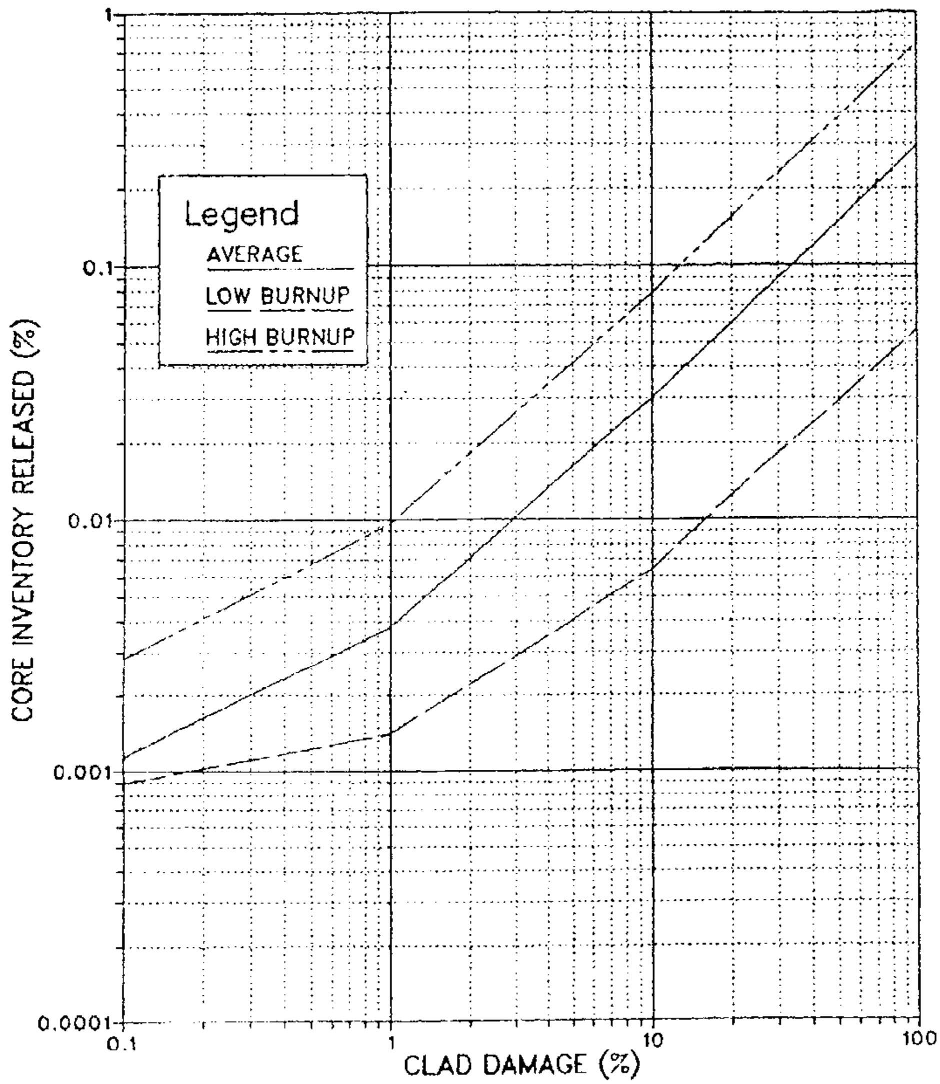


Figure 6 Relationship of % Clad Damage With % Core Inventory Released of Kr-87

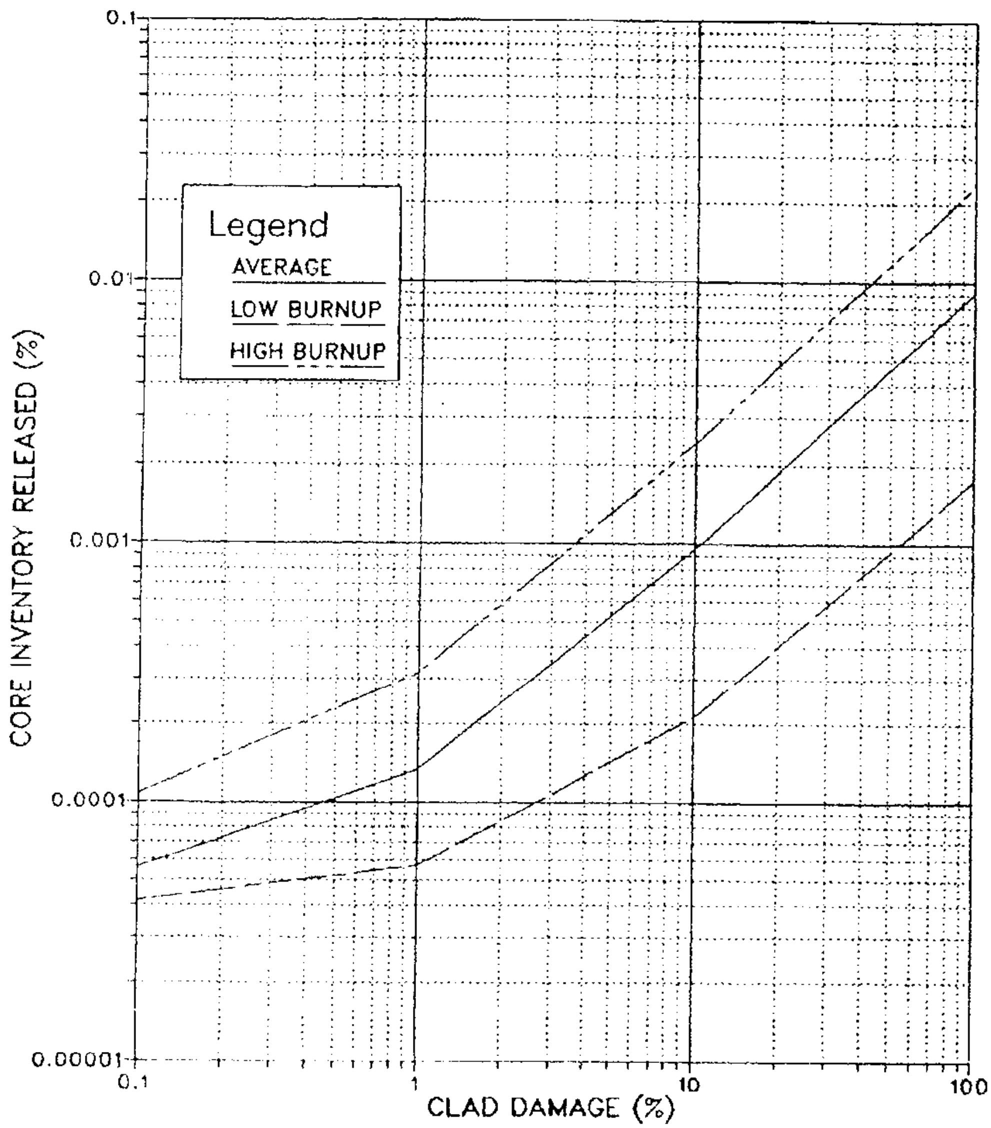


Figure 7 Relationship of % Clad Damage With % Core Inventory Released of Xe-131M

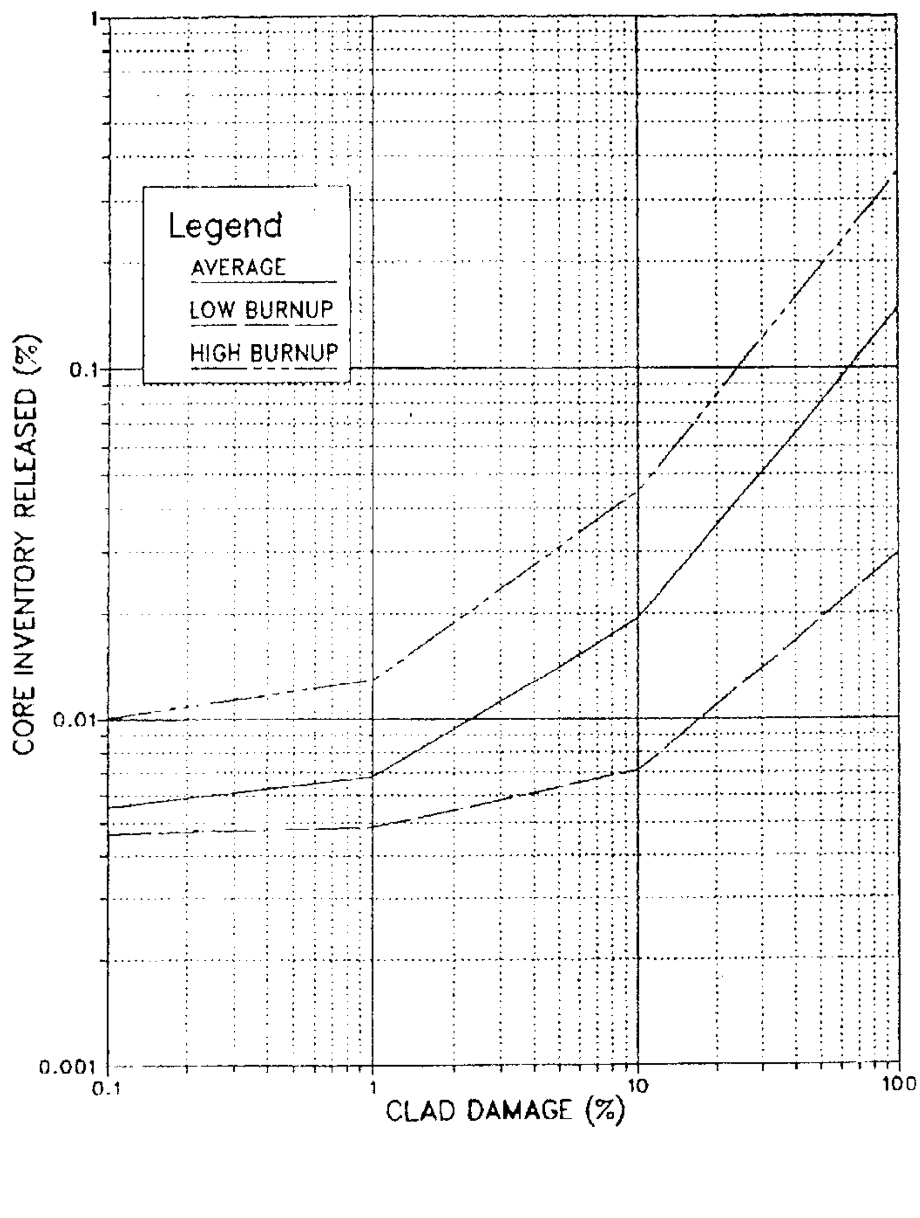


Figure 8 Relationship of % Clad Damage With % Core Inventory Released of I-132

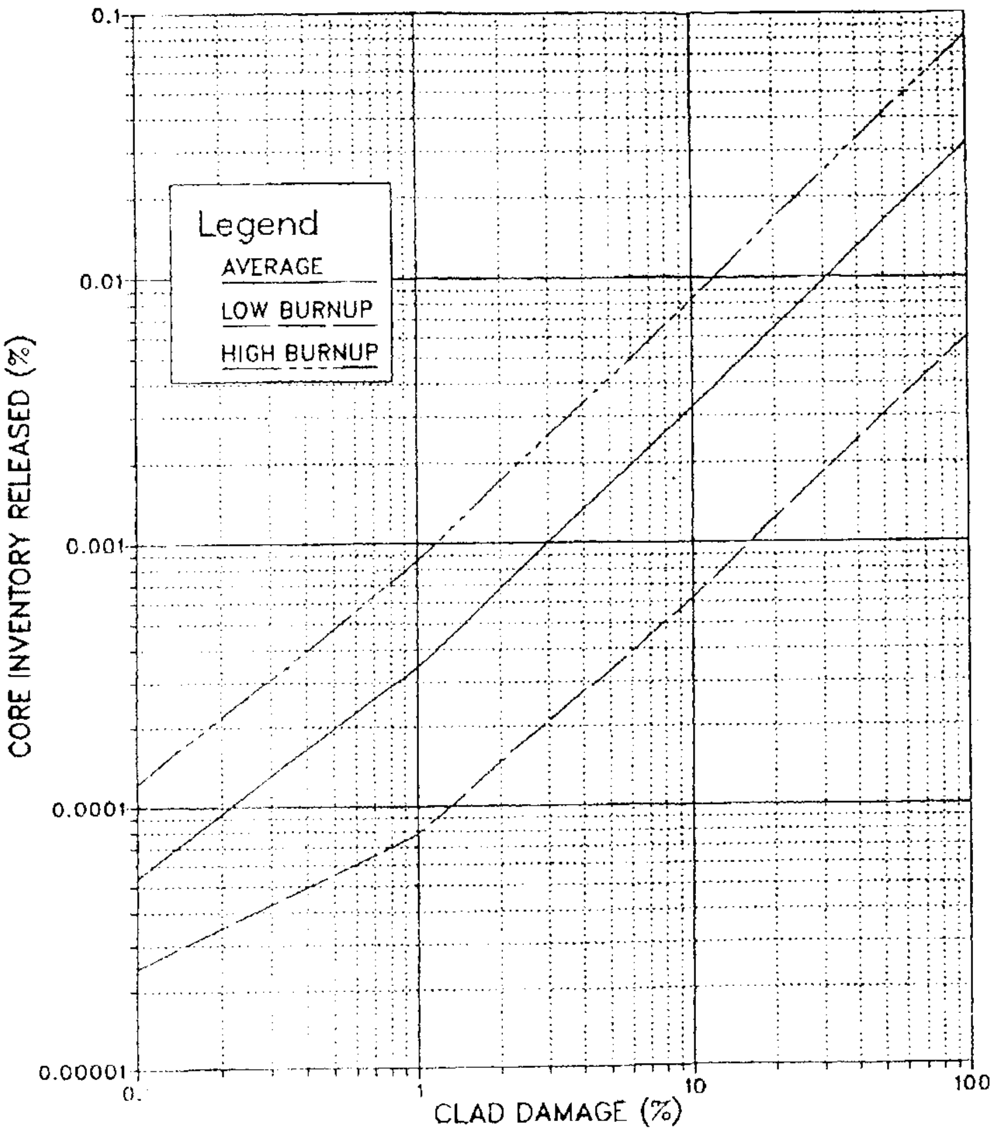


Figure 9 Relationship of % Clad Damage With % Core Inventory Released of I-133

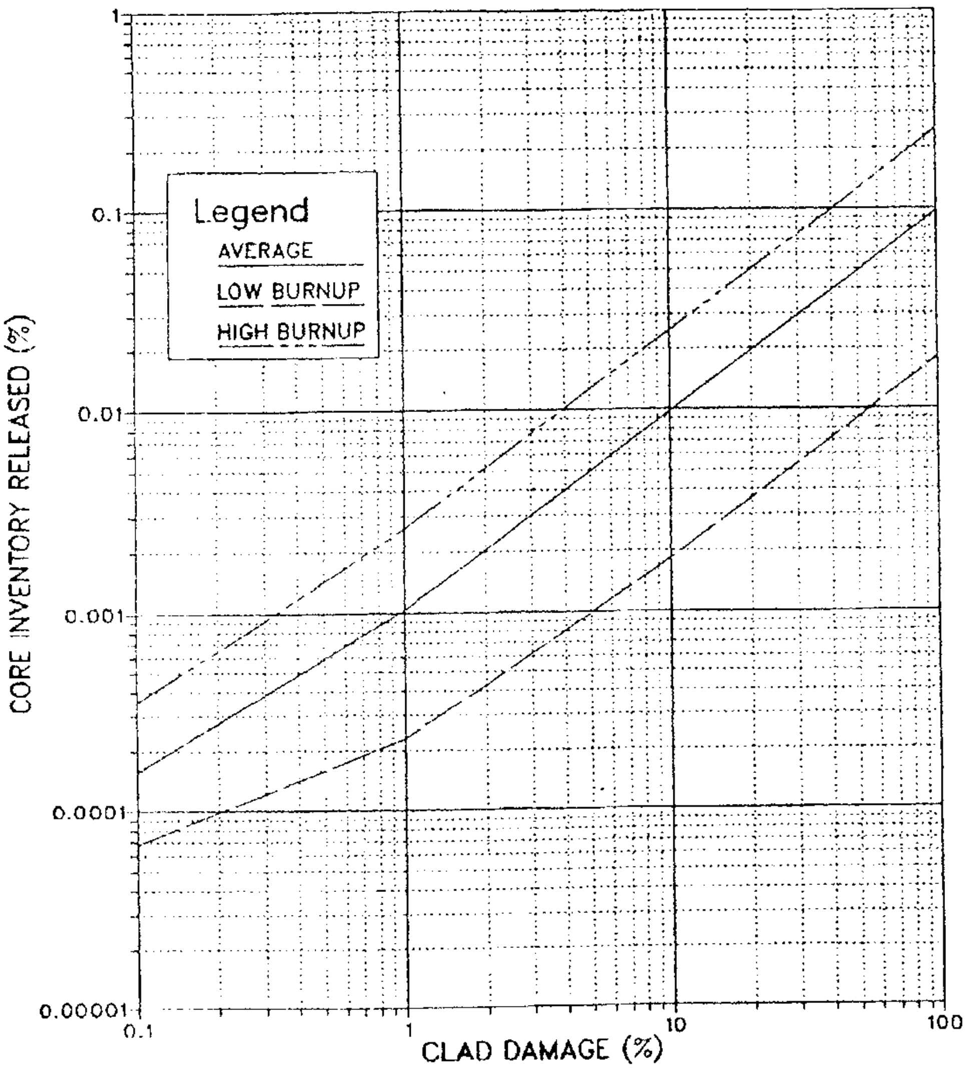


Figure 10 Relationship of % Clad Damage With % Core Inventory Released of I-135

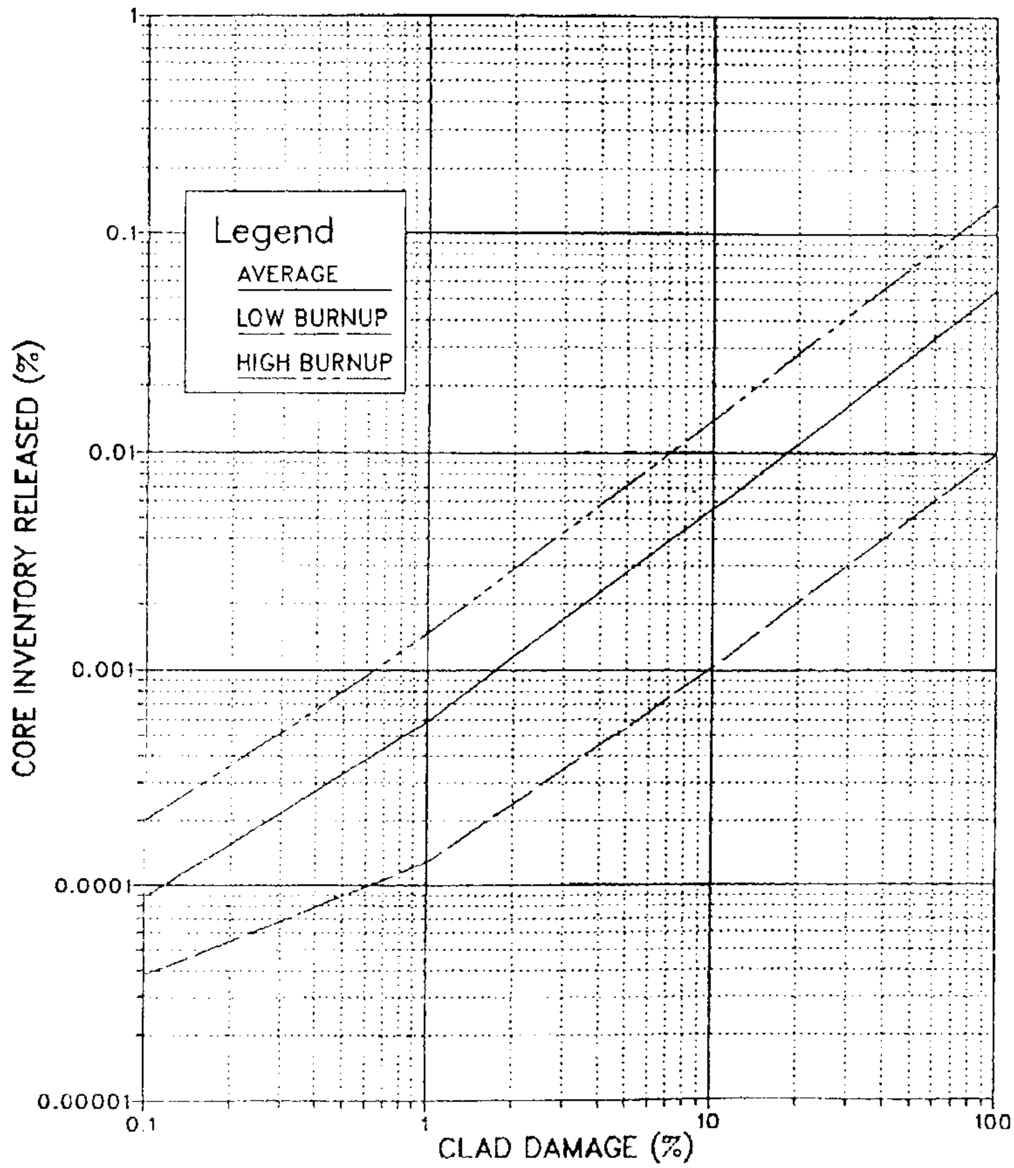
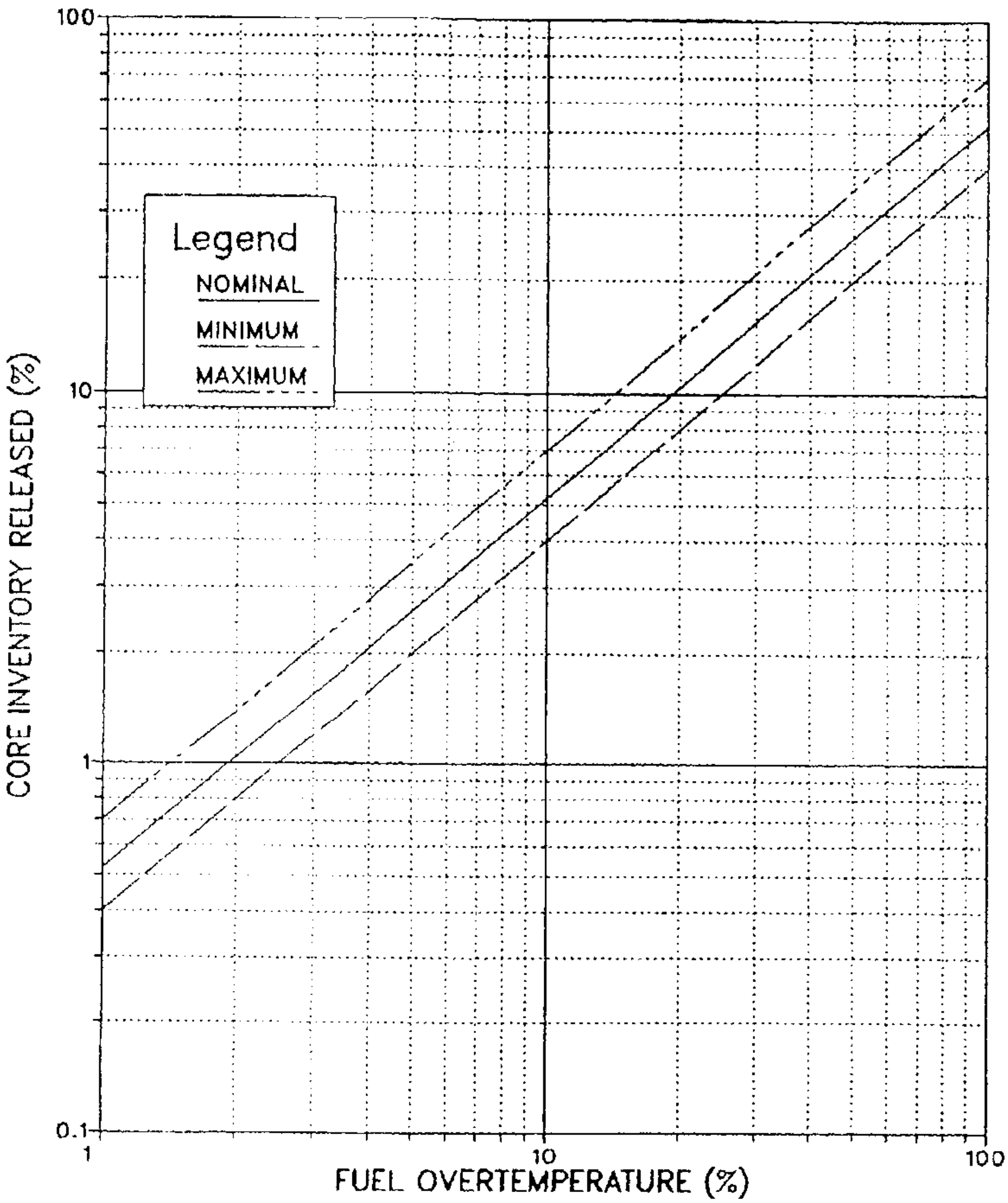


Figure 11 Relationship of % Fuel Over Temperature With % Core Inventory Released of Xe, Kr, I, or Cs



F3	CORE DAMAGE ASSESSMENT	NUMBER: F3-17
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Figure 12 Relationship of % Fuel Over Temperature With % Core Inventory Released of Ba or Sr

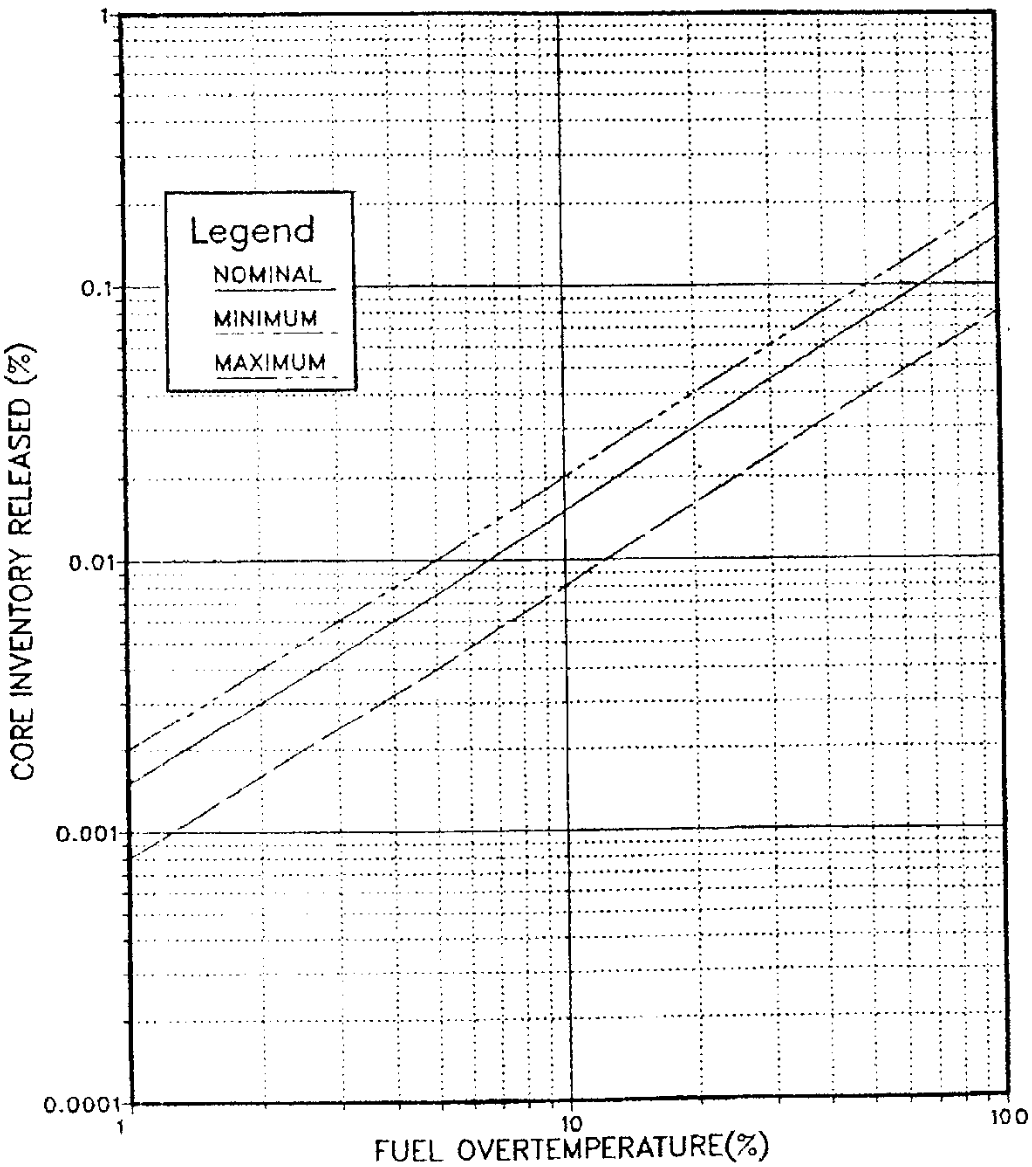


Figure 13 Relationship of % Fuel Melt With % Core Inventory Released of Xe, Kr, I, Cs or Te

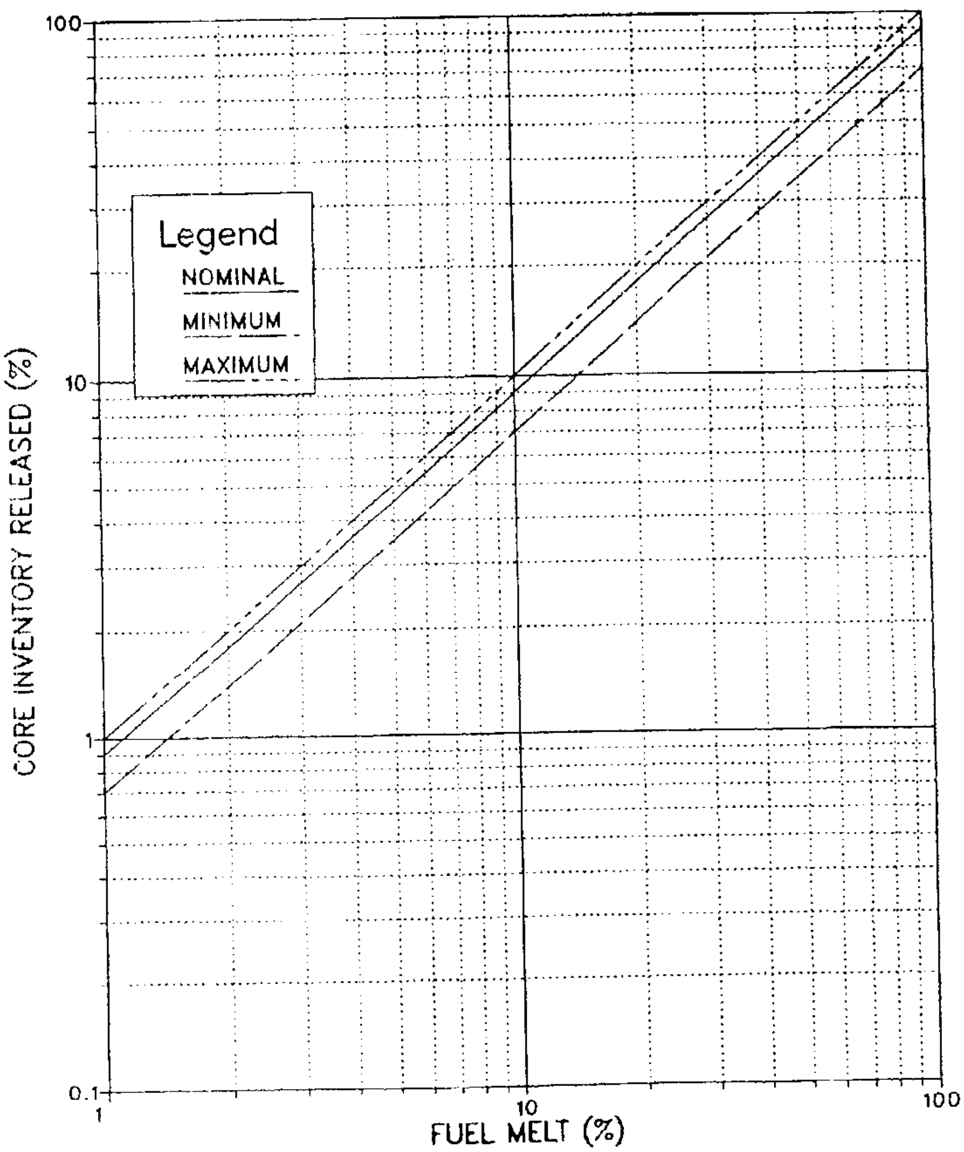


Figure 14Relationship of % Fuel Melt With % Core Inventory Released of Ba or Sr

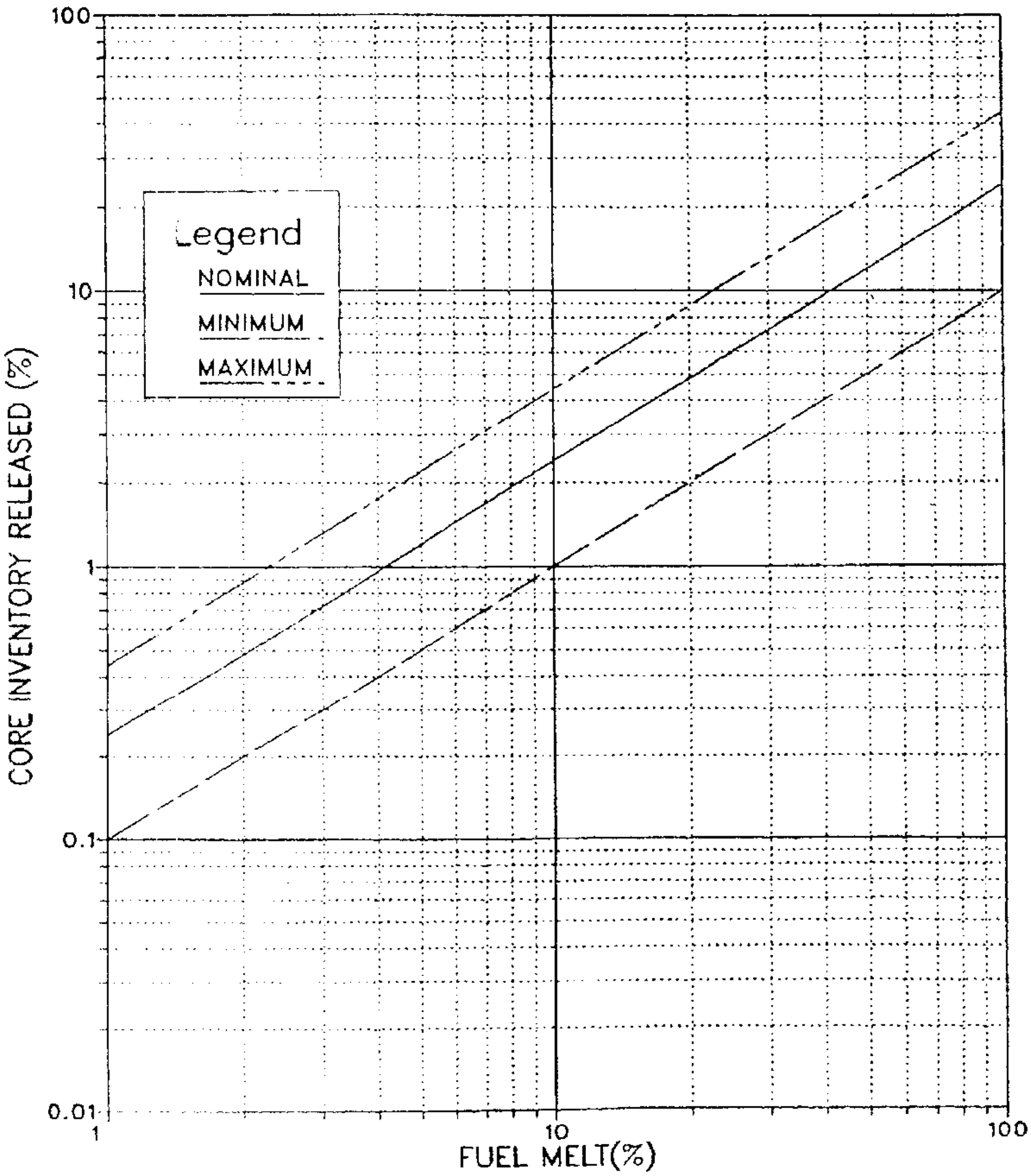


Figure 15 Relationship of % Fuel Melt With % Core Inventory Released of Pr

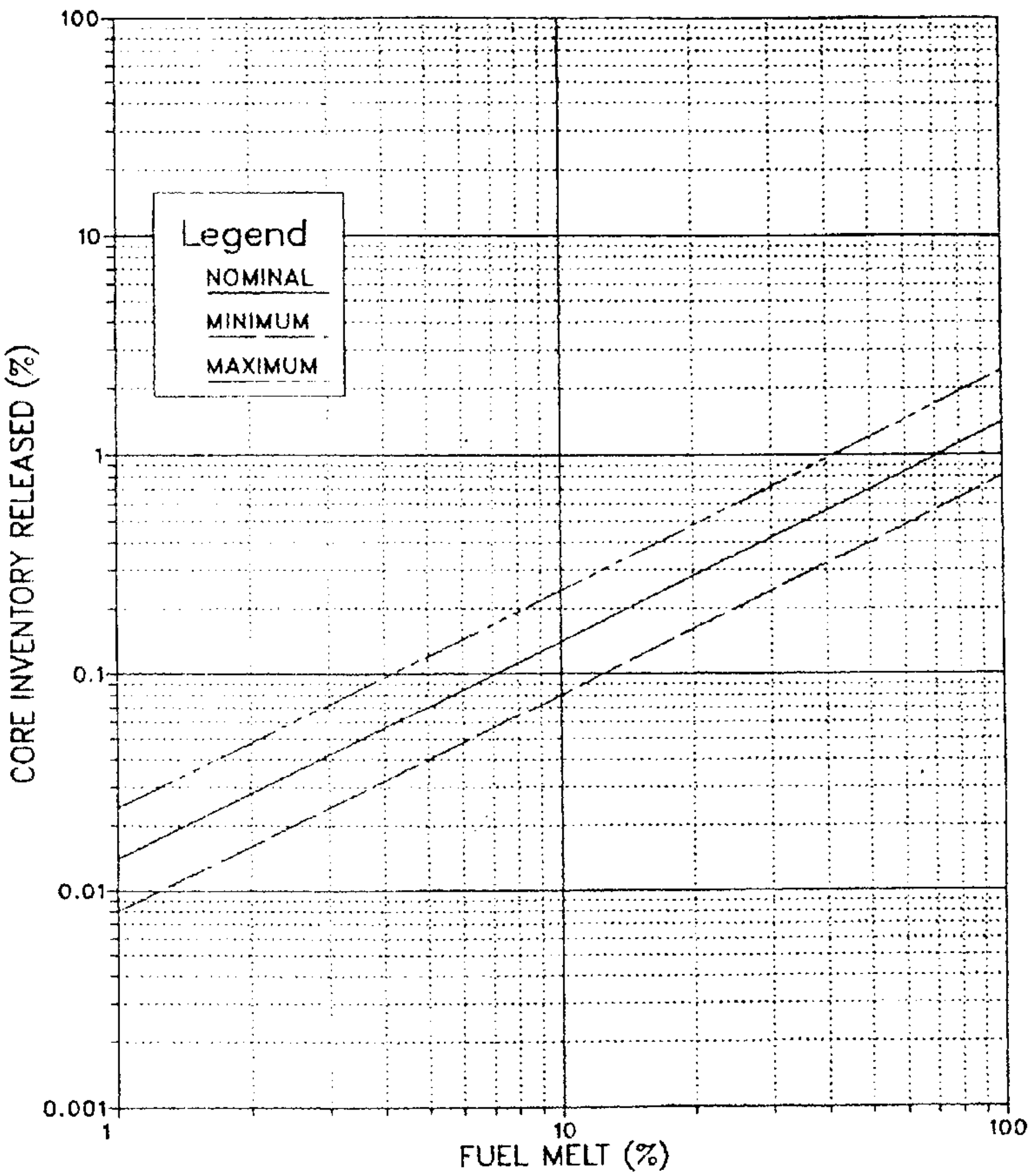


Figure 16 Containment Hydrogen Concentration Based on Zirconium Water Reaction

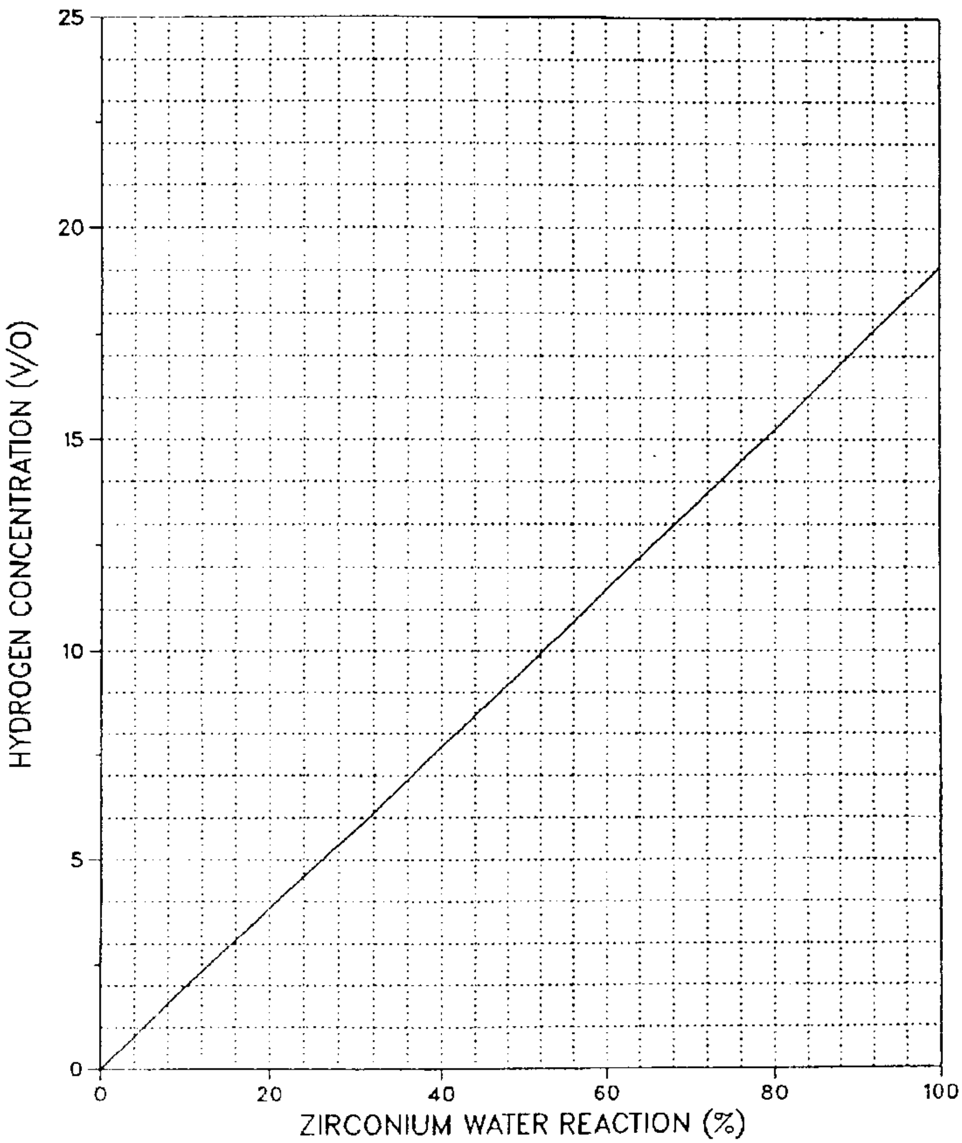
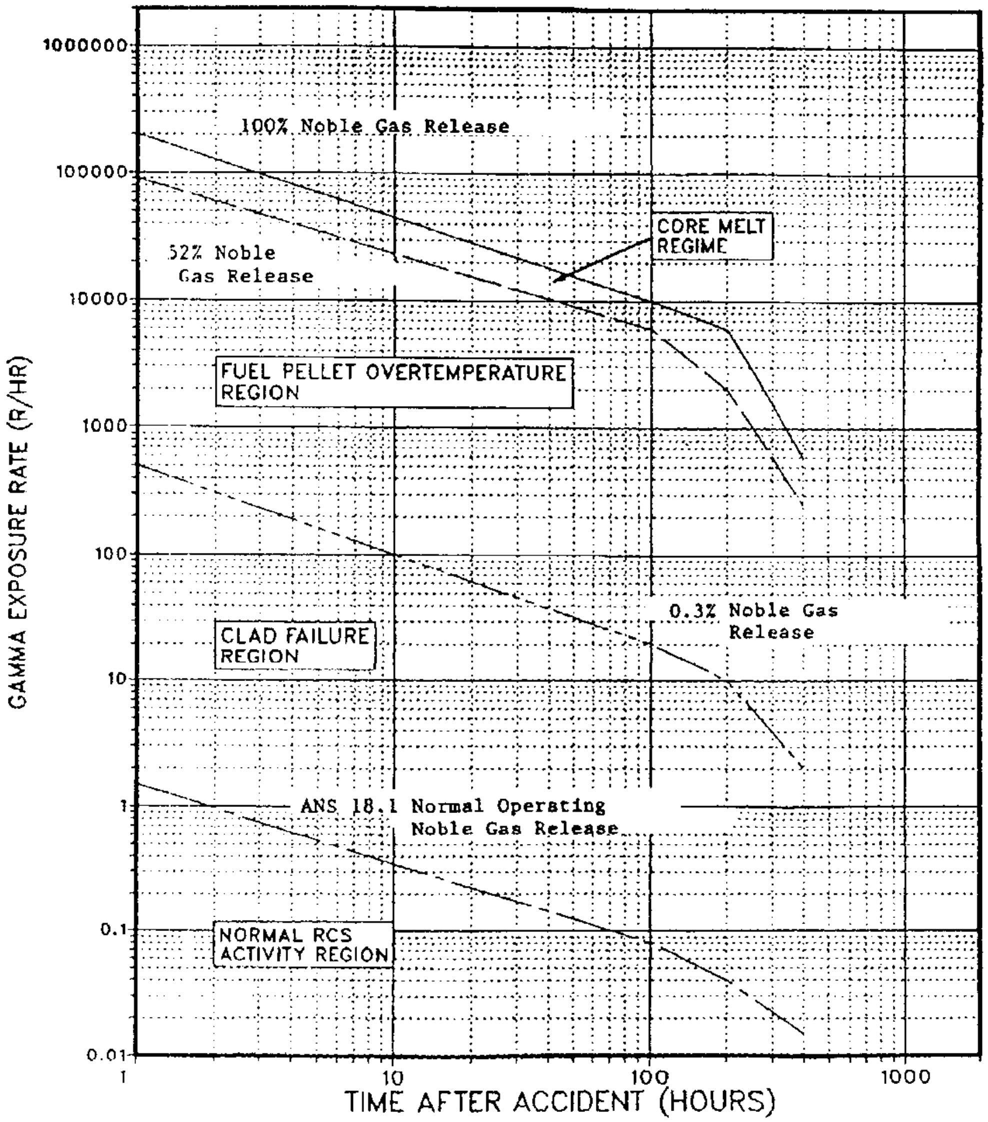


Figure 17 Percent Noble Gases in Containment



Attachment 1 Technical Evaluation High Range Containment Radiation Monitors Thermal Induced Current ECR 608000000176

NOTE:	Sections IV and X discussed in this attachment refer to the sections within the ECR. References listed in brackets can be found in the ECR.

1.0 Background

From recent NRC Environmental Qualification Component Design Bases Inspections (EQ CDBI) several stations were issued green non-cited violations (NCVs) for not adequately documenting operability of their CHRRMs or to restoring the capability to classify emergency action levels [12, 13, 14]. NRC Information Notice 97-45 Supplement 1 had identified a potential transient operational deficiency on the coaxial signal cables associated with CHRRMs from thermally induced currents (TIC). The potential exists for thermally induced currents on the signal cables associated with these monitors to cause erratic and inaccurate readings.

The TIC phenomena appears to be dependent on the temperature change magnitude, rate of change, the type of cabling used for the plant, the service temperature history, and the uniformity of the cable manufacturing run. TR-112582 [19] notes the sources of TIC include trapped space charge in the insulation of coaxial cables and polar properties of a polysulfone layer used in some cables. The thermal resistance of the cable jacket and the high thermal conductivity of the cable braid create a thermal stimulus at the cable dielectric. The thermal wave resulting from an external temperature change expands the dielectric. If free charges are available in the dielectric, a charge is induced on the braid and center conductor of the cable. The charge is dissipated as current into the impedance of the detector electronics. TIC therefore causes small currents to be induced due to temperature differential between the inner and outer conductors of the coax cable. The currents are too small to impact most instrumentation and control cables, are transient in nature, and are only present during significant temperature transients. The current/dose relationship for Sorrento/GA High Range Radiation Monitoring Systems is shown in TR-112582 [19] as:

Cable Current (nA)	Rad Monitor Indicated Dose Rate (R/hr)
10^{-2}	1
10^{-1}	10
1	10^{2}
10	10^{3}
10^{2}	10^{4}
10^{3}	10^{5}
10^{4}	10^{6}

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Testing performed for Southern California Edison confirmed the magnitude and direction of the spurious signal was a function of the temperature gradient across the cable insulation. The NRC Information Notice indicated that the duration of the spurious signal could be as long as 15 minutes, with the worst impact over in approximately 1 minute.

PINGP evaluated the potential impact of TIC on PINGP instrumentation and the only devices impacted at PINGP are the containment high range radiation monitors (CHRRM). From SAP FLOC information, cable lengths inside containment associated with these detectors are 1R-48 (1CMR-1): 80ft, 1R-49 (1CMW-1): 80ft, 2R-48 (2CMR-1): 90ft, and 2R-49 (2CMW-1): 65ft. The PINGP cable is type CBLTP 362 (Rockbestos type RSS-6-104) 2-1/C-22 COAX with a solid #22 AWG conductor. The TIC phenomenon could cause these monitors to go into high alarm during the rapid temperature increases associated with LOCAs and MSLBs inside containment. They could also alarm as failed low due to negative TIC effects during rapid cooling for the same events. CHRRMs are used to estimate post-accident fuel damage which factors into post-accident emergency classifications and in certain EOPs. PINGP's response to this issue was addressed in CHAMPS Condition Reports No. 19980371 [5] and No. 19980809 [6].

A recent review of condition reports no. 19980371 and no. 19980809 determined that no attempt was made to quantify the TIC effects from a PINGP LOCA or MSLB. Also, discussions about whether the CHRRMs meet RG 1.97 requirements with the TIC phenomenon were based on having alternate indications for fuel damage and not expecting to have fuel damage in the first 10-15 minutes of a LOCA. They did not describe how CHRRMs meet RG 1.97 requirements themselves. With recent NRC violations for other sites not adequately addressing the TIC phenomenon in their CHRRMs, a new Passport Action Request 01555551 [7], was written to provide the mechanism to assure a new review of the OE is performed.

Based on the lack of quantitative data of the TIC effect at PINGP for 1R-48/1R-49 & 2R-48/2R-49, Engineering Evaluation ECR 608000000013 was undertaken to determine the site-specific TIC effect of these CHRRMs during a worst-case LOCA and MSLB. Based on the results of this evaluation, CAP 501000001861 was written on 8/22/2017 and 1R-48/1R-49 & 2R-48/2R-49 were declared INOPERABLE by Operations.

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Since this time, the NRC has approved PINGP's LAR L-PI-17-006 [20] in NRC SER on March 6, 2018 [21] revising fission product barrier containment radiation EALs for R-48 & R-49 based on approved calculation GEN-PI-092 [9]. Specifically, the EAL containment radiation levels for a RCS barrier loss increase from 7 to 40 R/hr, a fuel cladding barrier loss increase from 200 to 5,500 R/hr, and a potential containment barrier loss increase from 800 to 23,000 R/hr. This evaluation will consider whether the changes in these EALs along with proposed changes to guidance in associated emergency and operations procedures are adequate to demonstrate that 1R-48/1R-49 & 2R-48/2R-49 can perform their intended functions when required.

2.0 Analysis – Design Basis Accidents and Quantifying Tic Effects

From PINGP accident analyses, the design basis accidents that cause the largest temperature transients in containment that would translate to the largest TIC effects are the Large-Break Loss-Of-Coolant Accident (LOCA) and the Main Steam Line Break (MSLB). From PINGP's LOCA analyses of record [8], the LOCA scenario with the highest peak temperature ramps from 120°F to around 266°F within approximately 15 seconds (See Figure A-1 below). The black line represents a temperature that bounds all analyzed scenarios with margin for EQ Program purposes and is not an actual DBA temperature profile.

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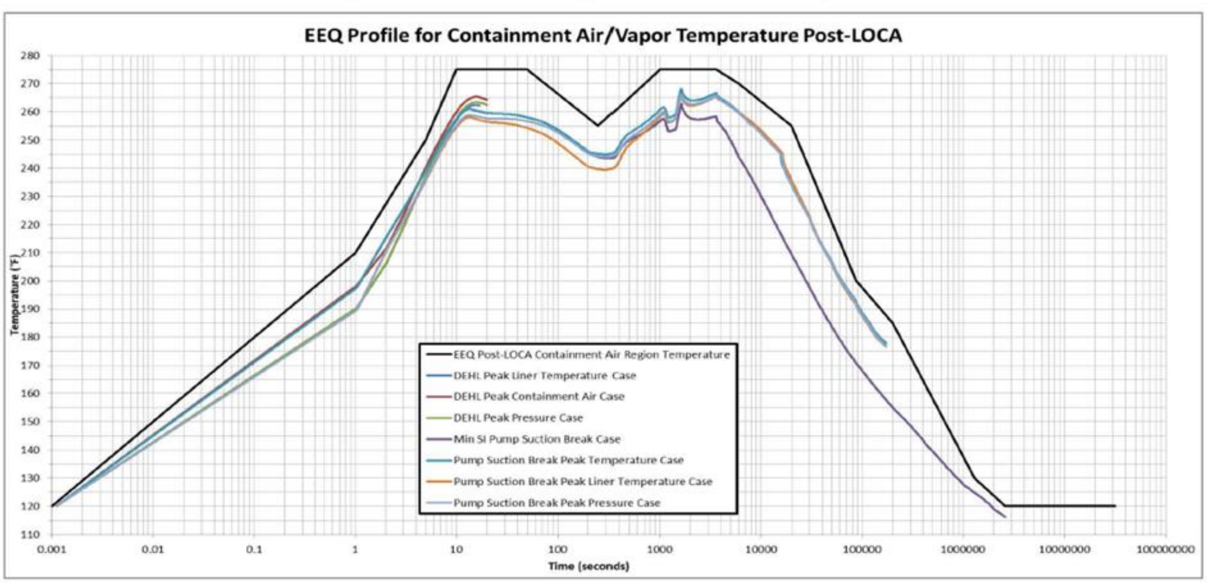

Figure A-1: Containment Air/Vapor Temperature Post-LOCA with Bounding EEQ Profile

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Only four of the seven analyzed LOCA profiles extend to the highest LOCA temperature peak that occurs at around 27 minutes into the event. Since all the LOCA temperature profiles have a very similar shape, it's not expected that TIC effects would be greatly different between the individual LOCA profiles. However, to account for the uncertainty of not obtaining the highest TIC effect from the shorter profiles, all the longer temperature profiles were run on the TIC software and the maximum difference between TIC effect results were added to the maximum TIC result from each of the longer profiles to obtain the maximum TIC effect result. Also, as previously mentioned, cable lengths have a direct effect on the TIC values and cable lengths differ between each radiation monitor. Therefore, TIC values for each radiation monitor were calculated from the TIC software results and cable lengths to show the variation in readings operators or observers would see during a LOCA.

For a MSLB, containment temperature ramps from 120°F to 311°F in 80 seconds followed by a relatively fast temperature reduction to 245 °F after 800 seconds (13 min.) [10] (See Figure B-1 below).

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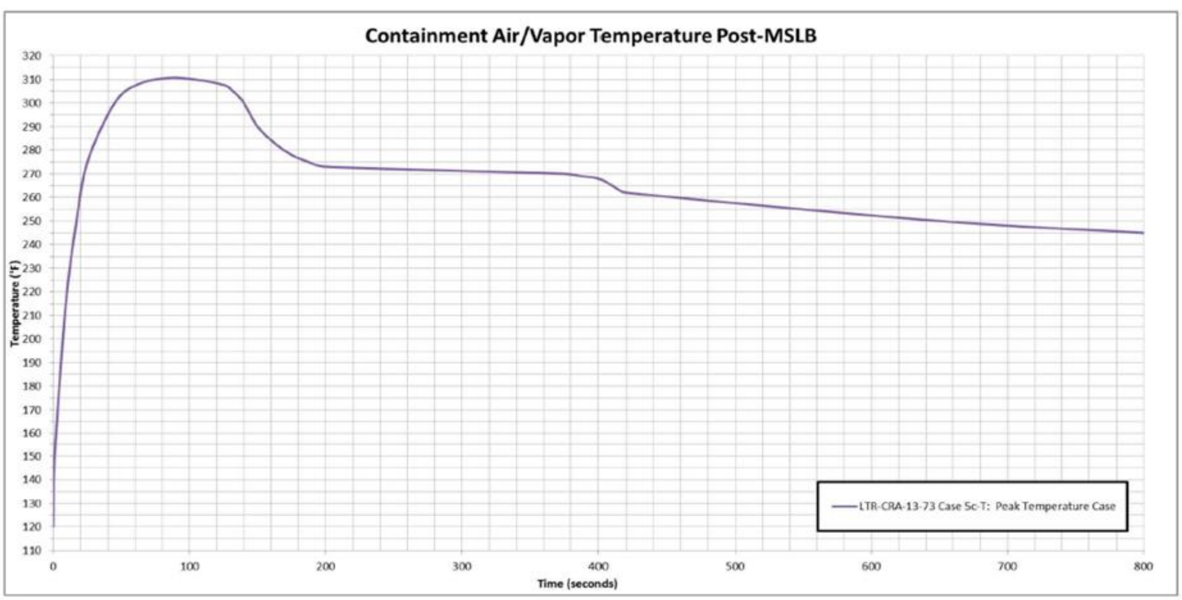

Figure B-1: Containment MSLB Peak Temperature Case

Although the MSLB temperature profile extends to only 800 seconds post-accident, from steam line break descriptions in USAR Section – 14 [2], no additional temperature spikes are expected from the event. So, the trend should continue as a slow decrease in temperature.

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3.0 RESULTS - CHRRM POST-ACCIDENT RESPONSE TO A LOCA

TIC software results for the LOCA accident profiles [35] show that during the initial temperature spike that peaks at approximately 15 seconds, the CHRRMs will produce peak TIC errors greater than +2,550 R/hr at around 9 seconds post-LOCA in the radiation monitor most effected by the TIC phenomenon (2R-48) because it has the longest cable length. As containment cools, the TIC errors decrease quickly until they become negative at around 90 seconds. The TIC errors become more negative until about 120 seconds (2 minutes) when they bottom in the -27 R/hr. in 2R-48. TIC errors then trend upward again until they reach zero at around 300 seconds (5 minutes) post-LOCA. Then, as temperatures increase again, a secondary positive TIC error peak of +22 R/hr in 2R-48 occurs around 7 minutes post-LOCA. Then, the TIC errors slowly decrease until around 18 minutes when they quickly drop and turn negative until they bottom out around -11 R/hr. in 2R-48 at approximately 20 minutes post-LOCA. Fluctuation in TIC errors continue in decreasing magnitudes as containment air temperatures change throughout the first 65 minutes of the event. At the end of the 6,000 seconds (100 minutes) post-LOCA simulation, TIC values are slightly negative ranging from -0.49 R/hr on 2R-48 to -0.35 R/hr on 2R-49. The full LOCA TIC response chart for all CHRRMs is shown in Figure D below. Note: The TIC error values used to construct Figure D were calculated from the TIC software output of the LOCA profile that produced the highest peak and lowest trough response in the 5-20 minute timeframe multiplied by 1.26 to account for the 26% maximum variation in the peak magnitudes seen in all four LOCA responses. The reasoning behind using the output with highest TIC error magnitudes from the 5-20 minute timeframe is that it would be the worst case for making an EAL classification decision during an event.

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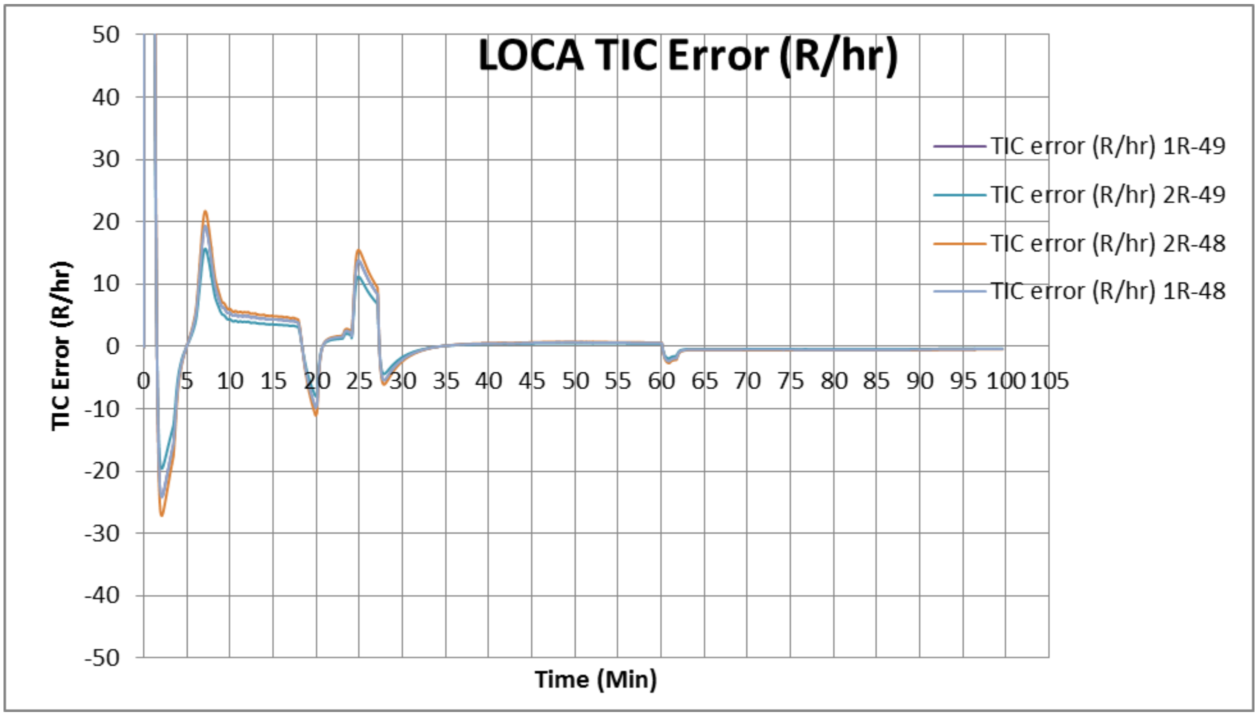

Figure D: Post-LOCA TIC Error for 1R-48, 1R-49, 2R-48, and 2R-49.

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4.0 Results - CHRRM Post-Accident Response to a MSLB

TIC software results for the MSLB accident profile [34] show that during the initial temperature spike that peaks at approximately 85 seconds, the CHRRMs will produce a peak TIC error of +1,922 R/hr at around 27 seconds post-MSLB in the radiation monitor most effected by the TIC phenomenon (2R-48). As containment cools, the TIC error decreases quickly until it becomes negative at around 117 seconds (1.95 minutes). The TIC error becomes more negative until around 158 seconds (2.6 minutes) when it bottoms out at -144 R/hr for 2R-48. TIC errors then trend less negative as temperature in containment slowly cool. However, at the end of the 800 seconds (13.3 minutes) post-MSLB simulation, TIC values are still negative and range from -7 R/hr on 2R-48 to -5 R/hr on 2R-49. See Figure E below.

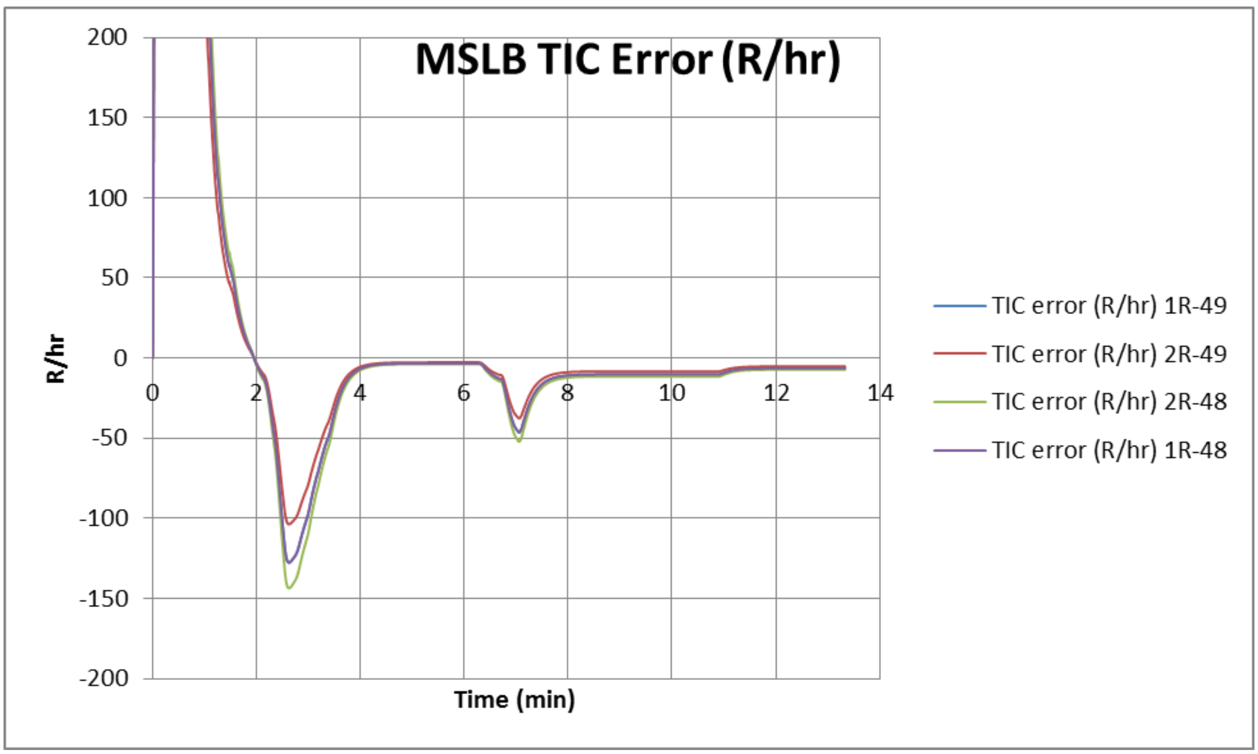

Figure E: Post-MSLB TIC Error for 1R-48, 1R-49, 2R-48, and 2R-49.

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5.0 Evaluation - TIC Impact on Design and License Bases Functions

For Emergency Planning, the Containment High-Range Radiation Monitors are used to determine the condition of fission product barriers during an event. Specifically, they are used to identify a RCS Barrier Loss, a Fuel Clad Barrier Loss, and a Containment Barrier Potential Loss from the Emergency Action Level (EAL) Matrix in the Emergency Plan [24]. The recently approved License Amendment Request [20, 21] raises the EAL for an RCS Barrier Loss from 7 R/hr to 40 R/hr on both R-48 and R-49; a Fuel Clad Barrier Loss from 200 R/hr to 5,500 R/hr, and a potential Containment Barrier Loss from 800 R/hr to 20,000 R/hr. So, this evaluation will use and will only apply when these new EALs that are implemented.

In evaluating the ability of the CHRRMs to perform their functions of detection of significant releases and emergency plan actuation, the time constraint of significance is the requirement “to assess, classify, and declare an emergency condition within 15 minutes after the availability of indications to plant operators that an EAL has been exceeded.[24]” However, in practice, the individual making the event classification needs the information no later than 13 minutes post-accident to document and verify the correct classification in the EAL tables to meet this deadline. The likelihood of an emergency classification being declared in the first 5 minutes of an event is small due to the high demands on the Main Control Room staff at the beginning of an event. So, for this evaluation, the critical timeframe when information from the CHRRMs will be needed to perform their design and license bases functions is considered to be between 5 and 13 minutes post-accident.

For the CHRRMs to function properly during a LOCA or MSLB, their readings need to provide Operations information to correctly classify the event. Radiation readings must be accurate enough that they don’t lead to under- or over-classification of the actual event. With this criteria in mind, CHHRM acceptance criteria for emergency plan actuation and release assessment is determined to be the following: Expected radiation level for fission product barrier loss + TIC error falls in the correct EAL range given for the fission product barrier loss during the 5-13 minute post-event timeframe.

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By definition, during a LOCA there is a loss of the RCS fission product barrier. Loss of the RCS fission product barrier alone would result in an alert event classification according to the EAL tables. From the current PINGP calculation of record [9], a loss of the RCS barrier without fuel failure would produce radiation levels in containment measured at 43 R/hr on the CHRRMs (not including the TIC error). The next-highest level of classification based on CHRRM readings is site area emergency due to fuel cladding failure. From the calculation of record [9], a 5% fuel cladding damage event (corresponding to a loss of fuel cladding barrier) would produce radiation levels in containment measured at 5,957 R/hr on the CHRRMs (not including the TIC error). Finally, the calculation of record [9] determined that a 20% fuel cladding damage event (corresponding to a potential loss of containment) would produce radiation levels of 23,830 R/hr. These radiation values will be used as the expected radiation levels in containment during loss of fission product barriers when evaluating the ability of CHRRMs to perform their intended functions of emergency plan actuation and release assessment during the 5-13 minute post-event timeframe.

During a MSLB in containment, the breach should be in only the non-radioactive secondary coolant system. Containment radiation should show no significant increase. Thus, based solely on CHRRM readings, acceptance criteria for the CHRRMs during an MSLB is indicated radiation levels should be below the RCS barrier loss EAL in the 5-13 minute post-accident timeframe resulting in no event being declared (The classification would be based on other plant indications). A review of MSLB data from the TIC software output [34] indicates that the range of expected TIC errors during the 5-13 minutes post-MSLB timeframe are -4 R/hr at 5 minutes to -52 R/hr at 7 minutes post-accident. A summary of expected radiation readings from the CHRRMS during MSLBs and LOCAs with the corresponding acceptance criteria is presented in Table 1 below.

One additional item of note is TR-112582 mentions the Sorrento/GA detector has a 10^{-11} amp normal output or “keep alive” signal current. If this signal is lost, the system issues a “fail” alarm to the operator. Therefore, if a negative TIC error signal of 10^{-11} amp or more occurs, the “keep alive” signal will be negated and a “fail” alarm will be generated. The following table compiles all the previously described TIC software results, information, acceptance criteria, and required time frames to summarize the ability of Operations to correctly classify the LOCA and MSLB events based on CHRRMs readings indications:

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Event	Expected Radiation Level (R/hr)	TIC Error 5-13 min. (R/hr)	CHRRMs Readings 5 -13 min. (R/hr) (Expected + TIC)	Acceptance Criteria (R/hr)	Acceptance Criteria Met?
LOCA	43	0 to 22	43 to 65	40 to 5,500	Yes.
LOCA w/Cladding Failure	5,957	0 to 22	5,957 to 5,979	5,500 to 20,000	Yes
LOCA w/potential loss of containment	23,830	0 to 22	23,830 to 23,852	> 20,000	Yes
MSLB	Normal area value <1	-4 to -52	“Failed low”	< 40	Yes, with additional guidance to address “Failed low”

Table 1: Comparison of CHRRM Indication vs. Acceptance Criteria

Table 1 shows that TIC errors during the 5 to 13 minutes period post-LOCA (without fuel cladding damage) are expected to produce readings on all four CHRRMs in the range of 43 to 65 R/hr. This range of radiation readings falls within the new EAL range of >40 and <5,500 R/hr corresponding to a RCS barrier loss. So, this meets the established acceptance criteria. For the case of a LOCA with fuel cladding failure, Table 1 shows CHRRMs would produce readings between 5,957 and 5,979 R/hr. These values also fall in the corresponding EAL acceptance criteria range for fuel cladding failure of 5,501 to 20,000 R/hr. For the case of potential loss of containment barrier, Table 1 shows CHRRMs would produce readings between 23,830 and 23,852 R/hr. These readings are above the EAL level of 20,000 R/hr that would indicate a potential loss of containment barrier. So, CHRRMs readings during all potential loss of fission product barriers due to LOCAs meet the acceptance criteria for emergency plan actuation and release assessment during the 5-13 minute post-accident timeframe.

For the case of an MSLB, Table 1 shows that TIC errors during the 5 to 13 minutes are expected to be of sufficient magnitude to negate the detector signal from normal containment radiation and the “keep alive” signal, producing a “fail low” alarm according to the detector logic diagram [29]. If appropriate guidance were provided in procedures and documents identified in Section IV that “failed low” alarms were a normal response to a cooldown event in containment for these detectors and were not failed detectors, operators and/or the emergency director could correctly conclude the fission product barriers were intact and would not classify the event based on the CHRRM readings. Therefore, CHRRMs acceptance criteria are also met for emergency plan actuation and release assessment during the 5-13 minute post-event timeframe post-MSLB.

Assessing the ability of CHRRMs to perform its remaining intended function, long-term surveillance, it can be seen in both LOCA and MSLB TIC Error charts the TIC errors decrease as stored energy from the accident has fully released and temperatures stabilize in containment. TIC errors will trend to zero with no additional releases of energy and CHRRMs will provide accurate measurement of actual radiation conditions in containment as time passes. Therefore, it is concluded that CHRRMs are able to perform their design and license bases function of long-term surveillance of radiation levels in containment during accident conditions.

6.0 Conclusions and Recommendations

Based on this evaluation with the new containment radiation EALs in place, enhanced procedural guidance on the TIC effect specified in Section X., and associated training of Operations, RP, and Emergency Response performed, the CHRRMs are able to perform their license and design bases functions. This evaluation shows that five minutes after either a LOCA or a MSLB Operations will be able to use CHRRMs measurements to detect significant radiation releases and actuate the emergency plan (Classify events) in the timeframe necessary (5-13 minutes). If personnel question alarms or data indicating high readings in the first five minutes after an event or failed low readings, there will be appropriate guidance in the alarm response or emergency procedure to assure them that this is expected after a thermal transient. Also, CHRRMs will be accurate enough after five minutes for personnel to monitor containment radiation levels in the long-term. The accuracy of CHRRMs five minutes after LOCAs and MSLBs meets the acceptance criteria for all its intended functions with enhanced procedural guidance and training in place.

One recommendation is to ensure all the conditions of this evaluation are met by creating a tracking action to verify all required procedure changes and training from Section X have been completed before returning 1R-48/1R-49 & 2R-48/2R-49 to operable.

REC/RPSS

HANDBOOK

INFORMATION USE
• Procedure should be available, but not necessarily at the work location. • Procedure may be performed from memory. • User remains responsible for procedure adherence.

REVISION 26

APPROVAL:
PCR #: 602000019055

PREFACE

This booklet contains information that may be useful to the Radiological Emergency Coordinator and/or Radiation Protection Support Supervisor for use during a radiological emergency at the Prairie Island Nuclear Generation Plant. Some of the information is factual data that may be used in formulating response decisions. Other information presents suggested solutions to pre-identified problems or situations that may exist during an emergency situation

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont.)

REPRESENTATIVE SHIELDING FACTORS FROM GAMMA CLOUD SOURCE*

Structure or Location	Shielding Factor [a]	Representative Range
Outside	1.0	--
Vehicles	1.0	--
Wood–frame House [b] (no basement)	0.9	--
Basement of wood house	0.6	0.1 to 0.7 [c]
Masonry House (no basement)	0.6	0.4 to 0.7 [c]
Basement of masonry house	0.4	0.1 to 0.5 [c]
Large office or industrial building	0.2	0.1 to 0.3 [c, d]

(a) The ratio of the dose received inside the structure to the dose that would be received outside the structure.

(b) A wood frame house with brick or stone veneer is approximately equivalent to a masonry house for shielding purposes.

(c) This range is mainly due to different wall materials and different geometries.

(d) The shielding factor depends on where the personnel are located within the building (e.g., the basement or an inside room)

SELECTED SHIELDING FACTORS FOR AIRBORNE RADIONUCLIDES

Wood house, no basement	0.9
Wood house, basement	0.6
Brick house, no basement	0.6
Brick house, basement	0.4
Large office or industrial building	0.2
Outside	1.0

* Taken from SAND 77–1725 (Unlimited Release)

REPRESENTATIVE SHIELDING FACTORS FOR SURFACE DEPOSITED RADIONUCLIDES *

Structure or Location	Shielding Factor (a)	Representative Range
1m above an infinite smooth surface	1.00	--
1m above ordinary ground	0.70	0.47 – 0.85
1m above center of 50–ft roadways, 50% decontaminated	0.55	0.4 – 0.6
Cars on 50–ft road:		
Road fully contaminated	0.5	0.4 – 0.7
Road 50% decontaminated	0.5	0.4 – 0.6
Road fully decontaminated	0.25	0.2 – 0.5
Trains	0.40	0.3 – 0.5
One and two–story wood–frame house (no basement)	0.4 (b)	0.2 – 0.5
One and two–story block and brick house (no basement)	0.2 (b)	0.04 – 0.40
House basement, one or two walls fully exposed	0.1 (b)	0.03 – 0.15
One story, less than 2 ft of basement, walls exposed	0.05 (b)	0.03 – 0.07
Two stories, less than 2 ft of basement, walls exposed	0.03 (b)	0.02 – 0.05
Three or four story structures, 5000 to 1,000 ft^2 per floor;		
First and second floors;	0.05(b)	0.01 – 0.08
Basement	0.01(b)	0.001 – 0.07
Multistory structures > 10,000 ft^2 per floor:		
Upper floors:	0.01 (b)	0.001 – 0.02
Basement	0.005 (b)	0.001 – 0.015

(a) The ratio of the dose received inside the structure to the dose that would be received outside the structure.

(b) Away from doors and windows.

* Taken from SAND 77–1725 (Unlimited Release)

GUIDELINES FOR CONTAMINATION OF HUMAN FOOD AND ANIMAL FEED*

PREVENTIVE PAG'S

1.5 rem projected dose commitment to thyroid
0.5 rem projected dose commitment to whole body, bone marrow, or any other organ

RESPONSE LEVELS FOR PREVENTIVE PAG	I–131	Cs–134	Cs–137	Sr–90	Sr–89
Initial Activity Area Deposition ($\mu Ci/m^2$)	0.13	2	3	0.5	8
Forage Concentration (μCi/kg)	0.05	0.8	1.3	0.18	3
Peak Milk Activity (μCi/liter)	0.015	0.15	0.24	0.009	0.14
Total Intake (μCi)	0.09	4	7	0.2	2.6

EMERGENCY PAG'S

15 rem projected dose commitment to thyroid
5 rem projected dose commitment to whole body, bone marrow, or any other organ

RESPONSE LEVELS FOR EMERGENCY PAG	I–131		Cs–134		Cs–137		Sr–90		Sr–89	
	Infant	Adult	Infant	Adult	Infant	Adult	Infant	Adult	Infant	Adult
Initial Activity Area Deposition ($\mu Ci/m^2$)	1.3	18	20	40	30	50	5	20	80	1600
Forage Concentration (μCi/kg)	0.5	7	8	17	13	15	1.8	8	30	700
Peak Milk Activity (μCi/liter)	0.15	2	1.5	3	2.4	4	0.09	0.04	1.4	30
Total Intake (μCi)	0.9	10	40	70	70	80	2	7	26	400

* Reference: Accidental Radioactive Contamination of Human Food and Animal Feeds; Food and Drug Administration Recommendations for State and Local Agencies, Federal Register, October 22, 1982.

R-50 OOS GRAB SAMPLES

I. ISSUE:

What are other means of monitoring high level rad releases via the Shield Building Stack when R-50 is Out Of Service?

II. SUGGESTIONS:

1.0 If the R-50 vacuum pump is out of service,

  1.1 Install and start a portable air pump.

  1.2 Spare portable air pumps are located at Access Control.

NOTE:	**If a portable air pump is used, the flow will usually bypass the R-50 monitor, but still allow sampling of the sample stream.**

  1.3 An R-50 grab sample may be taken as necessary per F3-20.2 (about once every hour).

2.0 If the monitor R-50 is out of service, refer to F3-20.2 for instructions on obtaining local samples and dose rates for releases from the Shield Building Vent Stacks.

R-51/R-52 OOS MONITORING ALTERNATIVES

I. ISSUE:

What are other means of monitoring high level rad levels in the steam lines when R-51/52 is Out Of Service?

II. SUGGESTIONS:

If R-51/52 is out of service, consider following the instructions provided in F3-20.1 concerning obtaining readings from the AM-2 remote monitor.

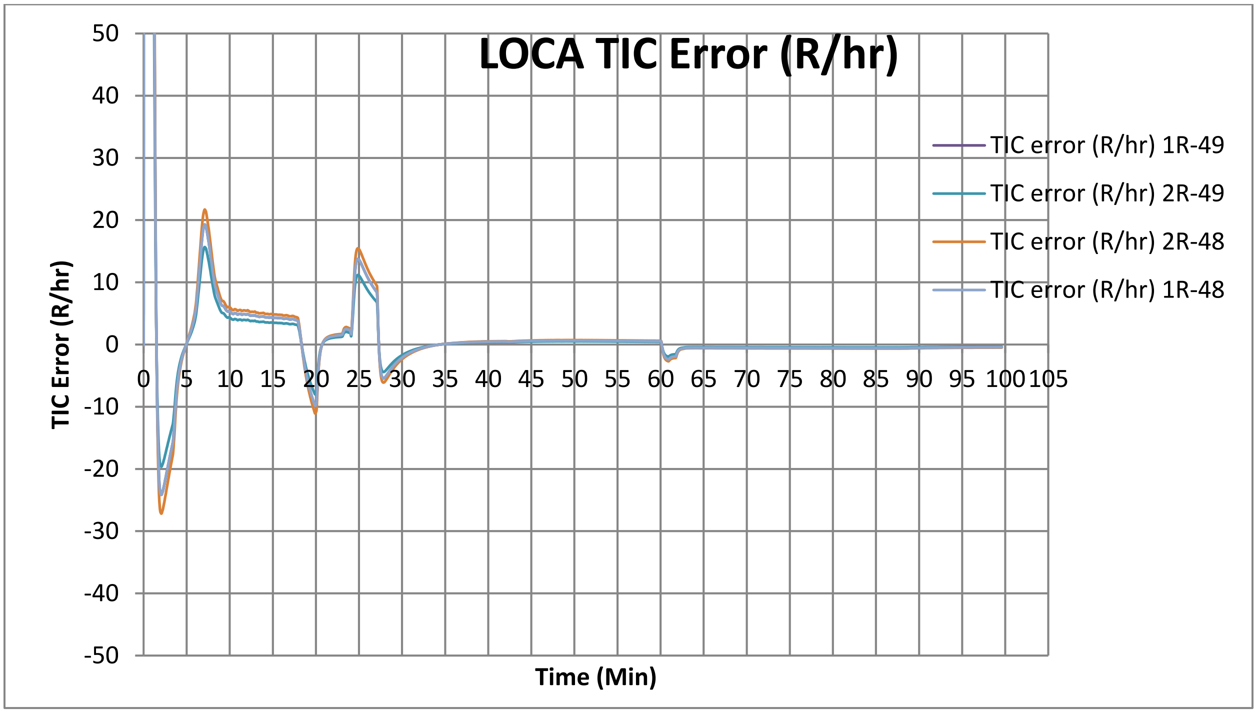
I. ISSUES:

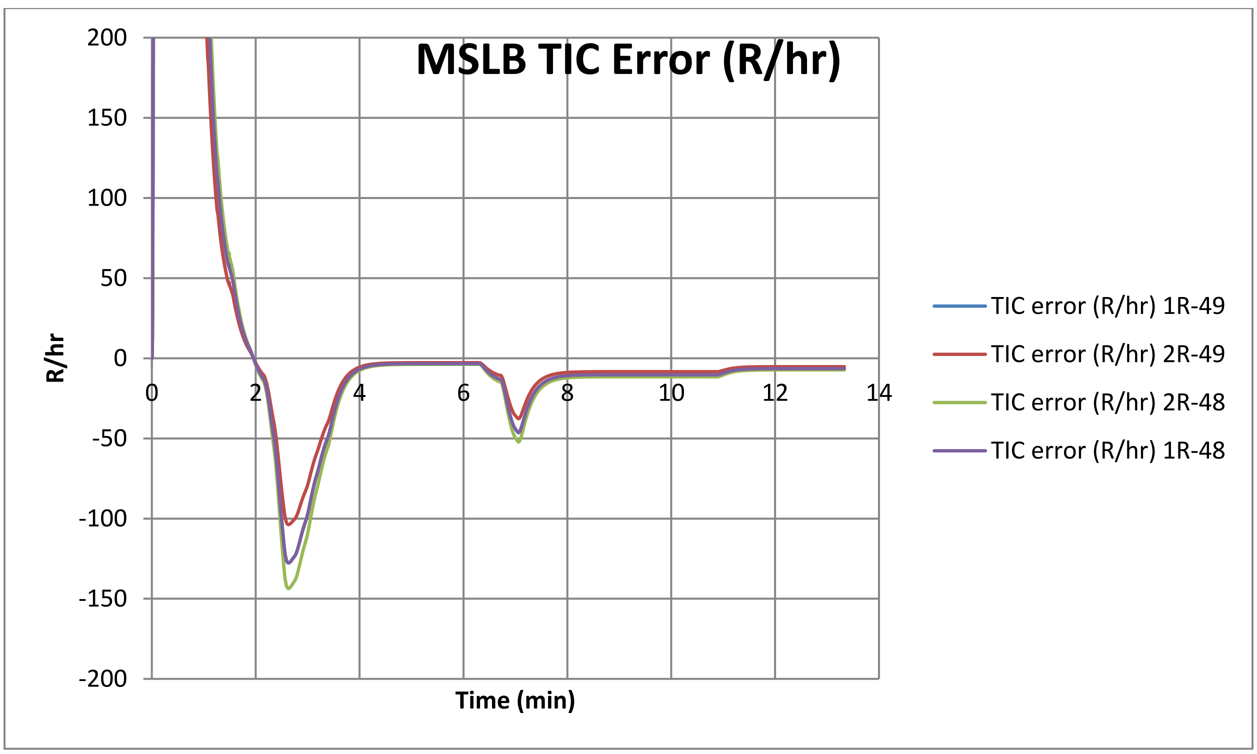
1. R-48/R-49 Thermally Induced Errors

R-48/R-49 Thermally Induced Current Errors

Erroneous readings will occur on R-48/R-49 monitors during LOCA or MSLB accidents due to thermally induced current (TIC) in the detector cabling. These detectors can still be used to classify an emergency as follows:

- Ignore the readings for the first five minutes following a LOCA or MSLB
- During a LOCA without clad damage, the TIC readings on R-48/R-49 will be higher than the Alert EAL threshold of 40 R/h and this classification will be corroborated by a separate RCS leakage EAL
- During a LOCA without clad damage, the TIC readings on R-48/R-49 will not be high enough to exceed the Site Area EAL threshold of 5,500 R/h
- R-48/R-49 readings that exceed the Site Area EAL threshold of 5,500 R/h or the General Emergency threshold of 20,000 R/h should considered valid readings and emergency classification proceed as required
- It is expected that the MSLB will cause the monitors to experience downscale failure after five minutes


LOCA TIC Error (R/hr)
TIC Error (R/hr)
Time (Min)
50
40
30
20
10
0
-10
-20
-30
-40
-50
0 5 10 15 20 25 30 35 40 45 50 55 60 65 70 75 80 85 90 95 100 105
TIC error (R/hr) 1R-49
TIC error (R/hr) 2R-49
TIC error (R/hr) 2R-48
TIC error (R/hr) 1R-48


MSLB TIC Error (R/hr)
R/hr
Time (min)
200
150
100
50
0
-50
-100
-150
-200
0 2 4 6 8 10 12 14
TIC error (R/hr) 1R-49
TIC error (R/hr) 2R-49
TIC error (R/hr) 2R-48
TIC error (R/hr) 1R-48

2. R-48/R-49 Out of Service

If R-48/49 are Out Of Service, see the alternative methods to monitor high level radiation in containment described below.

II. SUGGESTIONS:

1. Investigate the usefulness of low range R-2 or R-7 radiation monitors. Use if they are functional.

   1.1. If the loop monitors read low, then containment probably does not indicate failed fuel atmosphere.

   1.2. If the loop monitors read high, then the RCS and/or containment is highly contaminated. An air sample of containment will determine if containment is contaminated.

2. Obtain radiation monitor readings exterior to the containment shield building. Use the predicted results of F3-25, Reentry, as a guide for determining the percent of design basis accident fuel failure.

3. Obtain core exit temperatures.

Temperature	**Potential Damage**
< 750 Deg. F	No Cladding Damage
750 - 1300 Deg. F	0 - 50% Clad Damage
1300 - 1650 Deg. F	50 - 100% Clad Damage
> 1650 Deg. F	Zr-HOH Reaction, 0 - 50% Fuel Melt
> 3450 Deg. F	50 - 100% Fuel Melt
Reference: F3-17, Table 6 Characteristics of Categories of Fuel Damage.	

4. Obtain RVLIS readings.

NOTE:	**Reactor Vessel Full Range Indication displays the level from the bottom of the reactor vessel to top of reactor vessel when both RCPs are stopped.**

Reactor Vessel Full Range reading of < 50% indicates a level below top of core. (Reference: Ops Manual B4B, Rx Vessel Level Instrumentation System, Rev. 3.)

DELETED

Refer to C21.1.1 AOP , “Processing Condensate Following a Steam Generator Tube Rupture”

10CFR20 Appendix B

Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage

Attached are pages B-1 to B-88 of Appendix B to Part 20.

NOTE:	**The new liquid effluent limits are ten (10) times the Water Effluent Concentrations of Table 2, Column 2 IAW T.S. 5.5.4.b.**

Statement of Requirement:

Note: The accompanying tables (3) and footnotes/notes that comprise the rest of Appendix B can be found on the web at: <http://www.nrc.gov/NRC/CFR/TABLES/ISOTOPES/PART020-APPB/radionuclides.html>.

Appendix B to Part 20 — Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage

Introduction

For each radionuclide, Table 1 indicates the chemical form which is to be used for selecting the appropriate AL1 or DAC value. The ALIs and DACs for inhalation are given for an aerosol with an activity median aerodynamic diameter (AMAD) of 1 m and for three classes (D,W,Y) of radioactive material, which refer to their retention (approximately days, weeks or years) in the pulmonary region of the lung. This classification applies to a range of clearance half-times of less than 10 days for D, for W from 10 to 100 days, and for Y greater than 100 days. The class (D, W, or Y) given in the column headed "Class" applies only to the inhalation ALIs and DACs given in Table 1, columns 2 and 3. Table 2 provides concentration limits for airborne and liquid effluents released to the general environment. Table 3 provides concentration limits for discharges to sanitary sewer systems.

Notation

The values in Tables 1, 2, and 3 are presented in the computer "E" notation. In this notation a value of 6E-02 represents a value of 6×10^{-2} or 0.06, 6E+2 represents 6×10^{2} or 600, and 6E+0 represents 6×10^{0} or 6.

Note that the columns in Table 1 of this appendix captioned "Oral Ingestion ALI," "Inhalation ALI," and "DAC," are applicable to occupational exposure to radioactive material.

The ALIs in this appendix are the annual intakes of a given radionuclide by "Reference Man" which would result in either: (1) a CEDE of 5 rems (stochastic ALI); or (2) a committed dose equivalent of 50 rems to an organ or tissue (non-stochastic ALI). The stochastic ALIs were derived to result in a risk, due to irradiation of organs and tissues, comparable to the risk associated with deep-dose equivalent to the whole body of 5 rems. The derivation includes multiplying the committed dose equivalent to an organ or tissue by a weighting factor, w_T. This weighting factor is the proportion of the risk of stochastic effects resulting from irradiation of the organ or tissue, T, to the total risk of stochastic effects when the whole body is irradiated uniformly. The values of w_T are listed under the definition of weighting factor in 10 CFR 20.1003. The non-stochastic ALIs were derived to avoid non-stochastic effects, such as prompt damage to tissue or reduction in organ function.

A value of w_T=0.06 is applicable to each of the five organs or tissues in the "remainder" category receiving the highest dose equivalents, and the dose equivalents of all other remaining tissues may be disregarded. The following parts of the GI tract – stomach, small intestine, upper large intestine, and lower large intestine – are to be treated as four separate organs. Note that the dose equivalents for extremities (hands and forearms, feet and lower legs), skin, and lens of the eye are not considered in computing the CEDE, but are subject to limits that must be met separately.

When an AL1 is defined by the stochastic dose limit, this value alone, is given. When an AL1 is determined by the non-stochastic dose limit to an organ, the organ or tissue to which the limit applies is shown, and the AL1 for the stochastic limit is shown in parentheses. (Abbreviated organ or tissue designations are used: LLI wall = lower large intestine wall; St. wall = stomach wall; Blad wall = bladder wall; and Bone surf = bone surface.)

The use of the ALIs listed first, the more limiting of the stochastic and non-stochastic ALIs, will ensure that non-stochastic effects are avoided and that the risk of stochastic effects is limited to an acceptably low value. If, in a particular situation involving a radionuclide for which the non-stochastic AL1 is limiting, and use of that non-stochastic ALI is considered unduly conservative, the licensee may use the stochastic AL1 to determine the CEDE. However, the licensee shall also ensure that the 50-rem dose equivalent limit for any organ or tissue is not exceeded by the sum of the external deep-dose equivalent plus the internal committed dose to that organ (not the effective dose). For the case where there is no external dose contribution, this would be demonstrated if the sum of the fractions of the nonstochastic ALIs (ALI_{ns}) that contribute to the committed dose equivalent to the organ receiving the highest dose does not exceed unity (i.e., (intake (in μCi) of each radionuclide/ALI_{ns}) <1 .O). If there is an external deep-dose equivalent contribution of H_d then this sum must be less than l-(H_d/50) instead of being <1.0.

The derived air concentration (DAC) values are derived limits intended to control chronic occupational exposures. The relationship between the DAC and the AL1 is given by: DAC = AL1 (in μCi)/(2,000 hours per working year x 60 minutes/hour x 2 x 10^4 ml per minute) = [ALI/2.4 x 10^9] ,&i/ml, where 2 x 10^4 ml is the volume of air breathed per minute at work by "Reference Man" under working conditions of "light work."

The DAC values relate to one of two modes of exposure: either external submersion or the internal committed dose equivalents resulting from inhalation of radioactive materials. Derived air concentrations based upon submersion are for immersion in a semi-infinite cloud of uniform concentration and apply to each radionuclide separately.

The AL1 and DAC values relate to exposure to the single radionuclide named, but also include contributions from the in-growth of any daughter radionuclide produced in the body by the decay of the parent. However, intakes that include both the parent and daughter radionuclides should be treated by the general method appropriate for mixtures.

The value of AL1 and DAC do not apply directly when the individual both ingests and inhales a radionuclide, when the individual is exposed to a mixture of radionuclides by either inhalation or ingestion or both, or when the individual is exposed to both internal and external radiation (see 10 CFR 20.1202). When an individual is exposed to radioactive materials which fall under several of the translocation classifications (i.e., Class D, Class W, or Class Y) of the same radionuclide, the exposure may be evaluated as if it were a mixture of different radionuclides.

It should be noted that the classification of a compound as Class D, W, or Y is based on the chemical form of the compound and does not take into account the radiological half-life of different radioisotopes. For this reason, values are given for Class D, W, and Y compounds, even for very short-lived radionuclides.

The columns in Table 2 of this appendix captioned "Effluents," "Air," and "Water," are applicable to the assessment and control of dose to the public, particularly in the implementation of the provisions of 10 CFR 20.1302. The concentration values given in Columns 1 and 2 of Table 2 are equivalent to the radionuclide concentrations which, if inhaled or ingested continuously over the course of a year, would produce a total effective dose equivalent of 0.05 rem (50 millirem or 0.5 millisieverts).

Consideration of non-stochastic limits has not been included in deriving the air and water effluent concentration limits because non-stochastic effects are presumed not to occur at the dose levels established for individual members of the public. For radionuclides, where the non-stochastic limit was governing in deriving the occupational DAC, the stochastic AL1 was used in deriving the corresponding airborne effluent limit in Table 2. For this reason, the DAC and airborne effluent limits are not always proportional as was the case in Appendix B to 10 CFR 20.1- 20.601.

The air concentration values listed in Table 2, Cplumn 1, were derived by one of two methods. For those radionuclides for which the stochastic limit is governing, the occupational stochastic inhalation AL1 was divided by 2.4 x 10^9ml, relating the inhalation ALI to the DAC, as explained above, and then divided by a factor of 300. The factor of 300 includes the following components: a factor of 50 to relate the 5-rem annual occupational dose limit to the 0. l-rem limit for members of the public, a factor of 3 to adjust for the difference in exposure time and the inhalation rate for a worker and that for members of the public; and a factor of 2 to adjust the occupational values (derived for adults) so that they are applicable to other age groups.

For those radionuclides for which submersion (external dose) is limiting, the occupational DAC in Table 1, Column 3, was divided by 2 19. The factor of 2 19 is composed of a factor of 50, as described above, and a factor of 4.38 relating occupational exposure for 2,000 hours per year to full-time exposure (8,760 hours per year). Note that an additional factor of 2 for age considerations is not warranted in the submersion case.

The water concentrations were derived by taking the most restrictive occupational stochastic oral ingestion AL1 and dividing by 7.3×10^7. The factor of 7.3×10^7 (ml) includes the following components: the factors of 50 and 2 described above and a factor of 7.3×10^5 (ml) which is the annual water intake of "Reference Man."

Note 2 of this appendix provides groupings of radionuclides which are applicable to unknown mixtures of radionuclides. These groupings (including occupational inhalation ALIs and DACs, air and water effluent concentrations and sewerage) require demonstrating that the most limiting radionuclides in successive classes are absent. The limit for the unknown mixture is defined when the presence of one of the listed radionuclides cannot be definitely excluded either from knowledge of the radionuclide composition of the source or from actual measurements.

The monthly average concentrations for release to sanitary sewers are applicable to the provisions in 10 CFR 20.2003. The concentration values were derived by taking the most restrictive occupational stochastic oral ingestion AL1 and dividing by 7.3×10^6(ml). The factor of 7.3×10^6 (ml) is composed of a factor of 7.3×10^5 (ml), the annual water intake by "Reference Man," and a factor of 10, such that the concentrations, if the sewage released by the licensee were the only source of water ingested by a reference man during a year, would result in a CEDE of 0.5 rem.

List of Elements

Name	Atomic	
	Symbol	No.
Actinium	Ac	89
Aluminum	Al	13
Americium	Am	95
Antimony	Sb	51
Argon	Ar	18
Arsenic	As	33
Astatine	At	85
Barium	Ba	56
Berkelium	Bk	97
Beryllium	Be	4
Bismuth	Bi	83
Bromine	Br	35

Name	Atomic	
	Symbol	No.
Cadmium	Cd	48
Calcium	Ca	20
Califomium	Cf	98
Carbon	C	6
Cerium	Ce	58
Cesium	c s	55
Chlorine	Cl	17
Chromium	Cr	24
Cobalt	c o	27
Copper	c u	29
Curium	Cm	96
Dysprosium	Dy	66
Einsteinium	Es	99
Erbium	Er	68
Europium	Eu	63
Fermium	Fm	100
Fluorine	F	9
Francium	Fr	87
Gadolinium	Gd	64
Gallium	Ga	31
Germanium	Ge	32
Gold	Au	79
Hafnium	Hf	72
Holmium	Ho	67
Hydrogen	H	1

Name	Atomic	
	Symbol	No.
Indium	In	49
Iodine	I	53
Iridium	Ir	77
Iron	Fe	26
Krypton	Kr	36
Lanthanum	La	57
Lead	Pb	82
Lutetium	Lu	71
Magnesium	Mg	12
Manganese	Mn	25
Mendelevium	Md	101
Mercury	Hg	80
Molybdenum	MO	42
Neodymium	Nd	60
Neptunium	Np	93
Nickel	Ni	28
Niobium	Nb	41
Osmium	Os	76
Palladium	Pd	46
Phosphorus	P	15
Platinum	Pt	78
Plutonium	Pu	94
Polonium	PO	84
Potassium	K	19
Praseodymium	Pr	59

Name	Atomic	
	Symbol	No.
Promethium	Pm	61
Protactinium	Pa	91
Radium	Ra	88
Radon	Rn	86
Rhenium	Re	75
Rhodium	Rh	45
Rubidium	Rb	37
Ruthenium	Ru	44
Samarium	Sm	62
Scandium	SC	21
Selenium	Se	34
Silicon	Si	14
Silver	Ag	47
Sodium	Na	11
Strontium	Sr	38
Sulfur	S	16
Tantalum	Ta	73
Technetium	Tc	43
Tellurium	Te	52
Terbium	Tb	65
Thallium	Tl	81
Thorium	Th	90
Thulium	Tm	69
Tin	Sn	50
Titanium	Ti	22

Name	Atomic	
	Symbol	No.
Tungsten	W	74
Uranium	U	92
Vanadium	V	23
Xenon	Xe	54
Ytterbium	Yb	70
Yttrium	Y	39
Zinc	Zn	30
Zirconium	Zr	40

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
	Hydrogen-3	Water, DAC includes skin absorption	8E+4	8E+4	2E-5	1E-7	1E-3	1E-2
		Gas (HT or T_2) Submersion [1]: Use above values as HT and T, oxidize in air and in the body to HTO						
4	Beryllium-7	W, all compounds except those given for Y	4E+4	2E+4	9E-6	3E-8	6E-4	6E-3
		Y, oxides, halides, and nitrates	-	2E+4	8E-6	3E-8	-	-
4	Beryllium-10	W, see 'Be	1E+3 LLI wall	2E+2	6E-8	2E-10	-	-
			(1E+3)	-			2E-5	2E-4
		Y, see 'Be	-	1E+1	6E-9	2E-11	-	
6	Carbon-11[2]	Monoxide	-	1E+6	5E-4	2E-6		
		Dioxide	-	6E+5	3E-4	9E-7	-	
		Compounds	4E+5	4E+5	2E-4	6E-7	6E-3	6E-2
6	Carbon-14	Monoxide		2E+6	7E-4	2E-6	-	-
		Dioxide	-	2E+5	9E-5	3E-7	-	
		Compounds	2E+3	2E+3	I E-6	3E-9	3E-5	3E-4
9	Fluorine-18[2]	D, fluorides of H, Li. Na. K. Rb, Cs, and Fr	5E+4 St wall	7E+4	3E-5	I E-7	I-	-
			(5E+4)	-	-	-	7E-4	7E-3
		W, fluorides of Be. Mg, Ca, Sr. Ba. Ra, Al. Ga. In. Tl, As, Sb, Bi. Fe, Ru. Os. Co, Ni. Pd. Pt. Cu. Ag, Au. Zn. Cd, Hg, Sc. Y. Ti. Zr. V, Nb, Ta, Mn, Tc, and Re	-	9E+4	4E-5	1E-7	-	-
		Y, lanthanum fluoride		8E+4	3E-5	1E-7		
11	Sodium-22	D, all compounds	4E+2	6E+2	3E-7	9E-10	6E-6	6E-5
11	Sodium-24	D, all compounds	4E+3	5E+3	2E-6	7E-9	SE-5	5E-4

Atomic No.	Radionuclide	Class	Table 1 Occupational Values — Col. 1 Oral Ingestion ALI (μCi)	Table 1 — Col. 2 Inhalation ALI (μCi)	Table 1 — Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations — Col. 1 Air (μCi/ml)	Table 2 — Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers — Monthly Average Concentration (μCi/ml)
12	Magnesium-28	D, all compounds except those given for W	7E+2	2E+3	7E-7	2E-9	9E-6	9E-5
		W, oxides, hydroxides, carbides, halides, and nitrates		1E+3	5E-7	2E-9		
13	Aluminum-26	D, all compounds except those given for W	4E+2	6E+1	3E-8	9E-11	6E-6	6E-5
		W, oxides, hydroxides, carbides, halides, and nitrates	-	9E+1	4E-8	1E-10	-	-
14	Silicon-31	D, all compounds except those given for W and Y	9E+3	3E+4	1E-5	4E-8	1E-4	1E-3
		W, oxides, hydroxides, carbides, and nitrates		3E+4	1E-5	5E-8	-	
		Y, aluminosilicate glass		3E+4	1E-5	4E-8		
14	Silicon-32	D, see ^{31}Si	2E+3 LLI wall	2E+2	1E-7	3E-10	-	
			(3E+3)	-			4E-5	4E-4
		W, see ^{31}Si		1E+2	5E-8	2E-10	-	
		Y, see ^{31}Si		5E+0	2E-9	2E-12	-	
15	Phosphorus-32	D, all compounds except phosphates given for W	6E+2	9E+2	4E-7	1E-9	9E-6	9E-5
		W, phosphates of Zn^{2+}, S^{3+}, Mg^{2+}, Fe^{3+}, Bi^{3+}, and lanthanides		4E+2	2E-7	5E-10	-	
15	Phosphorous-33	D, see ^{32}P	6E+3	8E+3	4E-6	1E-8	8E-5	8E-4
		W, see ^{32}P		3E+3	1E-6	4E-9		

			Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion	Inhalation				Monthly Average Concentration
Atomic No.	Radionuclide	Class	ALI (μCi)	ALI (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	(μCi/ml)
16	Sulfur-35	Vapor		1E+4	6E-6	2E-8		
		D, sulfides and sulfates except those given for W	1E+4 LLI wall	2E+4	7E-6	2E-8	-	
			(8E+3)			-	1E-4	1E-3
		W, elemental sulfur, sulfides of Sr, Ba, Ge, Sn, Pb, As, Sb, Bi, Cu, Ag, Au, Zn, Cd, Hg, W, and Mo. Sulfates of Ca, Sr, Ba, Ra, As, Sb, and Bi	6E+3	2E+3	9E-7	3E-9		
17	Chlorine-36	D, chlorides of H, Li, Na, K, Rb, Cs, and Fr	2E+3	2E+3	1E-6	3E-9	2E-5	2E-4
		W, chlorides of lanthanides. Be, Mg, Ca. Sr, Ba, Ra, Al, Ga, In. Tl, Ge, Sn, Pb, As, Sb, Bi, Fe, Ru, Os, Co, Rh, Ir, Ni, Pd, Pt, Cu, Ag, Au, Zn, Cd, Hg, Sc, Y, Ti, Zr, Hf, V, Nb, Ta, Cr, Mo, W, Mn, Tc, and Re	-	2E+2	1E-7	3E-10		
17	Chlorine-38[2]	D, see ^{36}Cl	2E+4 St. wall	4E+4	2E-5	6E-8		
			(3E+4)	-	-	-	3E-4	3E-3
		W, see ^{36}Cl	-	5E+4	2E-5	6E-8		
17	Chlorine-39[2]	D, see ^{36}Cl	2E+4 St. wall	5E+4	2E-5	7E-8		
			(4E+4)	-	-	-	5E-4	5E-3
		W, see ^{36}Cl	-	6E+4	2E-5	8E-8		
18	Argon-37	Submersion[1]	-	-	1E+0	6E-3		
18	Argon-39	Submersion[1]	-	-	2E-4	8E-7		
18	Argon-41	Submersion[1]	-	-	3E-6	1E-8		
19	Potassium-40	D, all compounds	3E+2	4E+2	2E-7	6E-10	4E-6	4E-5
19	Potassium-42	D, all compounds	5E+3	5E+3	2E-6	7E-9	6E-5	6E-4
19	Potassium-43	D, all compounds	6E+3	9E+3	4E-6	1E-8	9E-5	9E-4

APPENDIX B

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion AL1 (μCi)	Table 1 Col. 2 Inhalation AL1 (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
19	Potassium-44(2)	D, all compounds	2E+4 St. wall	7E+4	3E-5	9E-8	-	
			(4E+4)	-			5E-4	5E-3
19	Potassium-45(2)	D, all compounds	3E+4 St. wall	1E+5	5E-5	2E-7		
			(5E+4)	-			7E-4	7E-3
20	Calcium-41	W, all compounds	3E+3 Bone Surf	4E+3 Bone Surf	2E-6			
			(4E+3)	(4E+3)	-	5E-9	6E-5	6E-4
20	Calcium-45	W. all compounds	2E+3	8E+2	4E-7	1E-9	2E-5	2E-4
20	Calcium-47	W. all compounds	8E+2	9E+2	4E-7	1E-9	1E-5	1E-4
21	Scandium-43	Y, all compounds	7E+3	2E+4	9E-6	3E-8	1E-4	1E-3
21	Scandium-44m	Y, all compounds	5E+2	7E+2	3E-7	1E-9	7E-6	7E-5
21	Scandium-44	Y, all compounds	4E+3	1E+4	5E-6	2E-8	5E-5	5E-4
21	Scandium-46	Y, all compounds	9E+2	2E+2	1E-7	3E-10	1E-5	1E-4
21	Scandium-47	Y, all compounds	2E+3 LLI wall	3E+3	1E-6	4E-9	-	-
			(3E+3)	-	-	-	4E-5	4E-4
21	Scandium-48	Y, all compounds	8E+2	1E+3	6E-7	2E-9	1E-5	1E-4
21	Scandium-49(2)	Y, all compounds	2E+4	5E+4	2E-5	8E-8	3E-4	3E-3
22	Titanium-44	D, all compounds except those given for W and Y	3E+2	1E+1	5E-9	2E-11	4E-6	4E-5
		W, oxides, hydroxides, carbides, halides, and nitrates		3E+1	1E-8	4E-11	-	
		Y, $SrTiO_3$		6E+0	2E-9	8E-12	-	
22	Titanium-45	D, see ^{44}Ti	9E+3	3E+4	1E-5	3E-8	1E-4	1E-3
		W, see ^{44}Ti		4E+4	1E-5	5E-8	-	
		Y, see ^{44}Ti		3E+4	1E-5	4E-8	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
23	Vanadium-47(2)	D, all compounds except those given for W	3E+4 St. wall	8E+4	3E-5	1E-7	-	
			(3E+4)	-			4E-4	4E-3
		W, oxides, hydroxides, carbides, and halides		1E+5	4E-5	1E-7	-	
23	Vanadium-48	D, see ^{47}V	6E+2	1E+3	5E-7	2E-9	9E-6	9E-5
		W, see ^{47}V		6E+2	3E-7	9E-10	-	
23	Vanadium-49	D, see ^{47}V	7E+4 LLI wall	3E+4 Bone Surf	1E-5			
			(9E+4)	(3E+4)	-	5E-8	1E-3	1E-2
		W, see ^{47}V		2E+4	8E-6	2E-8	-	
24	Chromium-48	D, all compounds except those given for W and Y	6E+3	1E+4	5E-6	2E-8	8E-5	8E-4
		W, halides and nitrates		7E+3	3E-6	1E-8		
		Y, oxides and hydroxides		7E+3	3E-6	IE-8		
24	Chromium-49(2)	D, see ^{48}Cr	3E+4	8E+4	4E-5	1E-7	4E-4	4E-3
		W, see ^{48}Cr		IE+5	4E-5	1E-7	-	
		Y, see ^{48}Cr		9E+4	4E-5	1E-7		
24	Chromium-51	D, see ^{48}Cr	4E+4	5E+4	2E-5	6E-8	5E-4	SE-3
		W, see ^{48}Cr		2E+4	IE-5	3E-8		
		Y, see ^{48}Cr		2E+4	8E-6	3E-8		
25	Manganese-51(2)	D, all compounds except those given for W	2E+4	5E+4	2E-5	7E-8	3E-4	3E-3
		W, oxides, hydroxides, halides, and nitrates		6E+4	3E-5	8E-8	-	
25	Manganese-52m(2)	D, see ^{51}Mn	3E+4 St. wall	9E+4	4E-5	1E-7		
			(4E+4)	-			5E-4	SE-3
		W, see ^{51}Mn		1E+5	4E-5	1E-7		
25	Manganese-52	D, see ^{51}Mn	7E+2	1E+3	5E-7	2E-9	1E-5	1E-4
		W, see ^{51}Mn		9E+2	4E-7	IE-9		

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
25	Manganese-53	D, see ^{51}Mn	5E+4	1E+4 Bone Surf	5E-6	-	7E-4	7E-3
			-	(2E+4)	-	3E-8	-	-
		W, see ^{51}Mn	-	1E+4	5E-6	2E-8	-	-
25	Manganese-54	D, see ^{51}Mn	2E+3	9E+2	4E-7	1E-9	3E-5	3E-4
		W, see ^{51}Mn	-	8E+2	3E-7	1E-9	-	-
25	Manganese-56	D, see ^{51}Mn	5E+3	2E+4	6E-6	2E-8	7E-5	7E-4
		W, see ^{51}Mn	-	2E+4	9E-6	3E-8	-	-
26	Iron-52	D, all compounds except those given for W	9E+2	3E+3	1E-6	4E-9	1E-5	1E-4
		W, oxides, hydroxides, and halides	-	2E+3	1E-6	3E-9	-	-
26	Iron-55	D, see ^{52}Fe	9E+3	2E+3	8E-7	3E-9	1E-4	1E-3
		W, see ^{52}Fe	-	4E+3	2E-6	6E-9	-	-
26	Iron-59	D, see ^{52}Fe	8E+2	3E+2	1E-7	5E-10	1E-5	1E-4
		W, see ^{52}Fe	-	5E+2	2E-7	7E-10	-	-
26	Iron-60	D, see ^{52}Fe	3E+1	6E+0	3E-9	9E-12	4E-7	4E-6
		W, see ^{52}Fe	-	2E+1	8E-9	3E-11	-	-
27	Cobalt-55	W, all compounds except those given for Y	1E+3	3E+3	1E-6	4E-9	2E-5	2E-4
		Y, oxides, hydroxides, halides, and nitrates	-	3E+3	1E-6	4E-9	-	-
27	Cobalt-56	W, see ^{55}Co	5E+2	3E+2	1E-7	4E-10	6E-6	6E-5
		Y, see ^{55}Co	4E+2	2E+2	8E-8	3E-10	-	-
27	Cobalt-57	W, see ^{55}Co	8E+3	3E+3	1E-6	4E-9	6E-5	6E-4
		Y, see ^{55}Co	4E+3	7E+2	3E-7	9E-10	-	-
27	Cobalt-58m	W, see ^{55}Co	6E+4	9E+4	4E-5	1E-7	8E-4	8E-3
		Y, see ^{55}Co	-	6E+4	3E-5	9E-8	-	-
27	Cobalt-58	W, see ^{55}Co	2E+3	1E+3	5E-7	2E-9	2E-5	2E-4
		Y, see ^{55}Co	1E+3	7E+2	3E-7	1E-9	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (&i/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
27	Cobalt-60m(2)	W, see ^{55}Co	1E+6 St. wall	4E+6	2E-3	6E-6	-	
			(1E+6)	-			2E-2	2E-1
		Y, see ^{55}Co		3E+6	1E-3	4E-6		
27	Cobalt-60	W, see ^{55}Co	5E+2	2E+2	7E-8	2E-10	3E-6	3E-5
		Y, see ^{55}Co	2E+2	3E+1	1E-8	SE-11	-	
27	Cobalt-61(2)	W, see ^{55}Co	2E+4	6E+4	3E-5	9E-8	3E-4	3E-3
		Y, see ^{55}Co	2E+4	6E+4	2E-5	8E-8	-	
27	Cobalt-62m(2)	W, see ^{55}Co	4E+4 St. wall	2E+5	7E-5	2E-7	-	
			(5E+4)	-			7E-4	7E-3
		Y, see ^{55}Co		2E+5	6E-5	2E-7		
28	Nickel-56	D, all compounds except those given for W	1E+3	2E+3	8E-7	3E-9	2E-5	2E-4
		W, oxides, hydroxides, and carbides	-	1E+3	5E-7	2E-9		
		vapor		1E+3	SE-7	2E-9		
28	Nickel-57	D, see ^{56}Ni	2E+3	5E+3	2E-6	7E-9	2E-5	2E-4
		W, see ^{55}Ni		3E+3	1E-6	4E-9	-	
		Vapor		6E+3	3E-6	9E-9	-	
28	Nickel-59	D. see ^{56}Ni	2E+4	4E+3	2E-6	5E-9	3E-4	3E-3
		W, see ^{56}Ni		7E+3	3E-6	1E-8		
		vapor		2E+3	8E-7	3E-9		
28	Nickel-63	D, see ^{56}Ni	9E+3	2E+3	7E-7	2E-9	1E-4	1E-3
		W, see ^{56}Ni		3E+3	1E-6	4E-9	-	
		Vapor		8E+2	3E-7	1E-9		
28	Nickel-65	D, see ^{56}Ni	8E+3	2E+4	IE-5	3E-8	1 E-4	1E-3
		W, see ^{56}Ni		3E+4	1E-5	4E-8		
		Vapor		2E+4	7E-6	2E-8		
28	Nickel-66	D, see ^{56}Ni	4E+2 LLI wall	2E+3	7E-7	2E-9	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
			(5E+2)	-	-	-	6E-6	6E-5
		W, see ^{56}Ni	-	6E+2	3E-7	9E-10	-	-
		Vapor	-	3E+3	1E-6	4E-9	-	-
29	Copper-60[2]	D, all compounds except those given for W and Y	3E+4 St. wall	9E+4	4E-5	1E-7	-	-
			(3E+4)	-	-	-	4E-4	4E-3
		W, sulfides, halides, and nitrates	-	1E+5	5E-5	2E-7	-	-
		Y,.oxides and hydroxides	-	1E+5	4E-5	1E-7	-	-
29	Copper-61	D, see ^{60}Cu	1E+4	3E+4	1E-5	4E-8	2E-4	2E-3
		W, see ^{60}Cu	-	4E+4	2E-5	6E-8	-	-
		Y, see ^{60}Cu	-	4E+4	1E-5	5E-8	-	-
29	Copper-64	D, see ^{60}Cu	1E+4	3E+4	1E-5	4E-8	2E-4	2E-3
		W, see ^{60}Cu	-	2E+4	1E-5	3E-8	-	-
		Y, see ^{60}Cu	-	2E+4	9E-6	3E-8	-	-
29	Copper-67	D, see ^{60}Cu	5E+3	8E+3	3E-6	1E-8	6E-5	6E-4
		W, see ^{60}Cu	-	5E+3	2E-6	7E-9	-	-
		Y, see ^{60}Cu	-	5E+3	2E-6	6E-9	-	-
30	Zinc-62	Y, all compounds	1E+3	3E+3	1E-6	4E-9	2E-5	2E-4
30	Zinc-63[2]	Y, all compounds	2E+4 St. wall	7E+4	3E-5	9E-8	-	-
			(3E+4)	-	-	-	3E-4	3E-3
30	Zinc-65	Y, all compounds	4E+2	3E+2	1E-7	4E-10	5E-6	5E-5
30	Zinc-69m	Y, all compounds	4E+3	7E+3	3E-6	1E-8	6E-5	6E-4
30	Zinc-69”’	Y, all compounds	6E+4	1E+5	6E-5	2E-7	8E-4	8E-3
30	Zinc-7 1 m	Y, all compounds	6E+3	2E+4	7E-6	2E-8	8E-5	8E-4
30	Zinc-72	Y, call compounds	1E+3	1E+3	5E-7	2E-9	IE-5	1E-4
31	Gallium-65[2]	D, all compounds except those given for W	5E+4 St. wall	2E+5	7E-5	2E-7		
			(6E+4)	-			9E-4	9E-3

			Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion	Inhalation				
Atomic No.	Radionuclide	Class	ALI (μCi)	ALI (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	Monthly Average Concentration (μCi/ml)
		W, oxides, hydroxides, carbides, halides, and nitrates		2E+5	8E-5	3E-7	-	
31	Gallium-66	D, see ^{65}Ga	1E+3	4E+3	1E-6	5E-9	1E-5	1E-4
		W, see ^{65}Ga		3E+3	1E-6	4E-9		
31	Gallium-67	D, see ^{65}Ga	7E+3	1E+4	6E-6	2E-8	1E-4	1E-3
		W, see ^{65}Ga		1E+4	4E-6	1E-8		
31	Gallium-68”’	D, see ^{65}Ga	2E+4	4E+4	2E-5	6E-8	2E-4	2E-3
		W, see ^{65}Ga		5E+4	2E-5	7E-8	-	
31	Gallium-70(2)	D, see ^{65}Ga	5E+4 St. wall	2E+5	7E-5	2E-7		
			(7E+4)	-			1E-3	1E-2
		W, see ^{65}Ga		2E+5	8E-5	3E-7	-	
31	Gallium-72	D, see ^{65}Ga	1E+3	4E+3	1E-6	5E-9	2E-5	2E-4
		W, see ^{65}Ga		3E+3	1E-6	4E-9	-	
31	Gallium-73	D, see ^{65}Ga	5E+3	2E+4	6E-6	2E-8	7E-5	7E-4
		W, see ^{65}Ga		2E+4	6E-6	2E-8	-	
32	Germanium-66	D. all compounds except those given for W	2E+4	3E+4	1E-5	4E-8	3E-4	3E-3
		W, oxides, sulfides, and halides		2E+4	8E-6	3E-8	-	
32	Germanium-67(2)	D, see “Ge	3E+4 St. wall	9E+4	4E-5	1E-7		
			(4E+4)	-			6E-4	6E-3
		W, see ^{66}Ge		1E+5	4E-5	1E-7	-	
32	Germanium-68	D, see “Ge	5E+3	4E+3	2E-6	5E-9	6E-5	6E-4
		W, see ^{66}Ge		1E+2	4E-8	1E-10	-	
32	Germanium-69	D, see ^{66}Ge	1E+4	2E+4	6E-6	2E-8	2E-4	2E-3
		W, see ^{66}Ge		8E+3	3E-6	1E-8		
32	Germanium-7 1	D, see ^{66}Ge	5E+5	4E+5	2E-4	6E-7	7E-3	7E-2
		W, see “Ge		4E+4	2E-5	6E-8		

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
32	Germanium-75[2]	D, see ^{66}Ge	4E+4 St. wall	8E+4	3E-5	1E-7		
			(7E+4)	-	-	-	9E-4	9E-3
		W, see ^{66}Ge	-	8E+4	4E-5	1E-7		
32	Germanium-77	D, see ^{66}Ge	9E+3	1E+4	4E-6	1E-8	1E-4	1E-3
		W, see ^{66}Ge	-	6E+3	2E-6	8E-9		
32	Germanium-78[2]	D, see ^{66}Ge	2E+4 St. wall	2E+4	9E-6	3E-8		
			(2E+4)	-	-	-	3E-4	3E-3
		W, see ^{66}Ge	-	2E+4	9E-6	3E-8		
33	Arsenic-69[2]	W, all compounds	3E+4 St. wall	1E+5	5E-5	2E-7		
			(4E+4)	-	-	-	6E-4	6E-3
33	Arsenic-70[2]	W, all compounds	1E+4	5E+4	2E-5	7E-8	2E-4	2E-3
33	Arsenic-71	W, all compounds	4E+3	5E+3	2E-6	6E-9	5E-5	5E-4
33	Arsenic-72	W, all compounds	9E+2	1E+3	6E-7	2E-9	1E-5	1E-4
33	Arsenic-73	W, all compounds	8E+3	2E+3	7E-7	2E-9	1E-4	1E-3
33	Arsenic-74	W, all compounds	1E+3	8E+2	3E-7	1E-9	2E-5	2E-4
33	Arsenic-76	W, all compounds	1E+3	1E+3	6E-7	2E-9	1E-5	1E-4
33	Arsenic-77	W, all compounds	4E+3 LLI wall	5E+3	2E-6	7E-9		
			(5E+3)	-			6E-5	6E-4
33	Arsenic-78[2]	W, all compounds	8E+3	2E+4	9E-6	3E-8	1E-4	1E-3

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
34	Selenium-70(2)	D, all compounds except those given for W	2E+4	4E+4	2E-5	5E-8	1E-4	1E-3
		W, oxides, hydroxides, carbides, and elemental Se	1E+4	4E+4	2E-5	6E-8		
34	Selenium-73m(2)	D, see ^{70}Se	6E+4	2E+5	6E-5	2E-7	4E-4	4E-3
		W, se ^{70}Se	3E+4	1E+5	6E-5	2E-7		
34	Selenium-73	D, see ^{70}Se	3E+3	1E+4	5E-6	2E-8	4E-5	4E-4
		W, see ^{70}Se	-	2E+4	7E-6	2E-8		
34	Selenium-75	D. see ^{70}Se	5E+2	7E+2	3E-7	1E-9	7E-6	7E-5
		W, see ^{70}Se	-	6E+2	3E-7	8E-10		
34	Selenium-79	D, see ^{70}Se	6E+2	8E+2	3E-7	1E-9	8E-6	8E-5
		W, see ^{70}Se	-	6E+2	2E-7	8E-10		
34	Selenium-81m(2)	D, see ^{70}Se	4E+4	7E+4	3E-5	9E-8	3E-4	3E-3
		W, see ^{70}Se	2E+4	7E+4	3E-5	1E-7		
34	Selenium-81(2)	D. see ^{70}Se	6E+4 St. wall	2E+5	9E-5	3E-7		
			(8E+4)	-	-		1E-3	1E-2
		W, see ^{70}Se	-	2E+5	1E-4	3E-7		
34	Selenium-83(2)	D, see ^{70}Se	4E+4	1E+5	5E-5	2E-7	4E-4	4E-3
		W. see ^{70}Se	3E+4	1E+5	5E-5	2E-7		
35	Bromine-74m(2)	D, bromides of H, Li, Na, K, Rb, Cs, and Fr	1E+4 St. wall	4E+4	2E-5	5E-8		
			(2E+4)	-			3E-4	3E-3
		W, bromides of lanthanides, Be, Mg, Ca, Sr, Ba, Ra, Al. Ga, In, Tl, Ge, Sn, Pb, As, Sb, Bi, Fe, Ru, Os, Co, Rh, Ir, Ni, Pd, Pt, Cu, Ag, Au, Zn, Cd, Hg, Sc, Y, Ti, Zr, Hf, V, Nb, Ta, Mn, Tc, and Re		4E+4	2E-5	6E-8		
35	Bromine-74(2)	D. see ^{74m}Br	2E+4 St. Wall	7E+4	3E-5	1E-7		

Atomic No.	Radionuclide	Class	Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion	Inhalation				Monthly Average
			ALI (μCi)	ALI (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	Concentration (μCi/ml)
			(4E+4)	-	-	-	5E-4	5E-3
		W, see ^{74m}Br	-	8E+4	4E-5	1E-7	-	-
35	Bromine-75[2]	D, see ^{74m}Br	3E+4 St. wall	5E+4	2E-5	7E-8	-	-
			(4E+4)	-	-	-	5E-4	5E-3
		W, see ^{74m}Br	-	5E+4	2E-5	7E-8	-	-
35	Bromine-76	D, see ^{74m}Br	4E+3	5E+3	2E-6	7E-9	5E-5	5E-4
		W, see ^{74m}Br	-	4E+3	2E-6	6E-9	-	-
3s	Bromine-77	D, see ^{74m}Br	2E+4	2E+4	1E-5	3E-8	2E-4	2E-3
		W, see ^{74m}Br	-	2E+4	8E-6	3E-8	-	-
35	Bromine-80m	D, see ^{74m}Br	2E+4	2E+4	7E-6	2E-8	3E-4	3E-3
		W, see ^{74m}Br	-	1E+4	6E-6	2E-8	-	-
35	Bromine-80[2]	D, see ^{74m}Br	5E+4 St. wall	2E+5	8E-5	3E-7	-	-
			(9E+4)	-	-	-	1E-3	1E-2
		W, see ^{74m}Br	-	2E+5	9E-5	3E-7	-	-
35	Bromine-82	D, see ^{74m}Br	3E+3	4E+3	2E-6	6E-9	4E-5	4E-4
		W, see ^{74m}Br	-	4E+3	2E-6	5E-9	-	-
3s	Bromine-83	D, see ^{74m}Br	5E+4 St. wall	6E+4	3E-5	9E-8	-	-
			(7E+4)	-	-	-	9E-4	9E-3
		W, see ^{74m}Br	-	6E+4	3E-5	9E-8	-	-
35	Bromine-84[2]	D, see ^{74m}Br	2E+4 St. wall	6E+4	2E-5	8E-8	-	-
			(3E+4)	-	-	-	4E-4	4E-3
		W, see ^{74m}Br	-	6E+4	3E-5	9E-8	-	-
36	Krypton-74[2]	Submersion[1]	-	-	3E-6	1E-8	-	-
36	Krypton-76	Submersion[1]	-	-	9E-6	4E-8	-	-
36	Krypton-77[2]	Submersion[1]	-	-	4E-6	2E-8	-	-
36	Krypton-79	Submersion[1]	-	-	2E-5	7E-8	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values, Col. 1, Oral Ingestion ALI (μCi)	Table 1 Col. 2, Inhalation ALI (μCi)	Table 1 Col. 3, Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations, Col. 1, Air (μCi/ml)	Table 2 Col. 2, Water (μCi/ml)	Table 3 Releases to Sewers, Monthly Average Concentration (μCi/ml)
36	Krypton-8 1	Submersion(1)	-	-	7E-4	3E-6	-	-
36	Krypton-83m(2)	Submersion(1)	-	-	1E-2	5E-5	-	-
36	Krypton-85m	Submersion(1)	-	-	2E-5	1E-7	-	-
36	Krypton-85	Submersion(1)	-	-	1E-4	7E-7	-	-
36	Krypton-87(2)	Submersion(1)	-	-	5E-6	2E-8	-	-
36	Krypton-88	Submersion(1)	-	-	2E-6	9E-9	-	-
37	Rubidium-79(2)	D, all compounds	4E+4 St. wall	1E+5	5E-5	2E-7	-	-
			(6E+4)	-	-	-	8E-4	8E-3
37	Rubidium-81 m(2)	D, all compounds	2E+5 St. wall	3E+5	1E-4	5E-7	-	-
			(3E+5)	-	-	-	4E-3	4E-2
37	Rubidium-81	D, all compounds	4E+4	5E+4	2E-5	7E-8	5E-4	5E-3
37	Rubidium-82m	D, all compounds	1E+4	2E+4	7E-6	2E-8	2E-4	2E-3
37	Rubidium-83	D, all compounds	6E+2	1E+3	4E-7	1E-9	9E-6	9E-5
37	Rubidium-84	D, all compounds	5E+2	8E+2	3E-7	1E-9	7E-6	7E-5
37	Rubidium-86	D, all compounds	5E+2	8E+2	3E-7	1E-9	7E-6	7E-5
37	Rubidium-87	D, all compounds	1E+3	2E+3	6E-7	2E-9	1E-5	1E-4
37	Rubidium-88(2)	D, all compounds	2E+4 St. wall	6E+4	3E-5	9E-8	-	-
			(3E+4)	-	-	-	4E-4	4E-3
37	Rubidium-89(2)	D, all compounds	4E+4 St. wall	1E+5	6E-5	2E-7	-	-
			(6E+4)	-	-	-	9E-4	9E-3
38	Strontium-80(2)	D, all soluble compounds except $SrTiO_3$	4E+3	1E+4	5E-6	2E-8	6E-5	6E-4
		Y, all insoluble compounds and $SrTiO_3$	-	1E+4	5E-6	2E-8	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
38	Strontium-81[2]	D, see ^{80}Sr	3E+4	8E+4	3E-5	1E-7	3E-4	3E-3
		Y, see ^{80}Sr	2E+4	8E+4	3E-5	1E-7	-	-
38	Strontium-82	D, see ^{80}Sr	3E+2 LLI wall	4E+2	2E-7	6E-10	-	-
			(2E+2)	-	-	-	3E-6	3E-5
		Y, see ^{80}Sr	2E+2	9E+1	4E-8	1E-10	-	-
38	Strontium-83	D, see ^{80}Sr	3E+3	7E+3	3E-6	1E-8	3E-5	3E-4
		Y, see “Sr	2E+3	4E+3	1E-6	5E-9	-	-
38	Strontium-85m[2]	D, see ^{80}Sr	2E+5	6E+5	3E-4	9E-7	3E-3	3E-2
		Y, see ^{80}Sr	-	8E+5	4E-4	1E-6	-	-
38	Strontium-85	D, see ^{80}Sr	3E+3	3E+3	1E-6	4E-9	4E-5	4E-4
		Y, see ^{80}Sr	-	2E+3	6E-7	2E-9	-	-
38	Strontium-87m	D, see ^{80}Sr	5E+4	1E+5	5E-5	2E-7	6E-4	6E-3
		Y, see ^{80}Sr	4E+4	2E+5	6E-5	2E-7	-	-
38	Strontium-89	D, see ^{80}Sr	6E+2 LLI wall	8E+2	4E-7	1E-9	-	-
			(6E+2)	-	-	-	8E-6	8E-5
		Y, see ^{80}Sr	5E+2	1E+2	6E-8	2E-10	-	-
38	Strontium-90	D. see ^{80}Sr	3E+1 Bone Surf	2E+1 Bone Surf	8E-9	-	-	-
			(4E+1)	(2E+1)	-	3E-11	5E-7	5E-6
		Y, see ^{80}Sr	-	4E+0	2E-9	6E-12	-	-
38	Strontium-91	D, see ^{80}Sr	2E+3	6E+3	2E-6	8E-9	2E-5	2E-4
		Y, see ^{80}Sr	-	4E+3	1E-6	5E-9	-	-
38	Strontium-92	D, see ^{80}Sr	3E+3	9E+3	4E-6	1E-8	4E-5	4E-4
		Y, see ^{80}Sr	-	7E+3	3E-6	9E-9	-	-
39	Yttrium-86m[2]	W, all compounds except those given for Y	2E+4	6E+4	2E-5	8E-8	3E-4	3E-3
		Y, oxides and hydroxides	-	5E+4	2E-5	8E-8	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
39	Yttrium-86	W, see ^{86m}Y	1E+3	3E+3	1E-6	5E-9	2E-5	2E-4
		Y, see ^{86m}Y		3E+3	1E-6	5E-9	-	-
39	Yttrium-87	W, see ^{86m}Y	2E+3	3E+3	1E-6	5E-9	3E-5	3E-4
		Y, see ^{86m}Y	-	3E+3	1E-6	5E-9	-	-
39	Yttrium-88	W, see ^{86m}Y	1E+3	3E+2	1E-7	3E-10	1E-5	1E-4
		Y, see ^{86m}Y		2E+2	1E-7	3E-10	-	-
39	Yttrium-90m	W, see ^{86m}Y	8E+3	1E+4	5E-6	2E-8	1E-4	1E-3
		Y, see ^{86m}Y	-	1E+4	5E-6	2E-8	-	-
39	Yttrium-90	W, see ^{86m}Y	4E+2 LLI wall	7E+2	3E-7	9E-10	-	-
			(5E+2)	-	-	-	7E-6	7E-5
		Y, see ^{86m}Y	-	6E+2	3E-7	9E-10	-	-
39	Yttrium-91m[2]	W, see ^{86m}Y	1E+5	2E+5	1E-4	3E-7	2E-3	2E-2
		Y, see ^{86m}Y	-	2E+5	7E-5	2E-7	-	-
39	Yttrium-91	W, see ^{86m}Y	5E+2 LLI wall	2E+2	7E-8	2E-10	-	-
			(6E+2)	-	-	-	8E-6	8E-5
		Y, see ^{86m}Y	-	1E+2	5E-8	2E-10		
39	Yttrium-92	W, see ^{86m}Y	3E+3	9E+3	4E-6	1E-8	4E-5	4E-4
		Y, see ^{86m}Y	-	8E+3	3E-6	1E-8		
39	Yttrium-93	W, see ^{86m}Y	1E+3	3E+3	1E-6	4E-9	2E-5	2E-4
		Y, see ^{86m}Y	-	2E+3	1E-6	3E-9		
39	Yttrium-94[2]	W, see ^{86m}Y	2E+4 St. wall	8E+4	3E-5	1E-7	-	-
			(3E+4)	-	-	-	4E-4	4E-3
		Y, see ^{86m}Y	-	8E+4	3E-5	1E-7		
39	Yttrium-95[2]	W, see ^{86m}Y	4E+4 St. wall	2E+5	6E-5	2E-7	-	-
			(5E+4)	-	-	-	7E-4	7E-3
		Y, see ^{86m}Y	-	1E+5	6E-5	2E-7	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
40	Zirconium-86	D, all compounds except those given for W and Y	1E+3	4E+3	2E-6	6E-9	2E-5	2E-4
		W, oxides, hydroxides, halides, and nitrates		3E+3	1E-6	4E-9	-	
		Y, carbide		2E+3	1E-6	3E-9		
40	Zirconium-88	D, see ^{86}Zr	4E+3	2E+2	9E-8	3E-10	5E-5	5E-4
		W, see ^{86}Zr		5E+2	2E-7	7E-10	-	
		Y, see ^{86}Zr		3E+2	1E-7	4E-10	-	
40	Zirconium-89	D, see ^{86}Zr	2E+3	4E+3	1E-6	5E-9	2E-5	2E-4
		W, see ^{86}Zr		2E+3	1E-6	3E-9	-	
		Y, see ^{86}Zr		2E+3	1E-6	3E-9		
40	Zirconium-93	D, see ^{86}Zr	1E+3 Bone Surf	6E+0 Bone Surf	3E-9	-		
			(3E+3)	(2E+1)	-	2E-11	4E-5	4E-4
		W, see ^{86}Zr	-	2E+1 Bone Surf	1E-8	-	-	
			-	(6E+1)	-	9E-11	-	-
		Y, see ^{86}Zr		6E+1 Bone Surf	2E-8			
				(7E+1)	-	9E-11	-	
40	Zirconium-95	D, see ^{86}Zr	1E+3	1E+2 Bone Surf	5E-8	-	2E-5	2E-4
				(3E+2)	-	4E-10	-	
		W, see ^{86}Zr		4E+2	2E-7	5E-10	-	
		Y, see ^{86}Zr		3E+2	1E-7	4E-10	-	
40	Zirconium-97	D, see ^{86}Zr	6E+2	2E+3	8E-7	3E-9	9E-6	9E-5
		W, see ^{86}Zr		1E+3	6E-7	2E-9		
		Y, see ^{86}Zr		1E+3	5E-7	2E-9		
41	Niobium-88"'	W, all compounds except those given for Y	5E+4 St. wall	2E+5	9E-5	3E-7	-	
			(7E+4)	-	-	-	1E-3	1E-2

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
		Y, oxides and hydroxides		2E+5	9E-5	3E-7	-	
41	Niobium-89m² (66min)	W, see ^{88}Nb	1E+4	4E+4	2E-5	6E-8	lE-4	IE-3
		Y, see ^{88}Nb		4E+4	2E-5	5E-8		
41	Niobium-89 (122min)	W, see ^{88}Nb	5E+3	2E+4	XE-6	3E-8	7E-5	7E-4
		Y, see ^{88}Nb		2E+4	6E-6	2E-8		
41	Niobium-90	W, see ^{88}Nb	1E+3	3E+3	1E-6	4E-9	1E-5	IE-4
		Y, see ^{88}Nb		2E+3	1E-6	3E-9	-	
41	Niobium-93m	W, see ^{88}Nb	9E+3 LLI wall	2E+3	8E-7	3E-9		
			(1E+4)	-			2E-4	2E-3
		Y, see ^{88}Nb		2E+2	7E-8	2E-10	-	
41	Niobium-94	W, see ^{88}Nb	9E+2	2E+2	8E-8	3E-10	IE-5	IE-4
		Y, see ^{88}Nb		2E+1	6E-9	2E-11	-	
41	Niobium-95m	W, see ^{88}Nb	2E+3 LLI wall	3E+3	1 E-6	4E-9		
			(2E+3)	-			3E-5	3E-4
		Y, see ^{88}Nb		2E+3	9E-7	3E-9	-	
41	Niobium-95	W, see ^{88}Nb	2E+3	1E+3	5E-7	2E-9	3E-5	3E-4
		Y, see ^{88}Nb		1E+3	5E-7	2E-9	-	
41	Niobium-96	W, see ^{88}Nb	lE+3	3E+3	1 E-6	4E-9	2E-5	2E-4
		Y, see ^{88}Nb		2E+3	I E-6	3E-9	-	
41	Niobium-97(2)	W, see ^{88}Nb	2E+4	8E+4	3E-5	IE-7	3E-4	3E-3
		Y, see ^{88}Nb		7E+4	3E-5	IE-7	-	
41	Niobium-98'''	W, see ^{88}Nb	lE+4	5E+4	2E-5	8E-8	2E-4	2E-3
		Y, see ^{88}Nb		5E+4	2E-5	7E-8	-	
42	Molybdenum-90	D, all compounds except those given for Y	4E+3	7E+3	3E-6	IE-8	3E-5	3E-4
		Y, oxides, hydroxides, and MoS_2	2E+3	5E+3	2E-6	6E-9	-	
42	Molybdenum-93m	D, see ^{90}Mo	9E+3	2E+4	7E-6	2E-8	6E-5	6E-4

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
		Y, see ^{90}Mo	4E+3	1E+4	6E-6	2E-8	-	-
42	Molybdenum-93	D, see ^{90}Mo	4E+3	5E+3	2E-6	8E-9	5E-5	5E-4
		Y, see ^{90}Mo	2E+4	2E+2	8E-8	2E-10	-	-
42	Molybdenum-99	D, see ^{90}Mo	2E+3 LLI wall	3E+3	1E-6	4E-9	-	-
			(1E+3)	-	-	-	2E-5	2E-4
		Y, see ^{90}Mo	1E+3	1E+3	6E-7	2E-9	-	-
42	Molybdenum-101[2]	D, see ^{90}Mo	4E+4 St. wall	1E+5	6E-5	2E-7	-	-
			(5E+4)	-	-	-	7E-4	7E-3
		Y, see ^{90}Mo	-	1E+5	6E-5	2E-7	-	-
43	Technetium-93m[2]	D, all compounds except those given for W	7E+4	2E+5	6E-5	2E-7	1E-3	1E-2
		W, oxides, hydroxides, halides, and nitrates	-	3E+5	1E-4	4E-7	-	-
43	Technetium-93	D, see ^{93m}Tc	3E+4	7E+4	3E-5	1E-7	4E-4	4E-3
		W, see ^{93m}Tc	-	1E+5	4E-5	1E-7	-	-
43	Technetium-94m[2]	D, see ^{93m}Tc	2E+4	4E+4	2E-5	6E-8	3E-4	3E-3
		W, see ^{93m}Tc	-	6E+4	2E-5	8E-8	-	-
43	Technetium-94	D, see ^{93m}Tc	9E+3	2E+4	8E-6	3E-8	1E-4	1E-3
		W, see ^{93m}Tc		2E+4	1E-5	3E-8	-	
43	Technetium-95m	D, see ^{93m}Tc	4E+3	5E+3	2E-6	8E-9	5E-5	5E-4
		W, see ^{93m}Tc		2E+3	8E-7	3E-9	-	
43	Technetium-95	D, see ^{93m}Tc	1E+4	2E+4	9E-6	3E-8	1E-4	IE-3
		W, see ^{93m}Tc		2E+4	8E-6	3E-8	-	
43	Technetium-96m[2]	D, see ^{93m}Tc	2E+5	3E+5	1E-4	4E-7	2E-3	2E-2
		W, see ^{93m}Tc		2E+5	1E-4	3E-7	-	
43	Technetium-96	D, see ^{93m}Tc	2E+3	3E+3	IE-6	5E-9	3E-5	3E-4
		W, see ^{93m}Tc		2E+3	9E-7	3E-9	-	
43	Technetium-97m	D, see ^{93m}Tc	5E+3	7E+3 St. wall	3E-6	-	6E-5	6E-4

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
			-	(7E+3)	-	1E-8	-	-
		W, see ^{93m}Tc	-	1E+3	5E-7	2E-9	-	-
43	Technetium-97	D, see ^{93m}Tc	4E+4	5E+4	2E-5	7E-8	5E-4	5E-3
		W, see ^{93m}Tc	-	6E+3	2E-6	8E-9	-	-
43	Technetium-98	D, see ^{93m}Tc	1E+3	2E+3	7E-7	2E-9	1E-5	1E-4
		W, see ^{93m}Tc	-	3E+2	1E-7	4E-10	-	-
43	Technetium-99m	D, see ^{93m}Tc	8E+4	2E+5	6E-5	2E-7	1E-3	1E-2
		W, see ^{93m}Tc	-	2E+5	1E-4	3E-7	-	-
43	Technetium-99	D, see ^{93m}Tc	4E+3	5E+3 St. wall	2E-6	-	6E-5	6E-4
			-	(6E+3)	-	8E-9	-	-
		W, see ^{93m}Tc	-	7E+2	3E-7	9E-10	-	-
43	Technetium-101(2)	D, see ^{93m}Tc	9E+4 St. wall	3E+5	1E-4	5E-7	-	-
			(1E+5)	-	-	-	2E-3	2E-2
		W, see ^{93m}Tc	-	4E+5	2E-4	5E-7	-	-
43	Technetium-104(2)	D, see ^{93m}Tc	2E+4 St. wall	7E+4	3E-5	1E-7	-	-
			(3E+4)	-	-	-	4E-4	4E-3
		W, see ^{93m}Tc	-	9E+4	4E-5	1E-7	-	-
44	Ruthenium-94(2)	D, all compounds except those given for W and Y	2E+4	4E+4	2E-5	6E-8	2E-4	2E-3
		W, halides	-	6E+4	3E-5	9E-8	-	-
		Y, oxides and hydroxides	-	6E+4	2E-5	8E-8	-	-
44	Ruthenium-97	D, see ^{94}Ru	8E+3	2E+4	8E-6	3E-8	1E-4	1E-3
		W, see ^{94}Ru	-	1E+4	5E-6	2E-8	-	-
		Y, see ^{94}Ru	-	1E+4	5E-6	2E-8	-	-
44	Ruthenium-103	D, see ^{94}Ru	2E+3	2E+3	7E-7	2E-9	3E-5	3E-4
		W, see ^{94}Ru	-	1E+3	4E-7	1E-9	-	-
		Y, see ^{94}Ru	-	6E+2	3E-7	9E-10	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
44	Ruthenium-105	D, see ^{94}Ru	5E+3	1E+4	6E-6	2E-8	7E-5	7E-4
		W, see ^{94}Ru	-	1E+4	6E-6	2E-8	-	-
		Y, see ^{94}Ru	-	1E+4	5E-6	2E-8	-	-
44	Ruthenium- 106	D, see ^{94}Ru	2E+2 LLI wall	9E+1	4E-8	1E-10	-	-
			(2E+2)	-	-	-	3E-6	3E-5
		W, see ^{94}Ru	-	5E+1	2E-8	8E-11	-	-
		Y, see ^{94}Ru	-	1E+1	5E-9	2E-11	-	-
45	Rhodium-99m	D, all compounds except those given for W and Y	2E+4	6E+4	2E-5	8E-8	2E-4	2E-3
		W, halides	-	8E+4	3E-5	1E-7	-	-
		Y, oxides and hydroxides	-	7E+4	3E-5	9E-8	-	-
45	Rhodium-99	D, see ^{99m}Rh	2E+3	3E+3	1E-6	4E-9	3E-5	3E-4
		W, see ^{99m}Rh	-	2E+3	9E-7	3E-9	-	-
		Y, see ^{99m}Rh	-	2E+3	8E-7	3E-9	-	-
45	Rhodium-100	D, see ^{99m}Rh	2E+3	5E+3	2E-6	7E-9	2E-5	2E-4
		W, see ^{99m}Rh	-	4E+3	2E-6	6E-9	-	-
		Y, see ^{99m}Rh	-	4E+3	2E-6	5E-9	-	-
45	Rhodium-101m	D, see ^{99m}Rh	6E+3	1E+4	5E-6	2E-8	8E-5	8E-4
		W, see ^{99m}Rh	-	8E+3	4E-6	1E-8	-	-
		Y, see ^{99m}Rh	-	8E+3	3E-6	1E-8	-	-
45	Rhodium-101	D, see ^{99m}Rh	2E+3	5E+2	2E-7	7E-10	3E-5	3E-4
		W, see ^{99m}Rh	-	8E+2	3E-7	1E-9	-	-
		Y, see ^{99m}Rh	-	2E+2	6E-8	2E-10	-	-
45	Rhodium-102m	D, see ^{99m}Rh	1E+3 LLI wall	5E+2	2E-7	7E-10	-	-
			(1E+3)	-	-	-	2E-5	2E-4
		W, see ^{99m}Rh	-	4E+2	2E-7	5E-10	-	-
		Y, see ^{99m}Rh	-	1E+2	5E-8	2E-10	-	-
45	Rhodium-102	D, see ^{99m}Rh	6E+2	9E+1	4E-8	1E-10	8E-6	8E-5

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
		W, see ^{99m}Rh	-	2E+2	7E-8	2E-10	-	-
		Y, see ^{99m}Rh	-	6E+1	2E-8	8E-11	-	-
45	Rhodium-103m(2)	D, see ^{99m}Rh	4E+5	1E+6	5E-4	2E-6	6E-3	6E-2
		W, see ^{99m}Rh	-	1E+6	5E-4	2E-6	-	-
		Y, see ^{99m}Rh	-	1E+6	5E-4	2E-6	-	-
45	Rhodium-105	D, see ^{99m}Rh	4E+3 LLI wall	1E+4	5E-6	2E-8	-	-
			(4E+3)	-	-	-	5E-5	5E-4
		W, see ^{99m}Rh	-	6E+3	3E-6	9E-9	-	-
		Y, see ^{99m}Rh	-	6E+3	2E-6	8E-9	-	-
45	Rhodium-106m	D, see ^{99m}Rh	8E+3	3E+4	1E-5	4E-8	1E-4	1E-3
		W, see ^{99m}Rh	-	4E+4	2E-5	5E-8	-	-
		Y, see ^{99m}Rh	-	4E+4	1E-5	5E-8	-	-
45	Rhodium-107(2)	D, see ^{99m}Rh	7E+4 St. wall	2E+5	1E-4	3E-7	-	-
			(9E+4)	-	-	-	1E-3	1E-2
		W, see ^{99m}Rh	-	3E+5	1E-4	4E-7	-	-
		Y, see ^{99m}Rh	-	3E+5	1E-4	3E-7	-	-
46	Palladium-100	D, all compounds except those given for W and Y	1E+3	1E+3	6E-7	2E-9	2E-5	2E-4
		W, nitrates	-	1E+3	5E-7	2E-9	-	-
		Y, oxides and hydroxides	-	1E+3	6E-7	2E-9	-	-
46	Palladium-101	D, see ^{100}Pd	1E+4	3E+4	1E-5	5E-8	2E-4	2E-3
		W, see ^{100}Pd	-	3E+4	1E-5	5E-8	-	-
		Y, see ^{100}Pd		3E+4	1E-5	4E-8		
46	Palladium-103	D, see ^{100}Pd	6E+3 LLI wall	6E+3	3E-6	9E-9	-	
			(7E+3)	-			1E-4	1E-3
		W, see ^{100}Pd		4E+3	2E-6	6E-9	-	
		Y, see ^{100}Pd		4E+3	1 E-6	5E-9	-	

APPENDIX B

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
46	Palladium- 107	D, see ^{100}Pd	3E+4 LLI wall	2E+4 Kidneys	9E-6	-	-	-
			(4E+4)	(2E+4)	-	3E-8	5E-4	5E-3
		W, see ^{100}Pd	-	7E+3	3E-6	1E-8	-	-
		Y, see ^{100}Pd	-	4E+2	2E-7	6E-10	-	-
46	Palladium- 109	D, see ^{100}Pd	2E+3	6E+3	3E-6	9E-6	3E-5	3E-4
		W, see ^{100}Pd	-	5E+3	2E-6	8E-9	-	-
		Y, see ^{100}Pd	-	5E+3	2E-6	6E-9	-	-
47	Silver- 102(2)	D, all compounds except those given for W and Y	5E+4 St. wall	2E+5	8E-5	2E-7	-	-
			(6E+4)	-	-	-	9E-4	9E-3
		W, nitrates and sulfides	-	2E+5	9E-5	3E-7	-	-
		Y, oxides and hydroxides	-	2E+5	8E-5	3E-7	-	-
47	Silver- 103(2)	D, see ^{102}Ag	4E+4	1E+5	4E-5	1E-7	5E-4	5E-3
		W, see ^{102}Ag	-	1E+5	5E-5	2E-7	-	-
		Y, see ^{102}Ag	-	1E+5	5E-5	2E-7	-	-
47	Silver-104m(2)	D, see ^{102}Ag	3E+4	9E+4	4E-5	1E-7	4E-4	4E-3
		W, see ^{102}Ag	-	1E+5	5E-5	2E-7	-	-
		Y, see ^{102}Ag	-	1E+5	5E-5	2E-7	-	-
47	Silver- 104(2)	D, see ^{102}Ag	2E+4	7E+4	3E-5	1E-7	3E-4	3E-3
		W, see ^{102}Ag	-	1E+5	6E-5	2E-7	-	-
		Y, see ^{102}Ag	-	1E+5	6E-5	2E-7	-	-
47	Silver-105	D, see ^{102}Ag	3E+3	1E+3	4E-7	1E-9	4E-5	4E-4
		W, see ^{102}Ag	-	2E+3	7E-7	2E-9	-	-
		Y, see ^{102}Ag	-	2E+3	7E-7	2E-9	-	-
47	Silver- 106m	D, see ^{102}Ag	8E+2	7E+2	3E-7	1E-9	1E-5	1E-4
		W, see ^{102}Ag	-	9E+2	4E-7	1E-9	-	-
		Y, see ^{102}Ag	-	9E+2	4E-7	1E-9	-	-
47	Silver-106"	D, see ^{102}Ag	6E+4 St. wall	2E+5	8E-5	3E-7	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
			(6E+4)	-			9E-4	9E-3
		W, see ^{102}Ag		2E+5	9E-5	3E-7	-	
		Y, see ^{102}Ag		2E+5	8E-5	3E-7	-	
47	Silver-108m	D, see ^{102}Ag	6E+2	2E+2	8E-8	3E-10	9E-6	9E-5
		W, see ^{102}Ag		3E+2	1E-7	4E-10	-	
		Y, see ^{102}Ag		2E+1	1E-8	3E-11	-	
47	Silver-110m	D, see ^{102}Ag	5E+2	1E+2	5E-8	2E-10	6E-6	6E-5
		W, see ^{102}Ag		2E+2	8E-8	3E-10	-	
		Y, see ^{102}Ag		9E+1	4E-8	1E-10	-	
47	Silver-111	D, see ^{102}Ag	9E+2 LLI wall	2E+3 Liver	6E-7			
			(1E+3)	(2E+3)	-	2E-9	2E-5	2E-4
		W, see ^{102}Ag		9E+2	4E-7	1E-9	-	
		Y, see ^{102}Ag		9E+2	4E-7	1E-9		
47	Silver-112	D, see ^{102}Ag	3E+3	8E+3	3E-6	1E-8	4E-5	4E-4
		W, see ^{102}Ag		1E+4	4E-6	1E-8		
		Y, see ^{102}Ag		9E+3	4E-6	1E-8		
47	Silver-115[2]	D, see ^{102}Ag	3E+4 St. wall	9E+4	4E-5	1E-7	-	
			(3E+4)	-			4E-4	4E-3
		W, see ^{102}Ag		9E+4	4E-5	1E-7		
		Y, see ^{102}Ag		8E+4	3E-5	1E-7	-	
48	Cadmium-104[2]	D, all compounds except those given for W and Y	2E+4	7E+4	3E-5	9E-8	3E-4	3E-3
		W, sulfides, halides, and nitrates		1E+5	5E-5	2E-7	-	
		Y, oxides and hydroxides		1E+5	5E-5	2E-7		
48	Cadmium-107	D, see ^{104}Cd	2E+4	5E+4	2E-5	8E-8	3E-4	3E-3
		W, see ^{104}Cd		6E+4	2E-5	8E-8	-	
		Y, see ^{104}Cd		5E+4	2E-5	7E-8	-	

APPENDIX B

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
48	Cadmium- 109	D, see ^{104}Cd	3E+2 Kidneys	4E+1 Kidneys	1E-8	-	-	--
			(4E+2)	(5E+1)	-	7E-11	6E-6	6E-5
		W, see ^{104}Cd	-	1E+2 Kidneys	5E-8	-	-	-
			-	(1E+2)	-	2E-10	-	-
		Y, see ^{104}Cd		1E+2	5E-8	2E-10	-	
48	Cadmium-113m	D, see ^{104}Cd	2E+1 Kidneys	2E+0 Kidneys	1E-9			
			(4E+1)	(4E+0)	-	5E-12	5E-7	5E-6
		W, see ^{104}Cd		8E+0 Kidneys	4E-9	-		
			-	(1E+1)	-	2E-11	-	-
		Y, see ^{104}Cd		1E+1	5E-9	2E-11	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion AL1 (μCi)	Table 1 Col. 2 Inhalation AL1 (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
48	Cadmium- 113	D, see ^{104}Cd	2E+1 Kidneys	2E+0 Kidneys	9E-10	-		
			(3E+1)	(3E+0)	-	5E-12	4E-7	4E-6
		W, see ^{104}Cd		8E+0 Kidneys	3E-9			
				(1E+1)	-	2E-11	-	
		Y, see ^{104}Cd		1E+1	6E-9	2E-11	-	
48	Cadmium-l 15m	D, see ^{104}Cd	3E+2	5E+1 Kidneys	2E-8		4E-6	4E-5
				(8E+1)	-	1E-10	-	
		W, see ^{104}Cd		1E+2	5E-8	2E-10	-	
		Y, see ^{104}Cd		1E+2	6E-8	2E-10	-	
48	Cadmium- 115	D, see ^{104}Cd	9E+2 LLI wall	1E+3	6E-7	2E-9		
			(1E+3)	-			1E-5	1 E-4
		W, see ^{104}Cd		1E+3	5E-7	2E-9	-	
		Y, see ““Cd		1E+3	6E-7	2E-9		
48	Cadmium-117m	D, see ^{104}Cd	5E+3	1E+4	5E-6	2E-8	6E-5	6E-4
		'W, see ^{104}Cd		2E+4	7E-6	2E-8		
		Y, see ^{104}Cd		1E+4	6E-6	2E-8		
48	Cadmium- 117	D. see ““Cd	5E+3	1E+4	5E-6	2E-8	6E-5	6E-4
		W, see ^{104}Cd		2E+4	7E-6	2E-8		
		Y. see ““Cd		1E+4	6E-6	2E-8		
49	Indium-109	D, all compounds except those given for W	2E+4	4E+4	2E-5	6E-8	3E-4	3E-3
		W, oxides, hydroxides, halides, and nitrates		6E+4	3E-5	9E-8	-	
49	Indium-110[2] (69.1min)	D. see ^{109}In	2E+4	4E+4	2E-5	6E-8	2E-4	2E-3
		W, see ““In		6E+4	2E-5	8E-8	-	
49	Indium- I IO (4.9h)	D, see ^{109}In	5E+3	2E+4	7E-6	2E-8	7E-5	7E-4
		W, see ^{109}In		2E+4	8E-6	3E-8	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values, Col. 1, Oral Ingestion, ALI (μCi)	Table 1 Occupational Values, Col. 2, Inhalation, ALI (μCi)	Table 1 Occupational Values, Col. 3, Inhalation, DAC (μCi/ml)	Table 2 Effluent Concentrations, Col. 1, Air (μCi/ml)	Table 2 Effluent Concentrations, Col. 2, Water (μCi/ml)	Table 3 Releases to Sewers, Monthly Average Concentration (μCi/ml)
49	Indium- 111	D, see ^{109}In	4E+3	6E+3	3E-6	9E-9	6E-5	6E-4
		W, see ^{109}In		6E+3	3E-6	9E-9	-	
49	Indium-112[2]	D, see ^{109}In	2E+5	6E+5	3E-4	9E-7	2E-3	2E-2
		W, see ^{109}In	-	7E+5	3E-4	1E-6	-	-
49	Indium-113m[2]	D, see ^{109}In	5E+4	1E+5	6E-5	2E-7	7E-4	7E-3
		W, see ^{109}In		2E+5	8E-5	3E-7	-	-
49	Indium-114m	D, see ^{109}In	3E+2 LLI wall	6E+1	3E-8	9E-11	-	-
			(4E+2)	-	-	-	5E-6	5E-5
		W, see ^{109}In		1E+2	4E-8	1E-10	-	-
49	Indium-115m	D, see ^{109}In	1E+4	4E+4	2E-5	6E-8	2E-4	2E-3
		W, see ^{109}In		5E+4	2E-5	7E-8	-	
49	Indium-11.5	D, see ^{109}In	4E+1	1E+0	6E-10	2E-12	5E-7	5E-6
		W, see ^{109}In		5E+0	2E-9	8E-12	-	-
	Indium- 116m[2]	D. see ^{109}In	2E+4	8E+4	3E-5	1E-7	3E-4	3E-3
		W, see ^{109}In		1E-5	5E-5	2E-7	-	-
49	Indium- 117m[2]	D, see ^{109}In	1E+4	3E+4	1E-5	5E-8	2E-4	2E-3
		W, see ^{109}In		4E+4	2E-5	6E-8	-	-
49	Indium- 117[2]	D. see ^{109}In	6E+4	2E+5	7E-5	2E-7	8E-4	8E-3
		W, see ^{109}In		2E+5	9E-5	3E-7	-	
49	Indium-119m[2]	D. see ^{109}In	4E+4 St. wall	1E+5	5E-5	2E-7	-	
			(5E+4)	-		-	7E-4	7E-3
		W, see ^{109}In		1E+5	6E-5	2E-7	-	-
50	Tin-110	D. all compounds except those given for W	4E+3	1E+4	5E-6	2E-8	5E-5	5E-4
		W, sulfides, oxides, hydroxides, halides, nitrates, and stannic phosphate		1E+4	5E-6	2E-8		

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
SO	Tin-l 11'''	D, see ^{110}Sn	7E+4	2E+5	9E-5	3E-7	1E-3	lE-2
		W, see ^{110}Sn		3E+5	1E-4	4E-7		
50	Tin-l 13	D, see ^{110}Sn	2E+3 LLI wall	1E+3	5E-7	2E-9	-	-
			(2E+3)	-			3E-5	3E-4
		W, see ^{110}Sn	-	SE+2	2E-7	8E-10	-	
50	Tin-117m	D, see ^{110}Sn	2E+3 LLI wall	1E+3 Bone Surf	SE-7	-		
			(2E+3)	(2E+3)	-	3E-9	3E-5	3E-4
		W, see ^{110}Sn		1E+3	6E-7	2E-9	-	
SO	Tin-119m	D, see ^{110}Sn	3E+3 LLI wall	2E+3	lE-6	3E-9	-	
			(4E+3)	-			6E-5	6E-4
		W, see ^{110}Sn		lE+3	4E-7	1E-9		
SO	Tin-12lm	D, see ^{110}Sn	3E+3 LLI wall	9E+2	4E-7	1E-9		
			(4E+3)	-			SE-S	SE-4
		W, see ^{110}Sn		5E+2	2E-7	8E-10	-	
50	Tin-121	D, see ^{110}Sn	6E+3 LLI wall	2E+4	6E-6	2E-8	-	-
			(6E+3)	-			8E-5	8E-4
		W, see ^{110}Sn		lE+4	SE-6	2E-8		
50	Tin-123m[(2)]	D, see ^{110}Sn	SE+4	1E+5	SE-S	2E-7	7E-4	7E-3
		W, see ^{110}Sn		1E+5	6E-5	2E-7	-	
50	Tin- 123	D, see ^{110}Sn	5E+2 LLI wall	6E+2	3E-7	9E-10	-	-
			(6E+2)	-	-	-	9E-6	9E-5
		W, see ^{110}Sn	-	2E+2	7E-8	2E-10	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALL (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
50	Tin-125	D, see ^{110}Sn	4E+2 LLI wall	9E+2	4E-7	1E-9	-	-
			(5E+2)	-	-	-	6E-6	6E-5
		W, see ^{110}Sn	-	4E+2	1E-7	5E-10	-	-
50	Tin-126	D, see ^{110}Sn	3E+2	6E+1	2E-8	8E-11	4E-6	4E-5
		W, see ^{110}Sn	-	7E+1	3E-8	9E-11	-	-
so	Tin-127	D, see ^{110}Sn	7E+3	2E+4	8E-6	3E-8	9E-5	9E-4
		W, see ^{110}Sn	-	2E+4	8E-6	3E-8	-	-
50	Tin-128[2]	D, see ^{110}Sn	9E+3	3E+4	1E-5	4E-8	1E-4	1E-3
		W, see ^{110}Sn	-	4E+4	1E-5	5E-8	-	-
51	Antimony-115[2]	D, all compounds except those given for W	8E+4	2E+5	1E-4	3E-7	1E-3	1E-2
		W, oxides, hydroxides, halides, sulfides, sulfates, and nitrates	-	3E+5	1E-4	4E-7	-	-
51	Antimony-116m[2]	D, see ^{115}Sb	2E+4	7E+4	3E-5	1E-7	3E-4	3E-3
		W, see ^{115}Sb	-	1E+5	6E-5	2E-7	-	-
51	Antimony-116[2]	D, see ^{115}Sb	7E+4 St. wall	3E+5	1E-4	4E-7	-	-
			(9E+4)	-	-	-	1E-3	1E-2
		W, see ^{115}Sb	-	3E+5	1E-4	5E-7	-	-
51	Antimony-117	D, see ^{115}Sb	7E+4	2E+5	9E-5	3E-7	9E-4	9E-3
		W, see ^{115}Sb	-	3E+5	1E-4	4E-7	-	-
51	Antimony-118m	D, see ^{115}Sb	6E+3	2E+4	8E-6	3E-8	7E-5	7E-4
		W, see ^{115}Sb	5E+3	2E+4	9E-6	3E-8	-	-
51	Antimony-119	D, see ^{115}Sb	2E+4	5E+4	2E-5	6E-8	2E-4	2E-3
		W, see ^{115}Sb	2E+4	3E+4	1E-5	4E-8	-	-
51	Antimony-120 (16min)	D, see ^{115}Sb	1E+5 St. wall	4E+5	2E-4	6E-7	-	-
			(2E+5)	-	-	-	2E-3	2E-2
		W, see ^{115}Sb	-	5E+5	2E-4	7E-7	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values: Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations: Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers: Monthly Average Concentration (μCi/ml)
51	Antimony-120 (5.76 d)	D, see ^{115}Sb	1E+3	2E+3	9E-7	3E-9	1E-5	1E-4
		W, see ^{115}Sb	9E+2	1E+3	SE-7	2E-9	-	
51	Antimony-122	D, see ^{115}Sb	8E+2 LLI wall	2E+3	1E-6	3E-9	-	
			(8E+2)				1E-5	1E-4
		W, see ^{115}Sb	7E+2	1E+3	4E-7	2E-9		
51	Antimony-124m[2]	D, see ^{115}Sb	3E+5	8E+5	4E-4	1E-6	3E-3	3E-2
		W, see ^{115}Sb	2E+5	6E+5	2E-4	8E-7	-	
51	Antimony-124	D, see ^{115}Sb	6E+2	9E+2	4E-7	1E-9	7E-6	7E-5
		W, see ^{115}Sb	5E+2	2E+2	1E-7	3E-10	-	-
51	Antimony-125	D, see ^{115}Sb	2E+3	2E+3	1E-6	3E-9	3E-5	3E-4
		W, see ^{115}Sb	-	5E+2	2E-7	7E-10	-	-
51	Antimony-126m[2]	D, see ^{115}Sb	5E+4 St. wall	2E+5	8E-5	3E-7	-	-
			(7E+4)	-	-	-	9E-4	9E-3
		W, see ^{115}Sb	-	2E+5	8E-5	3E-7	-	-
51	Antimony-126	D, see ^{115}Sb	6E+2	1E+3	5E-7	2E-9	7E-6	7E-5
		W, see ^{115}Sb	5E+2	5E+2	2E-7	7E-10	-	-
51	Antimony-127	D, see ^{115}Sb	8E+2 LLI wall	2E+3	9E-7	3E-9	-	-
			(8E+2)	-	-	-	1E-5	1E-4
		W, see ^{115}Sb	7E+2	9E+2	4E-7	1E-9	-	-
51	Antimony-128[2] (10.4min)	D, see ^{115}Sb	8E+4 St. wall	4E+5	2E-4	5E-7	-	-
			(1E+5)	-			1E-3	1E-2
		W, see ^{115}Sb		4E+5	2E-4	6E-7	-	
51	Antimony-128 (9.01h)	D, see ^{115}Sb	1E+3	4E+3	2E-6	6E-9	2E-5	2E-4
		W, see ^{115}Sb		3E+3	1E-6	5E-9	-	
51	Antimony-129	D, see ^{115}Sb	3E+3	9E+3	4E-6	1E-8	4E-5	4E-4
		W, see ^{115}Sb		9E+3	4E-6	1E-8		

Atomic No.	CRadionuclide	a s s	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
51	Antimony-130(2)	D, see ^{115}Sb	2E+4	6E+4	3E-5	9E-8	3E-4	3E-3
		W, see ^{115}Sb		8E+4	3E-5	1E-7		
51	Antimony-131(2)	D, see ^{115}Sb	1E+4 Thyroid	2E+4 Thyroid	1E-5	-		
			(2E+4)	(4E+4)	-	6E-8	2E-4	2E-3
		W, see ^{115}Sb	-	2E+4 Thyroid	1E-5	-		
				(4E+4)	-	6E-8		
52	Tellurium-116	D, all compounds except those given for W	8E+3	2E+4	9E-6	3E-8	1E-4	1E-3
		W, oxides, hydroxides, and nitrates		3E+4	1E-5	4E-8		
52	Tellurium-121m	D, see ^{116}Te	SE+2 Bone Surf	2E+2 Bone Surf	8E-8			
			(7E+2)	(4E+2)	-	5E-10	1E-5	1E-4
		W. see ^{116}Te	-	4E+2	2E-7	6E-10		
52	Tellurium-121	D, see ^{116}Te	3E+3	4E+3	2E-6	6E-9	4E-5	4E-4
		W, see ^{116}Te	-	3E+3	1E-6	4E-9		
52	Tellurium-123m	D, see ^{116}Te	6E+2 Bone Surf	2E+2 Bone Surf	9E-8	-		
			(1E+3)	(5E+2)	-	8E-10	1E-5	1E-4
		W, see ^{116}Te		SE+2	2E-7	8E-10		
52	Tellurium-123	D, see ^{116}Te	5E+2 Bone Surf	2E+2 Bone Surf	8E-8	-		
			(1E+3)	(5E+2)	-	7E-10	2E-5	2E-4
		W, see ^{116}Te	-	4E+2 Bone Surf	2E-7	-		
			-	(1E+3)	-	2E-9		

Atomic No.	Radionuclide	Class	Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion	Inhalation				Monthly Average Concentration (μCi/ml)
			ALI (μCi)	ALI (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	
52	Tellurium-125m	D. see ^{116}Te	1E+3 Bone Surf	4E+2 Bone Surf	2E-7	-		
			(1E+3)	(1E+3)	-	1E-9	2E-5	2E-4
		W, see ^{116}Te		7E+2	3E-7	1E-9		
52	Tellurium-l 27m	D, see ^{116}Te	6E+2	3E+2 Bone Surf	1E-7	-	9E-6	9E-5
				(4E+2)	-	6E-10		
		W, see ^{116}Te	-	3E+2	1E-7	4E-10		
52	Tellurium-127	D, see ^{116}Te	7E+3	2E+4	9E-6	3E-8	1E-4	1E-3
		W. see ^{116}Te		2E+4	7E-6	2E-8		
52	Tellurium-129m	D, see ^{116}Te	5E+2	6E+2	3E-7	9E-10	7E-6	7E-5
		W, see ^{116}Te		2E+2	1E-7	3E-10		
52	Tellurium- 129'''	D, see ^{116}Te	3E+4	6E+4	3E-5	9E-8	4E-4	4E-3
		W, see ^{116}Te		7E+4	3E-5	1E-7		
52	Tellurium- 1 lm	D, see ^{116}Te	3E+2 Thyroid	4E+2 Thyroid	2E-7	-		
			(6E+2)	(1E+3)	-	2E-9	8E-6	8E-5
		W, see ^{116}Te		4E+2 Thyroid	2E-7	-		
				(9E+2)	-	1E-9		
52	Tellurium-13 I'''	D, see ^{116}Te	3E+3 Thyroid	5E+3 Thyroid	2E-6	-		
			(6E+3)	(1E+4)	-	2E-8	8E-5	8E-4
		W. see ^{116}Te		5E+3 Thyroid	2E-6	-		
				(1E+4)	-	2E-8		
52	Tellurium- 132	D. see ^{116}Te	2E+2 Thyroid	2E+2 Thyroid	9E-8	-		
			(7E+2)	(8E+2)	-	1E-9	9E-6	9E-5
		W, see ^{116}Te		2E+2 Thyroid	9E-8	-		
				(6E+2)	-	9E-10		

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
52	Tellurium-133m[(2)]	D, see ^{116}Te	3E+3 Thyroid	5E+3 Thyroid	2E-6	-	-	-
			(6E+3)	(1E+4)	-	2E-8	9E-5	9E-4
		W, see ^{116}Te	-	5E+3 Thyroid	2E-6	-	-	-
			-	(1E+4)	-	2E-8	-	-
52	Tellurium-133[(2)]	D, see ^{116}Te	1E+4 Thyroid	2E+4 Thyroid	9E-6	-	-	-
			(3E+4)	(6E+4)	-	8E-8	4E-4	4E-3
		W, see ^{116}Te	-	2E+4 Thyroid	9E-6	-	-	-
			-	(6E+4)	-	8E-8	-	-
52	Tellurium- 134'''	D. see ^{116}Te	2E+4 Thyroid	2E+4 Thyroid	1E-5	-	-	-
			(2E+4)	(5E+4)	-	7E-8	3E-4	3E-3
		W, see ^{116}Te	-	2E+4 Thyroid	1E-5	-	-	-
			-	(5E+4)	-	7E-8	-	-
53	Iodine-120m[(2)]	D, all compounds	1E+4 Thyroid	2E+4	9E-6	3E-8	-	-
			(1E+4)	-	-	-	2E-4	2E-3
53	Iodine- I 20[(2)]	D, all compounds	4 E+3 Thyroid	9E+3 Thyroid	4E-6	-	-	-
			(8E+3)	(1E+4)	-	2E-8	1E-4	1E-3
53	Iodine-121	D, all compounds	1E+4 Thyroid	2E+4 Thyroid	8E-6	-	-	-
			(3E+4)	(5E+4)	-	7E-8	4E-4	4E-3
53	Iodine- 123	D. all compounds	3E+3 Thyroid	6E+3 Thyroid	3E-6	-		
			(1E+4)	(2E+4)	-	2E-8	1 E-4	1E-3
I 63 d i n e - 1 2 4		D, all compounds	5E+1 Thyroid	8E+1 Thyroid	3E-8	-		
			(2E+2)	(3E+2)	-	4E-10	2E-6	2E-5

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
53	Iodine- 125	D, all compounds	4E+1 Thyroid	6E+1 Thyroid	3E-8	-		
			(1E+2)	(2E+2)	-	3E-10	2E-6	2E-5
53	Iodine- 126	D, all compounds	2E+1 Thyroid	4E+1 Thyroid	1E-8	-		
			(7E+1)	(1E+2)	-	2E-10	1E-6	1E-5
53	Iodine-128[2]	D, all compounds	4E+4 St. wall	1E+5	5E-5	2E-7		
			(6E+4)	-			8E-4	8E-3
53	Iodine- 129	D. all compounds	5E+0 Thyroid	9E+0 Thyroid	4E-9			
			(2E+1)	(3E+1)	-	4E-11	2E-7	2E-6
53	Iodine-130	D, all compounds	4E+2 Thyroid	7E+2 Thyroid	3E-7			
			(1E+3)	(2E+3)	-	3E-9	2E-5	2E-4
53	Iodine- 13 1	D, all compounds	3E+1 Thyroid	5E+1 Thyroid	2E-8			
			(9E+1)	(2E+2)	-	2E-10	1E-6	1E-5
53	Iodine-132m[2]	D, all compounds	4E+3 Thyroid	8E+3 Thyroid	4E-6			
			(1E+4)	(2E+4)	-	3E-8	1E-4	1E-3
53	Iodine-132	D. all compounds	4E+3 Thyroid	8E+3 Thyroid	3E-6	-		
			(9E+3)	(1E+4)	-	2E-8	1 E-4	1E-3
53	Iodine- 133	D, all compounds	1E+2 Thyroid	3E+2 Thyroid	1 E-7			
			(5E+2)	(9E+2)	-	1E-9	7E-6	7E-5
53	Iodine-134[2]	D, all compounds	2E+4 Thyroid	5E+4	2E-5	6E-8	-	
			(3E+4)	-			4E-4	4E-3
53	Iodine-1 35	D, all compounds	8E+2 Thyroid	2E+3 Thyroid	7E-7			
			(3E+3)	(4E+3)	-	6E-9	3E-5	3E-4

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
54	Xenon-120[2]	Submersion[1]			1E-5	4E-8	-	
54	Xenon-121[2]	Submersion[1]			2E-6	1E-8		
54	Xenon-122	Submersion[1]			7E-5	3E-7	-	
54	Xenon-123	Submersion[1]			6E-6	3E-8		
54	Xenon-125	Submersion[1]			2E-5	7E-8		
54	Xenon-127	Submersion[1]			1E-5	6E-8	-	
54	Xenon-129m	Submersion[1]	-	-	2E-4	9E-7	-	-
54	Xenon-131m	Submersion[1]	-	-	4E-4	2E-6	-	-
54	Xenon-133m	Submersion[1]	-	-	1E-4	6E-7	-	-
54	Xenon-133	Submersion[1]	-	-	1E-4	5E-7	-	-
54	Xenon-135m[2]	Submersion[1]	-	-	9E-6	4E-8	-	-
54	Xenon-135	Submersion[1]	-	-	1E-5	7E-8	-	-
54	Xenon-138[2]	Submersion[1]	-	-	4E-6	2E-8	-	-
55	Cesium-125[2]	D, all compounds	5E+4 St. wall	1E+5	6E-5	2E-7	-	-
			(9E+4)	-	-	-	1E-3	1E-2
55	Cesium-127	D, all compounds	6E+4	9E+4	4E-5	1E-7	9E-4	9E-3
55	Cesium-129	D, all compounds	2E+4	3E+4	1E-5	5E-8	3E-4	3E-3
55	Cesium-130[2]	D, all compounds	6E+4 St. wall	2E+5	8E-5	3E-7	-	-
			(1E+5)	-	-	-	1E-3	1E-2
55	Cesium-131	D, all compounds	2E+4	3E+4	1E-5	4E-8	3E-4	3E-3
55	Cesium-132	D, all compounds	3E+3	4E+3	2E-6	6E-9	4E-5	4E-4
55	Cesium-134m	D, all compounds	1E+5 St. wall	1E+5	6E-5	2E-7	-	-
			(1E+5)	-	-	-	2E-3	2E-2
55	Cesium-134	D, all compounds	7E+1	1E+2	4E-8	2E-10	9E-7	9E-6
55	Cesium-135m[2]	D, all compounds	1E+5	2E+5	8E-5	3E-7	1E-3	1E-2
55	Cesium-135	D, all compounds	7E+2	1E+3	5E-7	2E-9	1E-5	1E-4
55	Cesium-136	D, all compounds	4E+2	7E+2	3E-7	9E-10	6E-6	6E-5

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
55	Cesium-137	D, all compounds	1E+2	2E+2	6E-8	2E-10	1E-6	1E-5
55	Cesium-138[2]	D, all compounds	2E+4 St. wall	6E+4	2E-5	8E-8	-	-
			(3E+4)	-	-	-	4E-4	4E-3
56	Barium-126[2]	D, all compounds	6E+3	2E+4	6E-6	2E-8	8E-5	8E-4
56	Barium-128	D, all compounds	5E+2	2E+3	7E-7	2E-9	7E-6	7E-5
56	Barium-131m[2]	D, all compounds	4E+5 St. wall	1E+6	6E-4	2E-6	-	-
			(5E+5)	-	-	-	7E-3	7E-2
56	Barium-131	D, all compounds	3E+3	8E+3	3E-6	1E-8	4E-5	4E-4
56	Barium-133m	D, all compounds	2E+3 LLI wall	9E+3	4E-6	1E-8	-	-
			(3E+3)	-	-	-	4E-5	4E-4
56	Barium-133	D, all compounds	2E+3	7E+2	3E-7	9E-10	2E-5	2E-4
56	Barium-135m	D, all compounds	3E+3	1E+4	5E-6	2E-8	4E-5	4E-4
56	Barium-139[2]	D, all compounds	1E+4	3E+4	1E-5	4E-8	2E-4	2E-3
56	Barium-140	D, all compounds	5E+2 LLI wall	1E+3	6E-7	2E-9	-	-
			(6E+2)	-	-	-	8E-6	8E-5
56	Barium-141[2]	D, all compounds	2E+4	7E+4	3E-5	1E-7	3E-4	3E-3
56	Barium-142[2]	D, all compounds	5E+4	1E+5	6E-5	2E-7	7E-4	7E-3
57	Lanthanum-131[2]	D, all compounds except those given for W	5E+4	1E+5	5E-5	2E-7	6E-4	6E-3
		W, oxides and hydroxides	-	2E+5	7E-5	2E-7	-	-
57	Lanthanum-132	D, see ^{131}La	3E+3	1E+4	4E-6	1E-8	4E-5	4E-4
		W, see ^{131}La	-	1E+4	5E-6	2E-8	-	-
57	Lanthanum-1 35	D, see ^{131}La	4E+4	1E+5	4E-5	IE-7	5E-4	5E-3
		W. see ^{131}La		9E+4	4E-5	1E-7	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
57	Lanthanum-137	D, see ^{131}La	1E+4	6E+1 Liver	3E-8	-	2E-4	2E-3
			-	(7E+1)	-	1E-10	-	-
		W, see ^{131}La	-	3E+2 Liver	1E-7	-	-	-
			-	(3E+2)	-	4E-10	-	-
57	Lanthanum-138	D, see ^{131}La	9E+2	4E+0	1E-9	5E-12	1E-5	1E-4
		W, see ^{131}La	-	1E+1	6E-9	2E-11	-	-
57	Lanthanum-140	D, see ^{131}La	6E+2	1E+3	6E-7	2E-9	9E-6	9E-5
		W, see ^{131}La	-	1E+3	5E-7	2E-9	-	-
57	Lanthanum-141	D, see ^{131}La	4E+3	9E+3	4E-6	1E-8	5E-5	5E-4
		W, see ^{131}La	-	1E+4	5E-6	2E-8	-	-
57	Lanthanum-142$^{(2)}$	D, see ^{131}La	8E+3	2E+4	9E-6	3E-8	1E-4	1E-3
		W, see ^{131}La	-	3E+4	1E-5	5E-8	-	-
57	Lanthanum-143$^{(2)}$	D, see ^{131}La	4E+4 St. wall	1E+5	4E-5	1E-7	-	-
			(4E+4)	-	-	-	5E-4	5E-3
		W, see ^{131}La	-	9E+4	4E-5	1E-7	-	-
58	Cerium-134	W, all compounds except those given for Y	5E+2 LLI wall	7E+2	3E-7	1E-9	-	-
			(6E+2)	-	-	-	8E-6	8E-5
		Y, oxides, hydroxides, and fluorides	-	7E+2	3E-7	9E-10	-	-
58	Cerium-135	W, see ^{134}Ce	2E+3	4E+3	2E-6	5E-9	2E-5	2E-4
		Y, see ^{134}Ce	-	4E+3	1E-6	5E-9	-	-
58	Cerium-137m	W, see ^{134}Ce	2E+3 LLI wall	4E+3	2E-6	6E-9	-	
			(2E+3)	-			3E-5	3E-4
		Y, see ^{134}Ce		4E+3	2E-6	5E-9	-	
58	Cerium-137	W, see ^{134}Ce	5E+4	1E+5	6E-5	2E-7	7E-4	7E-3
		Y, see ^{134}Ce		1E+5	5E-5	2E-7	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
58	Cerium-139	W, see ^{134}Ce	5E+3	8E+2	3E-7	1E-9	7E-5	7E-4
		Y, see ^{134}Ce		7E+2	3E-7	9E-10	-	
58	Cerium- 141	W, see ^{134}Ce	2E+3 LLI wall	7E+2	3E-7	1E-9	-	
			(2E+3)	-	-	-	3E-5	3E-4
		Y, see ^{134}Ce	-	6E+2	2E-7	8E-10	-	
58	Cerium-143	W, see ^{134}Ce	1E+3 LLI wall	2E+3	8E-7	3E-9	-	
			(1E+3)	-	-	-	2E-5	2E-4
		Y, see ^{134}Ce	-	2E+3	7E-7	2E-9	-	
58	Cerium-144	W, see ^{134}Ce	2E+2 LLI wall	3E+1	1E-8	4E-11	-	
			(3E+2)	-	-	-	3E-6	3E-5
		Y, see ^{134}Ce	-	1E+1	6E-9	2E-11	-	
59	Praseodymium-136(2)	W, all compounds except those given for Y	5E+4 St. wall	2E+5	1E-4	3E-7	-	
			(7E+4)	-	-	-	1E-3	1E-2
		Y, oxides, hydroxides, carbides, and fluorides	-	2E+5	9E-5	3E-7	-	
59	Praseodymium-137(2)	W, see ^{136}Pr	4E+4	2E+5	6E-5	2E-7	5E-4	5E-3
		Y, see ^{136}Pr	-	1E+5	6E-5	2E-7	-	
59	Praseodymium-138m	W, see ^{136}Pr	1E+4	5E+4	2E-5	8E-8	1E-4	1E-3
		Y, see ^{136}Pr	-	4E+4	2E-5	6E-8	-	
59	Praseodymium-139	W, see ^{136}Pr	4E+4	1E+5	5E-5	2E-7	6E-4	6E-3
		Y, see ^{136}Pr	-	1E+5	5E-5	2E-7	-	
59	Praseodymium-142m(2)	W, see ^{136}Pr	8E+4	2E+5	7E-5	2E-7	1E-3	1E-2
		Y, see ^{136}Pr	-	1E+5	6E-5	2E-7	-	
59	Praseodymium-142	W, see ^{136}Pr	1E+3	2E+3	9E-7	3E-9	1E-5	1E-4
		Y, see ^{136}Pr	-	2E+3	8E-7	3E-9	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
59	Praseodymium-143	W, see ^{136}Pr	9E+2 LLI wall	8E+2	3E-7	1E-9	-	
			(1E+3)	-			2E-5	2E-4
		Y, see ^{136}Pr		7E+2	3E-7	9E-10	-	
59	Praseodymium-144(2)	W, see ^{136}Pr	3E+4 St. wall	1E+5	5E-5	2E-7	-	
			(4E+4)	-			6E-4	6E-3
		Y, see ^{136}Pr		1E+5	5E-5	2E-7	-	
59	Praseodymium-145	W, see ^{136}Pr	3E+3	9E+3	4E-6	1E-8	4E-5	4E-4
		Y, see ^{136}Pr		8E+3	3E-6	1E-8	-	
59	Praseodymium-147(2)	W, see ^{136}Pr	5E+4 St. wall	2E+5	8E-5	3E-7	-	
			(8E+4)	-			1E-3	1E-2
		Y, see ^{136}Pr		2E+5	8E-5	3E-7	-	
60	Neodymium-136(2)	W, all compounds except those given for Y	1E+4	6E+4	2E-5	8E-8	2E-4	2E-3
		Y, oxides, hydroxides, carbides, and fluorides		5E+4	2E-5	8E-8	-	
60	Neodymium- 138	W, see ^{136}Nd	2E+3	6E+3	3E-6	9E-9	3E-5	3E-4
		Y. see ^{136}Nd		5E+3	2E-6	7E-9		
60	Neodymium-139m	W, see ^{136}Nd	5E+3	2E+4	7E-6	2E-8	7E-5	7E-4
		Y, see ^{136}Nd		1E+4	6E-6	2E-8		
60	Neodymium-139(2)	W, see ^{136}Nd	9E+4	3E+5	1 E-4	5E-7	1E-3	1E-2
		Y, see ^{136}Nd		3E+5	1E-4	4E-7		
60	Neodymium-141	W, see ^{136}Nd	2E+5	7E+5	3E-4	1E-6	2E-3	2E-2
		Y, see ^{136}Nd		6E+5	3E-4	9E-7	-	
60	Neodymium-1 47	W, see ^{136}Nd	1E+3 LLI wall	9E+2	4E-7	1E-9	-	
			(1E+3)	-			2E-5	2E-4
		Y, see ^{136}Nd		8E+2	4E-7	1E-9		

			Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion	Inhalation				Monthly Average
Atomic No.	Radionuclide	Class	ALI (μCi)	ALI (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	Concentration (μCi/ml)
60	Neodymium-149(2)	W, see ^{136}Nd	1E+4	3E+4	1E-5	4E-8	1E-4	1E-3
		Y, see ^{136}Nd		2E+4	1E-5	3E-8	-	
60	Neodymium-151(2)	W, see ^{136}Nd	7E+4	2E+5	8E-5	3E-7	9E-4	9E-3
		Y, see ^{136}Nd		2E+5	8E-5	3E-7	-	
61	Promethium-141(2)	W, all compounds except those given for Y	5E+4 St. wall	2E+5	8E-5	3E-7	-	
			(6E+4)	-			8E-4	8E-3
		Y, oxides, hydroxides, carbides, and fluorides		2E+5	7E-5	2E-7	-	
61	Promethium- 143	W, see “‘Pm	5E+3	6E+2	2E-7	8E-10	7E-5	7E-4
		Y, see “‘Pm		7E+2	3E-7	1E-9		
61	Promethium- 144	W, see ^{141}Pm	1E+3	1E+2	5E-8	2E-10	2E-5	2E-4
		Y, see ^{141}Pm		1E+2	5E-8	2E-10	-	
61	Promethium- 145	W, see ^{141}Pm	1E+4	2E+2 Bone Surf	7E-8		1E-4	1E-3
				(2E+2)	-	3E-10	-	
		Y, see “‘Pm		2E+2	8E-8	3E-10	-	
61	Promethium- 146	W, see ““Pm	2E+3	5E+1	2E-8	7E-11	2E-5	2E-4
		Y, see “‘Pm		4E+1	2E-8	6E-11	-	
61	Promethium- 147	W, see “‘Pm	4E+3 LLI wall	1E+2 Bone Surf	5E-8	-		
			(5E+3)	(2E+2)	-	3E-10	7E-5	7E-4
		Y, see “‘Pm		1E+2	6E-8	2E-10	-	
61	Promethium-148m	W, see “‘Pm	7E+2	3E+2	1E-7	4E-10	1E-5	1E-4
		Y, see “‘Pm		3E+2	1E-7	5E-10	-	
61	Promethium-148	W, see “‘Pm	4E+2 LLI wall	5E+2	2E-7	8E-10	-	
			(5E+2)	-			7E-6	7E-5
		Y, see “‘Pm		5E+2	2E-7	7E-10	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
61	Promethium- 149	W, see ^{141}Pm	1E+3 LLI wall	2E+3	8E-7	3E-9	-	-
			(1E+3)	-	-	-	2E-5	2E-4
		Y, see ^{141}Pm	-	2E+3	8E-7	2E-9	-	-
61	Promethium- 150	W, see ^{141}Pm	5E+3	2E+4	8E-6	3E-8	7E-5	7E-4
		Y, see ^{141}Pm	-	2E+4	7E-6	2E-8	-	-
61	Promethium- 15 1	W, see ^{141}Pm	2E+3	4E+3	1E-6	5E-9	2E-5	2E-4
		Y, see ^{141}Pm	-	3E+3	1E-6	4E-9	-	-
62	Samarium-141m[2]	W, all compounds	3E+4	1E+5	4E-5	1E-7	4E-4	4E-3
62	Samarium-141[2]	W, all compounds	5E+4 St. wall	2E+5	8E-5	2E-7	-	-
			(6E+4)	-	-	-	8E-4	8E-3
62	Samarium- 142[2]	W, all compounds	8E+3	3E+4	1E-5	4E-8	1E-4	1E-3
62	Samarium- 145	W, all compounds	6E+3	5E+2	2E-7	7E-10	8E-5	8E-4
62	Samarium- 146	W, all compounds	1E+1 Bone Surf	4E-2 Bone Surf	1E-11	-	-	-
			(3E+1)	(6E-2)	-	9E-14	3E-7	3E-6
62	Samarium-1 47	W, all compounds	2E+1 Bone Surf	4E-2 Bone Surf	2E-11	-	-	-
			(3E+1)	(7E-2)	-	1E-13	4E-7	4E-6
62	Samarium-151	W, all compounds	1E+4 LLI wall	1E+2 Bone Surf	4E-8	-	-	-
			(1E+4)	(2E+2)	-	2E-10	2E-4	2E-3
62	Samarium- 153	W, all compounds	2E+3 LLI wall	3E+3	1E-6	4E-9	-	-
			(2E+3)	-			3E-5	3E-4
62	Samarium- 155[2]	W, all compounds	6E+4 St. wall	2E+5	9E-5	3E-7	-	
			(8E+4)	-			1E-3	1E-2
62	Samarium- 156	W, all compounds	5E+3	9E+3	4E-6	1E-8	7E-5	7E-4
63	Europium- 145	W, all compounds	2E+3	2E+3	8E-7	3E-9	2E-5	2E-4

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
63	Europium-146	W, all compounds	1E+3	1E+3	5E-7	2E-9	1E-5	1E-4
63	Europium-147	W, all compounds	3E+3	2E+3	7E-7	2E-9	4E-5	4E-4
63	Europium-148	W, all compounds	1E+3	4E+2	1E-7	5E-10	1E-5	1E-4
63	Europium-149	W, all compounds	1E+4	3E+3	1E-6	4E-9	2E-4	2E-3
63	Europium-150 (12.62h)	W, all compounds	3E+3	8E+3	4E-6	1E-8	4E-5	4E-4
63	Europium-150 (34.2y)	W, all compounds	8E+2	2E+1	8E-9	3E-11	1E-5	1E-4
63	Europium-152m	W, all compounds	3E+3	6E+3	3E-6	9E-9	4E-5	4E-4
63	Europium-152	W, all compounds	8E+2	2E+1	1E-8	3E-11	1E-5	1E-4
63	Europium-154	W, all compounds	5E+2	2E+1	8E-9	3E-11	7E-6	7E-5
63	Europium-155	W, all compounds	4E+3	9E+1 Bone Surf	4E-8	-	5E-5	5E-4
			-	(1E+2)	-	2E-10	-	
63	Europium-156	W, all compounds	6E+2	5E+2	2E-7	6E-10	8E-6	8E-5
63	Europium-157	W, all compounds	2E+3	5E+3	2E-6	7E-9	3E-5	3E-4
63	Europium-158(2)	W, all compounds	2E+4	6E+4	2E-5	8E-8	3E-4	3E-3
64	Gadolinium-145(2)	D, all compounds except those given for W	5E+4 St. wall	2E+5	6E-5	2E-7	-	
			(5E+4)	-	-	-	6E-4	6E-3
		W, oxides, hydroxides, and fluorides	-	2E+5	7E-5	2E-7	-	
64	Gadolinium-146	D, see ^{145}Gd	1E+3	1E+2	5E-8	2E-10	2E-5	2E-4
		W, see ^{145}Gd	-	3E+2	1E-7	4E-10	-	
64	Gadolinium-147	D, see ^{145}Gd	2E+3	4E+3	2E-6	6E-9	3E-5	3E-4
		W, see ^{145}Gd	-	4E+3	1E-6	5E-9	-	

Atomic No.	Radionuclide	Class	Table 1 Oc upational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion	Inhalation				Monthly Average Concentration
			ALI (μCi)	ALI (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	(μCi/ml)
64	Gadolinium- 148	D, see ^{145}Gd	1E+1 Bone Surf	8E+3 Bone Surf	3E-12	-	-	-
			(2E+1)	(2E+2)	-	2E-14	3E-7	3E-6
		W, see ^{145}Gd	-	3E-2 Bone Surf	1E-11	-	-	-
				(6E-2)	-	8E-14	-	
64	Gadolinium- 149	D, see ^{145}Gd	3E+3	2E+3	9E-7	3E-9	4E-5	4E-4
		W, see ^{145}Gd		2E+3	1E-6	3E-9	-	-
64	Gadolinium- 15 1	D, see ^{145}Gd	6E+3	4E+2 Bone Surf	2E-7	-	9E-5	9E-4
				(6E+2)	-	9E-10	-	-
		W, see ^{145}Gd		1E+3	5E-7	2E-9	-	
64	Gadolinium- 152	D, see ^{145}Gd	2E+1 Bone Surf	1E-2 Bone Surf	4E-12	-		
			(3E+1)	(2E-2)	-	3E-14	4E-7	4E-6
		W. see ^{145}Gd		4E-2 Bone Surf	2E-11	-	-	-
				(8E-2)	-	1E-13	-	-
64	Gadolinium-153	D. see ^{145}Gd	5E+3	1E+2 Bone Surf	6E-8		6E-5	6E-4
				(2E+2)	-	3E-10	-	
		W, see ^{145}Gd		6E+2	2E-7	8E-10	-	
64	Gadolinium- 159	D, see ^{145}Gd	3E+3	8E+3	3E-6	1E-8	4E-5	4E-4
		W, see ^{145}Gd		6E+3	2E-6	8E-9	-	
65	Terbium- 147[2]	W, all compounds	9E+3	3E+4	1E-5	5E-8	1E-4	1E-3
65	Terbium- 149	W, all compounds	5E+3	7E+2	3E-7	1E-9	7E-5	7E-4
65	rerbium- 150	W, all compounds	5E+3	2E+4	9E-6	3E-8	7E-5	7E-4
65	Terbium- 15 1	W. all compounds	4E+3	9E+3	4E-6	1E-8	5E-5	5E-4
65	rerbium- 153	W, all compounds	5E+3	7E+3	3E-6	1E-8	7E-5	7E-4
65	rerbium- 154	W, all compounds	2E+3	4E+3	2E-6	6E-9	2E-5	2E-4
65	rerbium- 155	W, all compounds	6E+3	8E+3	3E-6	1E-8	8E-5	8E-4

			Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion	Inhalation				Monthly Average
Atomic No.	Radionuclide	Class	ALI (μCi)	ALI (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	Concentration (μCi/ml)
65	Terbium-156m (5.0h)	W, all compounds	2E+4	3E+4	1E-5	4E-8	2E-4	2E-3
65	Terbium-156m (24.4h)	W, all compounds	7E+3	8E+3	3E-6	1E-8	1E-4	1E-3
65	Terbium-156	W, all compounds	1E+3	1E+3	6E-7	2E-9	1E-5	1E-4
65	Terbium-157	W, all compounds	5E+4 LLI wall	3E+2 Bone Surf	1E-7			
			(5E+4)	(6E+2)	-	8E-10	7E-4	7E-3
65	Terbium-158	W, all compounds	1E+3	2E+1	8E-9	3E-11	2E-5	2E-4
65	Terbium-160	W, all compounds	8E+2	2E+2	9E-8	3E-10	1E-5	1E-4
65	Terbium-161	W, all compounds	2E+3 LLI wall	2E+3	7E-7	2E-9		
			(2E+3)	-	-	-	3E-5	3E-4
66	Dysprosium-155	W, all compounds	9E+3	3E+4	1E-5	4E-8	1E-4	1E-3
66	Dysprosium-157	W, all compounds	2E+4	6E+4	3E-5	9E-8	3E-4	3E-3
66	Dysprosium-159	W, all compounds	1E+4	2E+3	1E-6	3E-9	2E-4	2E-3
66	Dysprosium-165	W, all compounds	1E+4	5E+4	2E-5	6E-8	2E-4	2E-3
66	Dysprosium-166	W, all compounds	6E+2 LLI wall	7E+2	3E-7	1E-9		
			(8E+2)	-			1E-5	1E-4
67	Holmium-155(2)	W, all compounds	4E+4	2E+5	6E-5	2E-7	6E-4	6E-3
67	Holmium-157(2)	W, all compounds	3E+5	1E+6	6E-4	2E-6	4E-3	4E-2
67	Holmium-159(2)	W, all compounds	2E+5	1E+6	4E-4	1E-6	3E-3	3E-2
67	Holmium-161	W, all compounds	1E+5	4E+5	2E-4	6E-7	1E-3	1E-2
67	Holmium-162m(2)	W, all compounds	5E+4	3E+5	1E-4	4E-7	7E-4	7E-3
67	Holmium-162(2)	W, all compounds	5E+5 St. wall	2E+6	1E-3	3E-6		
			(8E+5)	-			1E-2	1E-1
67	Holmium-164m(2)	W, all compounds	1E+5	3E+5	1E-4	4E-7	1E-3	1E-2

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
67	Holmium-164[2]	W, all compounds	2E+5 St. wall	6E+5	3E-4	9E-7	-	-
			(2E+5)	-	-	-	3E-3	3E-2
67	Holmium-166m	W, all compounds	6E+2	7E+0	3E-9	9E-12	9E-6	9E-5
67	Holmium-166	W, all compounds	9E+2 LLI wall	2E+3	7E-7	2E-9	-	-
			(9E+2)	-	-	-	1E-5	1E-4
67	Holmium-167	W, all compounds	2E+4	6E+4	2E-5	8E-8	2E-4	2E-3
68	Erbium-161	W, all compounds	2E+4	6E+4	3E-5	9E-8	2E-4	2E-3
68	Erbium-165	W. all compounds	6E+4	2E+5	8E-5	3E-7	9E-4	9E-3
68	Erbium-169	W, all compounds	3E+3 LLI wall	3E+3	1E-6	4E-9		
			(4E+3)	-			5E-5	5E-4
68	Erbium-171	W, all compounds	4E+3	1E+4	4E-6	1E-8	5E-5	5E-4
68	Erbium-172	W, all compounds	1E+3 LLI wall	1E+3	6E-7	2E-9	-	
			(1E+3)	-			2E-5	2E-4
69	Thulium-162[2]	W, all compounds	7E+4 St. wall	3E+5	1E-4	4E-7		
			(7E+4)	-			1E-3	1E-2
69	Thulium-166	W, all compounds	4E+3	1E+4	6E-6	2E-8	6E-5	6E-4
69	Thulium-167	W, all compounds	2E+3 LLI wall	2E+3	8E-7	3E-9		
			(2E+3)	-			3E-5	3E-4
69	Thulium-170	W, all compounds	8E+2 LLI wall	2E+2	9E-8	3E-10	-	
			(1E+3)	-	-	-	1E-5	1E-4
69	Thulium-171	W, all compounds	1E+4 LLI wall	3E+2 Bone Surf	1E-7	-		
			(1E+4)	(6E+2)	-	8E-10	2E-4	2E-3

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion AL1 (μCi)	Table 1 Col. 2 Inhalation AL1 (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
69	Thulium-172	W, all compounds	7E+2 LLI wall	1E+3	5E-7	2E-9	-	
			(8E+2)	-	-	-	1E-5	1E-4
69	Thulium-173	W, all compounds	4E+3	1E+4	5E-6	2E-8	6E-5	6E-4
69	Thulium-175(2)	W, all compounds	7E+4 St. wall	3E+5	1E-4	4E-7	-	
			(9E+4)	-	-	-	1E-3	1E-2
70	Ytterbium-162(2)	W, all compounds except those given for Y	7E+4	3E+5	1E-4	4E-7	1E-3	1E-2
		Y, oxides, hydroxides, and fluorides	-	3E+5	1E-4	4E-7	-	
70	Ytterbium-166	W, see ^{162}Yb	1E+3	2E+3	8E-7	3E-9	2E-5	2E-4
		Y, see ^{162}Yb	-	2E+3	8E-7	3E-9		
70	Ytterbium-167(2)	W, see ^{162}Yb	3E+5	8E+5	3E-4	1E-6	4E-3	4E-2
		Y, see ^{162}Yb	-	7E+5	3E-4	1E-6		
70	Ytterbium-169	W, see ^{162}Yb	2E+3	8E+2	4E-7	1E-9	2E-5	2E-4
		Y, see ^{162}Yb	-	7E+2	3E-7	1E-9		
70	Ytterbium-175	W, see ^{162}Yb	3E+3 LLI wall	4E+3	1E-6	5E-9		
			(3E+3)	-	-	-	4E-5	4E-4
		Y, see ^{162}Yb	-	3E+3	1E-6	5E-9		
70	Ytterbium-177(2)	W, see ^{162}Yb	2E+4	5E+4	2E-5	7E-8	2E-4	2E-3
		Y, see ^{162}Yb	-	5E+4	2E-5	6E-8		
70	Ytterbium-178(2)	W, see ^{162}Yb	1E+4	4E+4	2E-5	6E-8	2E-4	2E-3
		Y, see ^{162}Yb		4E+4	2E-5	5E-8		
71	Lutetium-169	W, all compounds except those given for Y	3E+3	4E+3	2E-6	6E-9	3E-5	3E-4
		Y, oxides, hydroxides, and fluorides	-	4E+3	2E-6	6E-9		
71	Lutetium-170	W, see ^{169}Lu	1E+3	2E+3	9E-7	3E-9	2E-5	2E-4
		Y, see ^{169}Lu		2E+3	8E-7	3E-9		

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
71	Lutetium-171	W, see ^{169}Lu	2E+3	2E+3	8E-7	3E-9	3E-5	3E-4
		Y, see ^{169}Lu	-	2E+3	8E-7	3E-9	-	-
71	Lutetium- 172	W, see ^{169}Lu	1E+3	1E+3	5E-7	2E-9	1E-5	1E-4
		Y, see ^{169}Lu	-	1E+3	5E-7	2E-9	-	-
71	Lutetium- 173	W, see ^{169}Lu	5E+3	3E+2 Bone Surf	1E-7	-	7E-5	7E-4
			-	(5E+2)	-	6E-10	-	-
		Y, see ^{169}Lu	-	3E+2	1E-7	4E-10	-	-
71	Lutetium-174m	W, see ^{169}Lu	2E+3 LLI wall	2E+2 Bone Surf	1E-7	-	-	-
			(3E+3)	(3E+2)	-	5E-10	4E-5	4E-4
		Y, see ^{169}Lu	-	2E+2	9E-8	3E-10	-	-
71	Lutetium- 174	W, see ^{169}Lu	5E+3	1E+2 Bone Surf	5E-8	-	7E-5	7E-4
			-	(2E+2)	-	3E-10	-	-
		Y, see ^{169}Lu	-	2E+2	6E-8	2E-10	-	-
71	Lutetium-176m	W, see ^{169}Lu	8E+3	3E+4	1E-5	3E-8	1E-4	1E-3
		Y, see ^{169}Lu	-	2E+4	9E-6	3E-8	-	-
71	Lutetium-176	W, see ^{169}Lu	7E+2	5E+0 Bone Surf	2E-9	-	1E-5	1E-4
			-	(1E+1)	-	2E-11	-	-
		Y, see ^{169}Lu	-	8E+0	3E-9	1E-11	-	-
71	Lutetium-177m	W, see ^{169}Lu	7E+2	1E+2 Bone Surf	5E-8	-	1E-5	1E-4
			-	(1E+2)	-	2E-10	-	-
		Y, see ^{169}Lu	-	8E+1	3E-8	1E-10	-	-
71	Lutetium- 177	W, see ^{169}Lu	2E+3 LLI wall	2E+3	9E-7	3E-9	-	-
			(3E+3)	-	-	-	4E-5	4E-4
		Y, see ^{169}Lu	-	2E+3	9E-7	3E-9	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
71	Lutetium-178m(2)	W, see ^{169}Lu	5E+4 St. wall	2E+5	8E-5	3E-7	-	
			(6E+4)	-			8E-4	8E-3
		Y, see ^{169}Lu		2E+5	7E-5	2E-7		
71	Lutetium-178(2)	W, see ^{169}Lu	4E+4 St. wall	1E+5	5E-5	2E-7	-	
			(4E+4)	-			6E-4	6E-3
		Y, see ^{169}Lu		1E+5	5E-5	2E-7	-	
71	Lutetium- 179	W, see ^{169}Lu	6E+3	2E+4	8E-6	3E-8	9E-5	9E-4
		Y, see ^{169}Lu		2E+4	6E-6	3E-8	-	
72	Hafnium- 170	D, all compounds except those given for W	3E+3	6E+3	2E-6	8E-9	4E-5	4E-4
		W, oxides, hydroxides, carbides, and nitrates		5E+3	2E-6	6E-9	-	
72	Hafnium- 172	D, see ^{170}Hf	1E+3	9E+0 Bone Surf	4E-9		2E-5	2E-4
				(2E+1)	-	3E-11	-	
		W, see ^{170}Hf		4E+1 Bone Surf	2E-8			
				(6E+1)	-	8E-11	-	
72	Hafnium- 173	D, see ^{170}Hf	5E+3	1E+4	5E-6	2E-8	7E-5	7E-4
		W, see ^{170}Hf		1E+4	5E-6	2E-8	-	
72	Hafnium-175	D, see ^{170}Hf	3E+3	9E+2 Bone Surf	4E-7		4E-5	4E-4
				(1E+3)	-	1E-9		
		W, see ^{170}Hf		1E+3	5E-7	2E-9	-	
72	Hafnium-177m(2)	D, see ^{170}Hf	2E+4	6E+4	2E-5	8E-8	3E-4	3E-3
		W, see ^{170}Hf		9E+4	4E-5	1E-7	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values, Col. 1: Oral Ingestion ALI (μCi)	Table 1 Col. 2: Inhalation ALI (μCi)	Table 1 Col. 3: Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations, Col. 1: Air (μCi/ml)	Table 2 Col. 2: Water (μCi/ml)	Table 3 Releases to Sewers: Monthly Average Concentration (μCi/ml)
72	Hafnium-178m	D, see ^{170}Hf	3E+2	1E+0 Bone Surf	5E-10	-	3E-6	3E-5
			-	(2E+0)	-	3E-12	-	-
		W, see ^{170}Hf		5E+0 Bone Surf	2E-9	-	-	-
				(9E+0)	-	1E-11	-	-
72	Hafnium-179m	D, see ^{170}Hf	1E+3	3E+2 Bone Surf	1E-7		1E-5	1E-4
				(6E+2)	-	8E-10	-	
		W, see ^{170}Hf	-	6E+2	3E-7	8E-10	-	
72	Hafnium-180m	D, see ^{170}Hf	7E+3	2E+4	9E-6	3E-8	1E-4	1E-3
		W, see ^{170}Hf	-	3E+4	1E-5	4E-8	-	-
72	Hafnium-181	D, see ^{170}Hf	1E+3	2E+2 Bone Surf	7E-8	-	2E-5	2E-4
			-	(4E+2)	-	6E-10	-	
		W, see ^{170}Hf	-	4E+2	2E-7	6E-10	-	
72	Hafnium-182m[2]	D, see ^{170}Hf	4E+4	9E+4	4E-5	1E-7	5E-4	5E-3
		W, see ^{170}Hf	-	1E+5	6E-5	2E-7	-	-
72	Hafnium-182	D, see ^{170}Hf	2E+2 Bone Surf	8E-1 Bone Surf	3E-10	-	-	-
			(4E+2)	(2E+0)	-	2E-12	5E-6	5E-5
		W. see ^{170}Hf	-	3E+0 Bone Surf	1E-9			
			-	(7E+0)	-	1E-11	-	
72	Hafnium-183[2]	D, see ^{170}Hf	2E+4	5E+4	2E-5	6E-8	3E-4	3E-3
		W, see ^{170}Hf	-	6E+4	2E-5	8E-8	-	
72	Hafnium-184	D, see ^{170}Hf	2E+3	8E+3	3E-6	1E-8	3E-5	3E-4
		W, see ^{170}Hf	-	6E+3	3E-6	9E-9	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
73	Tantalum-172[2]	W, all compounds except those given for Y	4E+4	1E+5	5E-5	2E-7	5E-4	5E-3
		Y, elemental Ta, oxides, hydroxides, halides, carbides, nitrates, and nitrides		1E+5	4E-5	1E-7		
73	Tantalum-173	W, see ^{172}Ta	7E+3	2E+4	8E-6	3E-8	9E-5	9E-4
		Y, see ^{172}Ta		2E+4	7E-6	2E-8		
73	Tantalum-174[2]	W, see ^{172}Ta	3E+4	1E+5	4E-5	1E-7	4E-4	4E-3
		Y, see ^{172}Ta		9E+4	4E-5	1E-7		
73	Tantalum-175	W, see ^{172}Ta	6E+3	2E+4	7E-6	2E-8	8E-5	8E-4
		Y, see ^{172}Ta		1E+4	6E-6	2E-8		
73	Tantalum-176	W, see ^{172}Ta	4E+3	1E+4	5E-6	2E-8	5E-5	5E-4
		Y, see ^{172}Ta		1E+4	5E-6	2E-8		
73	Tantalum-177	W, see ^{172}Ta	1E+4	2E+4	8E-6	3E-8	2E-4	2E-3
		Y, see ^{172}Ta		2E+4	7E-6	2E-8		
73	Tantalum-178	W, see ^{172}Ta	2E+4	9E+4	4E-5	1E-7	2E-4	2E-3
		Y, see ^{172}Ta		7E+4	3E-5	1E-7		
73	Tantalum-179	W, see ^{172}Ta	2E+4	5E+3	2E-6	8E-9	3E-4	3E-3
		Y, see ^{172}Ta		9E+2	4E-7	1E-9		
73	Tantalum-180m	W, see ^{172}Ta	2E+4	7E+4	3E-5	9E-8	3E-4	3E-3
		Y, see ^{172}Ta		6E+4	2E-5	8E-8		
73	Tantalum-180	W, see ^{172}Ta	1E+3	4E+2	2E-7	6E-10	2E-5	2E-4
		Y, see ^{172}Ta	-	2E+1	1E-8	3E-11		
73	Tantalum-182m[2]	W, see ^{172}Ta	2E+5 St. wall	5E+5	2E-4	8E-7		
			(2E+5)	-	-	-	3E-3	3E-2
		Y, see ^{172}Ta	-	4E+5	2E-4	6E-7		
73	Tantalum-182	W, see ^{172}Ta	8E+2	3E+2	1E-7	5E-10	1E-5	1E-4
		Y, see ^{172}Ta	-	1E+2	6E-8	2E-10		

Atomic No.	Radionuclide	Class	Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion	Inhalation				Monthly Average
			ALI (μCi)	ALI (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	Concentration (μCi/ml)
73	Tantalum-183	W, see ^{172}Ta	9E+2 LLI wall	1E+3	5E-7	2E-9	-	-
			(1E+3)	-	-	-	2E-5	2E-4
		Y, see ^{172}Ta	-	1E+3	4E-7	1E-9	-	-
73	Tantalum-184	W, see ^{172}Ta	2E+3	5E+3	2E-6	8E-9	3E-5	3E-4
		Y, see ^{172}Ta	-	5E+3	2E-6	7E-9	-	-
73	Tantalum-185[2]	W, see ^{172}Ta	3E+4	7E+4	3E-5	1E-7	4E-4	4E-3
		Y, see ^{172}Ta	-	6E+4	3E-5	9E-8	-	-
73	Tantalum-186[2]	W, see ^{172}Ta	5E+4 St. wall	2E+5	1E-4	3E-7	-	-
			(7E+4)	-	-	-	1E-3	1E-2
		Y, see ^{172}Ta	-	2E+5	9E-5	3E-7	-	-
74	Tungsten-176	D, all compounds	1E+4	5E+4	2E-5	7E-8	1E-4	1E-3
74	Tungsten-177	D, all compounds	2E+4	9E+4	4E-5	1E-7	3E-4	3E-3
74	Tungsten-178	D, all compounds	5E+3	2E+4	8E-6	3E-8	7E-5	7E-4
74	Tungsten-179[2]	D, all compounds	5E+5	2E+6	7E-4	2E-6	7E-3	7E-2
74	Tungsten-181	D, all compounds	2E+4	3E+4	1E-5	5E-8	2E-4	2E-3
74	Tungsten-185	D, all compounds	2E+3 LLI wall	7E+3	3E-6	9E-9	-	
			(3E+3)	-			4E-5	4E-4
74	Tungsten-187	D, all compounds	2E+3	9E+3	4E-6	1E-8	3E-5	3E-4
74	Tungsten-188	D, all compounds	4E+2 LLI wall	1E+3	5E-7	2E-9	-	-
			(5E+2)	-	-	-	7E-6	7E-5
75	Rhenium-177[2]	D, all compounds except those given for W	9E+4 St. wall	3E+5	1E-4	4E-7	-	-
			(1E+5)	-	-	-	2E-3	2E-2
		W, oxides, hydroxides, and nitrates	-	4E+5	1E-4	5E-7	-	-

			Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion	Inhalation				Monthly Average
Atomic No.	Radionuclide	Class	ALI (μCi)	ALI (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	Concentration (μCi/ml)
75	Rhenium-178[2]	D, see ^{177}Re	7E+4 St. wall	3E+5	1E-4	4E-7	-	-
			(1E+5)	-	-	-	1E-3	1E-2
		W, see ^{177}Re	-	3E+5	1E-4	4E-7	-	-
75	Rhenium-181	D, see ^{177}Re	5E+3	9E+3	4E-6	1E-8	7E-5	7E-4
		W, see ^{177}Re	-	9E+3	4E-6	1E-8	-	-
75	Rhenium-182 (12.7h)	D, see ^{177}Re	7E+3	1E+4	5E-6	2E-8	9E-5	9E-4
		W, see ^{177}Re	-	2E+4	6E-6	2E-8	-	-
75	Rhenium-182 (64.0h)	D, see ^{177}Re	1E+3	2E+3	1E-6	3E-9	2E-5	2E-4
		W, see ^{177}Re	-	2E+3	9E-7	3E-9	-	-
75	Rhenium-184m	D, see ^{177}Re	2E+3	3E+3	1E-6	4E-9	3E-5	3E-4
		W, see ^{177}Re	-	4E+2	2E-7	6E-10	-	-
75	Rhenium-184	D, see ^{177}Re	2E+3	4E+3	1E-6	5E-9	3E-5	3E-4
		W, see ^{177}Re	-	1E+3	6E-7	2E-9	-	-
75	Rhenium-186m	D, see ^{177}Re	1E+3 St. wall	2E+3 St. wall	7E-7	-	-	-
			(2E+3)	(2E+3)	-	3E-9	2E-5	2E-4
		W, see ^{177}Re	-	2E+2	6E-8	2E-10	-	-
75	Rhenium-186	D, see ^{177}Re	2E+3	3E+3	1E-6	4E-9	3E-5	3E-4
		W, see ^{177}Re	-	2E+3	7E-7	2E-9	-	-
75	Rhenium-187	D, see ^{177}Re	6E+5	8E+5 St. wall	4E-4	-	8E-3	8E-2
			-	(9E+5)	-	1E-6	-	-
		W, see ^{177}Re	-	1E+5	4E-5	1E-7	-	-
75	Rhenium-188m[2]	D, see ^{177}Re	8E+4	1E+5	6E-5	2E-7	1E-3	1E-2
		W, see ^{177}Re	-	1E+5	6E-5	2E-7	-	-
75	Rhenium-188	D, see ^{177}Re	2E+3	3E+3	1E-6	4E-9	2E-5	2E-4
		W, see ^{177}Re	-	3E+3	1E-6	4E-9	-	-
75	Rhenium-189	D, see ^{177}Re	3E+3	5E+3	2E-6	7E-9	4E-5	4E-4
		W, see ^{177}Re		4E+3	2E-6	6E-9	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
76	Osmium-180[2]	D, all compounds except those given for W and Y	1E+5	4E+5	2E-4	5E-7	1E-3	1E-2
		W, halides and nitrates	-	5E+5	2E-4	7E-7	-	-
		Y, oxides and hydroxides	-	5E+5	2E-4	6E-7	-	-
76	Osmium-181[2]	D, see ^{180}Os	1E+4	4E+4	2E-5	6E-8	2E-4	2E-3
		W, see ^{180}Os	-	5E+4	2E-5	6E-8	-	-
		Y, see ^{180}Os	-	4E+4	2E-5	6E-8	-	-
76	Osmium-182	D, see ^{180}Os	2E+3	6E+3	2E-6	8E-9	3E-5	3E-4
		W, see ^{180}Os	-	4E+3	2E-6	6E-9	-	-
		Y, see ^{180}Os	-	4E+3	2E-6	6E-9	-	-
76	Osmium-185	D, see ^{180}Os	2E+3	5E+2	2E-7	7E-10	3E-5	3E-4
		W, see ^{180}Os	-	8E+2	3E-7	1E-9	-	-
		Y, see ^{180}Os	-	8E+2	3E-7	1E-9	-	-
76	Osmium-189m	D, see ^{180}Os	8E+4	2E+5	1E-4	3E-7	1E-3	1E-2
		W, see ^{180}Os	-	2E+5	9E-5	3E-7	-	-
		Y, see ^{180}Os	-	2E+5	7E-5	2E-7	-	-
76	Osmium-191m	D, see ^{180}Os	1E+4	3E+4	1E-5	4E-8	2E-4	2E-3
		W, see ^{180}Os	-	2E+4	8E-6	3E-8	-	-
		Y, see ^{180}Os	-	2E+4	7E-6	2E-8	-	-
76	Osmium-191	D, see ^{180}Os	2E+3 LLI wall	2E+3	9E-7	3E-9	-	-
			(3E+3)	-	-	-	3E-5	3E-4
		W, see ^{180}Os	-	2E+3	7E-7	2E-9	-	-
		Y, see ^{180}Os	-	1E+3	6E-7	2E-9	-	-
76	Osmium-193	D, see ^{180}Os	2E+3 LLI wall	5E+3	2E-6	6E-9	-	-
			(2E+3)	-	-	-	2E-5	2E-4
		W, see ^{180}Os	-	3E+3	1E-6	4E-9	-	-
		Y, see ^{180}Os	-	3E+3	1E-6	4E-9	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Col. 2 Inhalation ALI (μCi)	Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
76	Osmium-194	D, see ^{180}Os	4E+2 LLI wall	4E+1	2E-8	6E-11	-	
			(6E+2)	-			8E-6	8E-5
		W, see ^{180}Os		6E+1	2E-8	8E-11	-	
		Y, see ^{180}Os		8E+0	3E-9	1E-11	-	
77	Iridium-182(2)	D, all compounds except those given for W and Y	4E+4 St. wall	1E+5	6E-5	2E-7	-	
			(4E+4)	-			6E-4	6E-3
		W, halides, nitrates, and metallic iridium	-	2E+5	6E-5	2E-7	-	-
		Y, oxides and hydroxides	-	1E+5	5E-5	2E-7	-	-
77	Iridium-184	D, see ^{182}Ir	8E+3	2E+4	1E-5	3E-8	1E-4	1E-3
		W, see ^{182}Ir	-	3E+4	1E-5	5E-8	-	-
		Y, see ^{182}Ir	-	3E+4	1E-5	4E-8	-	-
77	Iridium-185	D, see ^{182}Ir	5E+3	1E+4	5E-6	2E-8	7E-5	7E-4
		W, see ^{182}Ir	-	1E+4	5E-6	2E-8	-	-
		Y, see ^{182}Ir	-	1E+4	4E-6	1E-8	-	-
77	Iridium-186	D, see ^{182}Ir	2E+3	8E+3	3E-6	1E-8	3E-5	3E-4
		W, see ^{182}Ir	-	6E+3	3E-6	9E-9	-	-
		Y, see ^{182}Ir	-	6E+3	2E-6	8E-9	-	-
77	Iridium-187	D, see ^{182}Ir	1E+4	3E+4	1E-5	5E-8	1E-4	1E-3
		W, see ^{182}Ir	-	3E+4	1E-5	4E-8	-	-
		Y, see ^{182}Ir	-	3E+4	1E-5	4E-8	-	-
77	Iridium-188	D, see ^{182}Ir	2E+3	5E+3	2E-6	6E-9	3E-5	3E-4
		W, see ^{182}Ir	-	4E+3	1E-6	5E-9	-	-
		Y, see ^{182}Ir	-	3E+3	1E-6	5E-9	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
77	Iridium-1 89	D, see ^{182}Ir	5E+3 LLI wall	5E+3	2E-6	7E-9	-	
			(5E+3)	-		-	7E-5	7E-4
		W, see ^{182}Ir		4E+3	2E-6	5E-9	-	
		Y, see ^{182}Ir	-	4E+3	1E-6	5E-9	-	
77	Iridium-190m(2)	D, see ^{182}Ir	2E+5	2E+5	8E-5	3E-7	2E-3	2E-2
		W, see ^{182}Ir		2E+5	9E-5	3E-7	-	
		Y. see ^{182}Ir		2E+5	8E-5	3E-7	-	-
77	Iridium- 190	D, see ^{182}Ir	1E+3	9E+2	4E-7	1E-9	1E-5	1E-4
		W, see ^{182}Ir		1E+3	4E-7	1E-9		
		Y, see ^{182}Ir		9E+2	4E-7	1E-9	-	
77	Iridium-192m	D, see ^{182}Ir	3E+3	9E+1	4E-8	1E-10	4E-5	4E-4
		W, see ^{182}Ir		2E+2	9E-8	3E-10	-	
		Y. see ^{182}Ir		2E+1	6E-9	2E-11	-	
77	Iridium-192	D, see ^{182}Ir	9E+2	3E+2	1E-7	4E-10	1E-5	1E-4
		W, see ^{182}Ir	-	4E+2	2E-7	6E-10	-	-
		Y, see ^{182}Ir	-	2E+2	9E-8	3E-10	-	-
77	Iridium-194m	D, see ^{182}Ir	6E+2	9E+1	4E-8	1E-10	9E-6	9E-5
		W, see ^{182}Ir	-	2E+2	7E-8	2E-10	-	-
		Y, see ^{182}Ir	-	1E+2	4E-8	1E-10	-	-
77	Iridium-194	D, see ^{182}Ir	1E+3	3E+3	1E-6	4E-9	1E-5	1E-4
		W, see ^{182}Ir	-	2E+3	9E-7	3E-9	-	-
		Y, see ^{182}Ir	-	2E+3	8E-7	3E-9	-	-
77	Iridium-195m	D, see ^{182}Ir	8E+3	2E+4	1E-5	3E-8	1E-4	1E-3
		W, see ^{182}Ir	-	3E+4	1E-5	4E-8	-	-
		Y, see ^{182}Ir	-	2E+4	9E-6	3E-8	-	-
77	Iridium-195	D, see ^{182}Ir	1E+4	4E+4	2E-5	6E-8	2E-4	2E-3
		W, see ^{182}Ir		5E+4	2E-5	7E-8	-	
		Y, see ^{182}Ir		4E+4	2E-5	6E-8	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
78	Platinum-186	D, all compounds	1E+4	4E+4	2E-5	5E-8	2E-4	2E-3
78	Platinum-188	D, all compounds	2E+3	2E+3	7E-7	2E-9	2E-5	2E-4
78	Platinum-189	D, all compounds	1E+4	3E+4	1E-5	4E-8	1E-4	1E-3
78	Platinum-191	D, all compounds	4E+3	8E+3	4E-6	1E-8	5E-5	5E-4
78	Platinum-193m	D, all compounds	3E+3 LLI wall	6E+3	3E-6	8E-9	-	
			(3E+4)	-			4E-5	4E-4
78	Platinum-193	D, all compounds	4E+4 LLI wall	2E+4	1E-5	3E-8		
			(5E+4)	-			6E-4	6E-3
78	Platinum-195m	D, all compounds	2E+3 LLI wall	4E+3	2E-6	6E-9	-	
			(2E+3)	-			3E-5	3E-4
78	Platinum-197m[2]	D, all compounds	2E+4	4E+4	2E-5	6E-8	2E-4	2E-3
78	Platinum-197	D, all compounds	3E+3	1E+4	4E-6	1E-8	4E-5	4E-4
78	Platinum-199[2]	D, all compounds	5E+4	1E+5	6E-5	2E-7	7E-4	7E-3
78	Platinum-200	D, all compounds	1E+3	3E+3	1E-6	5E-9	2E-5	2E-4
79	Gold-193	D, all compounds except those given for W and Y	9E+3	3E+4	1E-5	4E-8	1E-4	1E-3
		W, halides and nitrates		2E+4	9E-6	3E-8	-	
		Y, oxides and hydroxides		2E+4	8E-6	3E-8	-	
79	Gold-194	D, see ^{193}Au	3E+3	8E+3	3E-6	1E-8	4E-5	4E-4
		W, see ^{193}Au		5E+3	2E-6	8E-9	-	
		Y, see ^{193}Au		5E+3	2E-6	7E-9	-	
79	Gold-195	D, see ^{193}Au	5E+3	1E+4	5E-6	2E-8	7E-5	7E-4
		W, see ^{193}Au		1E+3	6E-7	2E-9		
		Y, see ^{193}Au		4E+2	2E-7	6E-10	-	

			Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion	Inhalation				Monthly Average
Atomic No.	Radionuclide	Class	AL1 (μCi)	AL1 (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	Concentration (μCi/ml)
79	Gold-198m	D, see ^{193}Au	1E+3	3E+3	1E-6	4E-9	1E-5	1E-4
		W, see ^{193}Au	-	1E+3	5E-7	2E-9	-	-
		Y, see ^{193}Au	-	1E+3	5E-7	2E-9	-	-
79	Gold-198	D, see ^{193}Au	1E+3	4E+3	2E-6	5E-9	2E-5	2E-4
		W, see ^{193}Au	-	2E+3	8E-7	3E-9	-	-
		Y, see ^{193}Au	-	2E+3	7E-7	2E-9	-	-
79	Gold-199	D, see ^{193}Au	3E+3 LLI wall	9E+3	4E-6	1E-8	-	-
			(3E+3)	-	-	-	4E-5	4E-4
		W, see ^{193}Au	-	4E+3	2E-6	6E-9	-	-
		Y, see ^{193}Au	-	4E+3	2E-6	5E-9	-	-
79	Gold-200m	D, see ^{193}Au	1E+3	4E+3	1E-6	5E-9	2E-5	2E-4
		W, see ^{193}Au	-	3E+3	1E-6	4E-9	-	-
		Y, see ^{193}Au	-	2E+4	1E-6	3E-9	-	-
79	Gold-200[2]	D, see ^{193}Au	3E+4	6E+4	3E-5	9E-8	4E-4	4E-3
		W, see ^{193}Au	-	8E+4	3E-5	1E-7	-	-
		Y, see ^{193}Au	-	7E+4	3E-5	1E-7	-	-
79	Gold-201[2]	D, see ^{193}Au	7E+4 St. wall	2E+5	9E-5	3E-7	-	-
			(9E+4)	-	-	-	1E-3	1E-2
		W, see ^{193}Au	-	2E+5	1E-4	3E-7	-	-
		Y, see ^{193}Au	-	2E+5	9E-5	3E-7	-	-
80	Mercury-193m	Vapor	-	8E+3	4E-6	1E-8	-	-
		Organic D	4E+3	1E+4	5E-6	2E-8	6E-5	6E-4
		D, sulfates	3E+3	9E+3	4E-6	1E-8	4E-5	4E-4
		W, oxides, hydroxides, halides, nitrates, and sulfides	-	8E+3	3E-6	1E-8	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
80	Mercury- 193	Vapor	-	3E+4	1E-5	4E-8	-	-
		Organic D	2E+4	6E+4	3E-5	9E-8	3E-4	3E-3
		D, see ^{193m}Hg	2E+4	4E+4	2E-5	6E-8	2E-4	2E-3
		W, see ^{193m}Hg	-	4E+4	2E-5	6E-8	-	-
80	Mercury- 194	Vapor	-	3E+1	1E-8	4E-11	-	-
		Organic D	2E+1	3E+1	1E-8	4E-11	2E-7	2E-6
		D, see ^{193m}Hg	8E+2	4E+1	2E-8	6E-11	1E-5	1E-4
		W, see ^{193m}Hg	-	1E+2	5E-8	2E-10	-	-
80	Mercury- 195m	Vapor	-	4E+3	2E-6	6E-9	-	-
		Organic D	3E+3	6E+3	3E-6	8E-9	4E-5	4E-4
		D, see ^{193m}Hg	2E+3	5E+3	2E-6	7E-9	3E-5	3E-4
		W, see ^{193m}Hg	-	4E+3	2E-6	5E-9	-	-
80	Mercury- 195	Vapor	-	3E+4	1E-5	4E-8	-	-
		Organic D	2E+4	5E+4	2E-5	6E-8	2E-4	2E-3
		D, see ^{193m}Hg	1E+4	4E+4	1E-5	5E-8	2E-4	2E-3
		W, see ^{193m}Hg	-	3E+4	1E-5	5E-8	-	-
80	Mercury-197m	Vapor	-	5E+3	2E-6	7E-9	-	-
		Organic D	4E+3	9E+3	4E-6	1E-8	5E-5	5E-4
		D, see ^{193m}Hg	3E+3	7E+3	3E-6	1E-8	4E-5	4E-4
		W, see ^{193m}Hg	-	5E+3	2E-6	7E-9	-	-
80	Mercury-197	Vapor	-	8E+3	4E-6	1E-8	-	-
		Organic D	7E+3	1E+4	6E-6	2E-8	9E-5	9E-4
		D, see ^{193m}Hg	6E+3	1E+4	5E-6	2E-8	8E-5	8E-4
		W, see ^{193m}Hg		9E+3	4E-6	1E-8		

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
80	Mercury-199m(2)	Vapor	-	8E+4	3E-5	1E-7	-	-
		Organic D	6E+4 St. wall	2E+5	7E-5	2E-7	-	-
			(1E+5)	-	-	-	1E-3	1E-2
		D, see ^{193m}Hg	6E+4	1E+5	6E-5	2E-7	8E-4	8E-3
		W, see ^{193m}Hg	-	2E+5	7E-5	2E-7	-	-
80	Mercury-203	Vapor	-	8E+2	4E-7	1E-9	-	-
		Organic D	5E+2	8E+2	3E-7	1E-9	7E-6	7E-5
		D, see ^{193m}Hg	2E+3	1E+3	5E-7	2E-9	3E-5	3E-4
		W, see ^{193m}Hg	-	1E+3	5E-7	2E-9	-	-
81	Thallium-194m(2)	D, all compounds	5E+4 St. wall	2E+5	6E-5	2E-7	-	-
			(7E+4)	-	-	-	1E-3	1E-2
81	Thallium-194(2)	D, all compounds	3E+5 St. wall	6E+5	2E-4	8E-7	-	-
			(3E+5)	-	-	-	4E-3	4E-2
81	Thallium-195(2)	D, all compounds	6E+4	1E+5	5E-5	2E-7	9E-4	9E-3
81	Thallium-197	D, all compounds	7E+4	1E+5	5E-5	2E-7	1E-3	1E-2
81	Thallium-198m(2)	D, all compounds	3E+4	5E+4	2E-5	8E-8	4E-4	4E-3
81	Thallium-198	D, all compounds	2E+4	3E+4	1E-5	5E-8	3E-4	3E-3
81	Thallium-199	D, all compounds	6E+4	8E+4	4E-5	1E-7	9E-4	9E-3
81	Thallium-200	D, all compounds	8E+3	1E+4	5E-6	2E-8	1E-4	1E-3
81	Thallium-201	D, all compounds	2E+4	2E+4	9E-6	3E-8	2E-4	2E-3
81	Thallium-202	D, all compounds	4E+3	5E+3	2E-6	7E-9	5E-5	5E-4
81	Thallium-204	D, all compounds	2E+3	2E+3	9E-7	3E-9	2E-5	2E-4
82	Lead-195m(2)	D, all compounds	6E+4	2E+5	XE-5	3E-7	8E-4	8E-3
82	Lead-198	D, all compounds	3E+4	6E+4	3E-5	9E-8	4E-4	4E-3
82	Lead-199(2)	D, all compounds	2E+4	7E+4	3E-5	1E-7	3E-4	3E-3
82	Lead-200	D, all compounds	3E+3	6E+3	3E-6	9E-9	4E-5	4E-4
82	Lead-201	D, all compounds	7E+3	2E+4	8E-6	3E-8	1E-4	1E-3

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
82	Lead-202m	D, all compounds	9E+3	3E+4	1E-5	4E-8	1E-4	1E-3
82	Lead-202	D, all compounds	1E+2	5E+1	2E-8	7E-11	2E-6	2E-5
82	Lead-203	D, all compounds	5E+3	9E+3	4E-6	1E-8	7E-5	7E-4
82	Lead-205	D, all compounds	4E+3	1E+3	6E-7	2E-9	5E-5	5E-4
82	Lead-209	D, all compounds	2E+4	6E+4	2E-5	8E-8	3E-4	3E-3
82	Lead-210	D, all compounds	6E+1 Bone Surf	2E+1 Bone Surf	1E-10	-		
			(1E+0)	(4E-1)	-	6E-13	1E-8	1E-7
82	Lead-21 1[2]	D, all compounds	1E+4	6E+2	3E-7	9E-10	2E-4	2E-3
82	Lead-21 2	D, all compounds	8E+1 Bone Surf	3E+1	1E-8	SE-11	-	
			·(1E+2)	-			2E-6	2E-5
82	Lead-214[2]	D, all compounds	9E+3	8E+2	3E-7	1E-9	1E-4	1E-3
83	Bismuth-200[2]	D, nitrates	3E+4	8E+4	4E-5	1E-7	4E-4	4E-3
		W, all other compounds		1E+5	4E-5	1E-7	-	
83	Bismuth-201'''	D, see ^{200}Bi	1E+4	3E+4	1E-5	4E-8	2E-4	2E-3
		W, see ^{200}Bi		4E+4	2E-5	5E-8	-	
83	Bismuth-202[2]	D, see ^{200}Bi	1E+4	4E+4	2E-5	6E-8	2E-4	2E-3
		W, see ^{200}Bi		8E+4	3E-5	1E-7	-	
83	Bismuth-203	D. see ^{200}Bi	2E+3	7E+3	3E-6	9E-9	3E-5	3E-4
		W, see ^{200}Bi		6E+3	3E-6	9E-9	-	
83	Bismuth-205	D, see ^{200}Bi	1E+3	3E+3	1E-6	3E-9	2E-5	2E-4
		W, see ^{200}Bi		1E+3	5E-7	2E-9	-	
83	Bismuth-206	D, see ^{200}Bi	6E+2	1E+3	6E-7	2E-9	9E-6	9E-5
		W, see ^{200}Bi		9E+2	4E-7	1E-9	-	
83	Bismuth-207	D, see ^{200}Bi	1E+3	2E+3	7E-7	2E-9	1E-5	1E-4
		W, see ^{200}Bi		4E+2	1E-7	5E-10	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
83	Bismuth-210m	D, see ^{200}Bi	4E+1 Kidneys	5E+0 Kidneys	2E-9	-		
			(6E+1)	(6E+0)	-	9E-12	8E-7	8E-6
		W, see ^{200}Bi		7E-1	3E-10	9E-13	-	
83	Bismuth-210	D, see ^{200}Bi	8E+2	2E+2 Kidneys	1E-7		1E-5	1E-4
				(4E+2)	-	5E-10	-	
		W, see ^{200}Bi	-	3E+1	1E-8	4E-11	-	-
83	Bismuth-212(2)	D, see ^{200}Bi	5E+3	2E+2	1E-7	3E-10	7E-5	7E-4
		W, see ^{200}Bi	-	3E+2	1E-7	4E-10	-	-
83	Bismuth-213(2)	D, see ^{200}Bi	7E+3	3E+2	1E-7	4E-10	1E-4	1E-3
		W, see ^{200}Bi	-	4E+2	1E-7	5E-10	-	-
83	Bismuth-214(2)	D, see ^{200}Bi	2E+4 St. wall	8E+2	3E-7	1E-9	-	-
			(2E+4)	-	-	-	3E-4	3E-3
		W, see ^{200}Bi		9E-2	4E-7	1E-9		
84	Polonium-203(2)	D, all compounds except those given for W	3E+4	6E+4	3E-5	9E-8	3E-4	3E-3
		W, oxides, hydroxides, and nitrates	-	9E+4	4E-5	1E-7		
84	Polonium-205'''	D, see ^{203}Po	2E+4	4E+4	2E-5	5E-8	3E-4	3E-3
		W, see ^{203}Po	-	7E+4	3E-5	1E-7		-
84	Polonium-207	D, see ^{203}Po	8E+3	3E+4	1E-5	3E-8	1E-4	1E-3
		W, see ^{203}Po	-	3E+4	1E-5	4E-8	-	
84	Polonium-210	D, see ^{203}Po	3E+0	6E-1	3E-10	9E-13	4E-8	4E-7
		W, see ^{203}Po	-	6E-1	3E-10	9E-13	-	
85	Astatine-207(2)	D, halides	6E+3	3E+3	1E-6	4E-9	8E-5	8E-4
		W	-	2E+3	9E-7	3E-9	-	
85	Astatine-211	D, halides	1E+2	8E+1	3E-8	1E-10	2E-6	2E-5
		W	-	5E+1	2E-8	8E-11	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
86	Radon-220	With daughters removed	-	2E+4	7E-6	2E-8	-	-
		With daughters present		2E+1 (or 12 working level months)	9E-9 (or 1.0 working level)	3E-11		-
86	Radon-222	With daughters removed		1E+4	4E-6	1E-8	-	-
		With daughters present		1E+2 (or 4 working level months)	3E-8 (or 0.33 working level)	1E-10	-	-
87	Francium-222[2]	D, all compounds	2E+3	5E+2	2E-7	6E-10	3E-5	3E-4
87	Francium-223[2]	D, all compounds	6E+2	8E+2	3E-7	1 E-9	8E-6	8E-5
88	Radium-223	W, all compounds	5E+0 Bone Surf	7E-1	3E-10	9E-13	-	-
			(9E+0)				1E-7	1E-6
88	Radium-224	W, all compounds	8E+0 Bone Surf	2E+0	7E-10	2E-12	-	-
			(2E+1)				2E-7	2E-6
88	Radium-225	W, all compounds	8E+0 Bone Surf	7E-1	3E-10	9E-13	-	-
			(2E+1)				2E-7	2E-6
88	Radium-226	W, all compounds	2E+0 Bone Surf	6E-1	3E-10	9E-13	-	-
			(5E+0)				6E-8	6E-7
88	Radium-227[2]	W, all compounds	2E+4 Bone Surf	1E+4 Bone Surf	6E-6			
			(2E+4)	(2E+4)	-	3E-8	3E-4	3E-3
88	Radium-228	W, all compounds	2E+0 Bone Surf	1E+0	5E-10	2E-12	-	-
			(4E+0)				6E-8	6E-7

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
89	Actinium-224	D, all compounds except those given for W and Y	2E+3 LLI wall	3E+1 Bone Surf	1E-8	-	-	
			(2E+3)	(4E+1)	-	5E-11	3E-5	3E-4
		W, halides and nitrates	-	5E+1	2E-8	7E-11		
		Y, oxides and hydroxides	-	5E+1	2E-8	6E-11		-
89	Actinium-225	D, see ^{224}Ac	5E+1 LLI wall	3E-1 Bone Surf	1E-10	-	-	
			(5E+1)	(5E-1)	-	7E-13	7E-7	7E-6
		W, see ^{224}Ac	-	6E-1	3E-10	9E-13	-	
		Y, see ^{224}Ac	-	6E-1	3E-10	9E-13		
89	Actinium-226	D, see ^{224}Ac	1E+2 LLI wall	3E+0 Bone Surf	1E-9			-
			(1E+2)	(4E+0)	-	5E-12	2E-6	2E-5
		W, see ^{224}Ac	-	5E+0	2E-9	7E-12	-	
		Y, see ^{224}Ac		5E+0	2E-9	6E-12	-	
89	Actinium-227	D, see ^{224}Ac	2E-1 Bone Surf	4E-4 Bone Surf	2E-13	-		
			(4E-1)	(8E-4)	-	1E-15	5E-9	5E-8
		W, see ^{224}Ac		2E-3 Bone Surf	7E-13	-		
				(3E-3)	-	4E-15	-	
		Y, see ^{224}Ac	-	4E-3	2E-12	6E-15	-	
89	Actinium-228	D, see ^{224}Ac	2E+3	9E+0 Bone Surf	4E-9		3E-5	3E-4
				(2E+1)	-	2E-11	-	
		W, see ^{224}Ac		4E+1 Bone Surf	2E-8	-		
				(6E+1)	-	8E-11	-	
		Y, see ^{224}Ac		4E+1	2E-8	6E-11	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
90	Thorium-226[2]	W, all compounds except those given for Y	5E+3 St. wall	2E+2	6E-8	2E-10	-	
			(5E+3)		-	-	7E-5	7E-4
		Y, oxides and hydroxides	-	1E+2	6E-8	2E-10	-	-
90	Thorium-227	W, see ^{226}Th	1E+2	3E-1	1E-10	5E-13	2E-6	2E-5
		Y, see ^{226}Th	-	3E-1	1E-10	5E-13	-	
90	Thorium-228	W, see ^{226}Th	6E+0 Bone Surf	1E-2 Bone Surf	4E-12	-		
			(1E+1)	(2E-2)	-	3E-14	2E-7	2E-6
		Y, see ^{226}Th	-	2E-2	7E-12	2E-14	-	-
90	Thorium-229	W, see ^{226}Th	6E-1 Bone Surf	9E-4 Bone Surf	4E-13	-	-	-
			(1E+0)	(2E-3)	-	3E-15	2E-8	2E-7
		Y, see ^{226}Th		2E-3 Bone Surf	1E-12	-		
				(3E-3)	-	4E-15	-	
90	Thorium-230	W, see ^{226}Th	4E+0 Bone Surf	6E-3 Bone Surf	3E-12	-	-	-
			(9E+0)	(2E-2)	-	2E-14	1E-7	1E-6
		Y, see ^{226}Th	-	2E-2 Bone Surf	6E-12	-		
			-	(2E-2)	-	3E-14	-	
90	Thorium-231	W, see ^{226}Th	4E+3	6E+3	3E-6	9E-9	5E-5	5E-4
		Y, see ^{226}Th	-	6E+3	3E-6	9E-9	-	
90	Thorium-232	W, see ^{226}Th	7E-1 Bone Surf	1E-3 Bone Surf	5E-13	-	-	-
			(2E+0)	(3E-3)	-	4E-15	3E-8	3E-7
		Y, see ^{226}Th	-	3E-3 Bone Surf	1E-12	-	-	-
			-	(4E-3)	-	6E-15	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
90	Thorium-234	W, see ^{226}Th	3E+2 LLI wall	2E+2	8E-8	3E-10	-	-
			(4E+2)	-	-	-	5E-6	5E-5
		Y, see ^{226}Th	-	2E+2	6E-8	2E-10	-	-
91	Protactinium-227[2]	W, all compounds except those given for Y	4E+3	1E+2	5E-8	2E-10	5E-5	5E-4
		Y, oxides and hydroxides	-	1E+2	4E-8	1E-10	-	-
91	Protactinium-228	W, see ^{227}Pa	1E+3	1E+1 Bone Surf	5E-9	-	2E-5	2E-4
			-	(2E+1)	-	3E-11	-	-
		Y, see ^{227}Pa	-	1E+1	5E-9	2E-11	-	-
91	Protactinium-230	W, see ^{227}Pa	6E+2 Bone Surf	5E+0	2E-9	7E-12	-	-
			(9E+2)	-	-	-	1E-5	1E-4
		Y, see ^{227}Pa	-	4E+0	1E-9	5E-12	-	-
91	Protactinium-231	W, see ^{227}Pa	2E-1 Bone Surf	2E-3 Bone Surf	6E-13	-	-	-
			(5E-1)	(4E-3)	-	6E-15	6E-9	6E-8
		Y, see ^{227}Pa		4E-3 Bone Surf	2E-12			
				(6E-3)	-	8E-15	-	
91	Protactinium-232	W, see ^{227}Pa	1E+3	2E+1 Bone Surf	9E-9	-	2E-5	2E-4
				(6E+1)	-	8E-11	-	
		Y, see ^{227}Pa		6E+1 Bone Surf	2E-8	-		
				(7E+1)	-	1E-10	-	
91	Protactinium-233	W, see ^{227}Pa	1E+3 LLI wall	7E+2	3E-7	1E-9	-	
			(2E+3)	-	-	-	2E-5	2E-4
		Y, see ^{227}Pa		6E+2	2E-7	8E-10	-	
91	Protactinium-234	W, see ^{227}Pa	2E+3	8E+3	3E-6	1E-8	3E-5	3E-4

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
		Y, see ^{227}Pa	-	7E+3	3E-6	9E-9	-	
92	Uranium-230	D, UF_6, UO_2F_2, $UO_2(NO_3)_2$	4E+0 Bone Surf	4E-1 Bone Surf	2E-10	-	-	-
			(6E+0)	(6E-1)	-	8E-13	SE-8	8E-7
		W, UO_3, UF_4, UCl_4	-	4E-1	1E-10	5E-13	-	-
		Y, UO_2, U_3O_8	-	3E-1	1E-10	4E-13	-	-
92	Uranium-231	D, see ^{230}U	5E+3 LLI wall	8E+3	3E-6	1E-8	-	-
			(4E+3)	-	-	-	6E-5	6E-4
		W, see ^{230}U	-	6E+3	2E-6	8E-9	-	
		Y, see ^{230}U	-	5E+3	2E-6	6E-9	-	-
92	Uranium-232	D, see ^{230}U	2E+0 Bone Surf	2E-1 Bone Surf	9E-11	-	-	-
			(4E+0)	(4E-1)	-	6E-13	6E-8	6E-7
		W, see ^{230}U	-	4E-1	2E-10	5E-13	-	
		Y, see ^{230}U	-	8E-3	3E-12	1E-14	-	-
92	Uranium-233	D. see ^{230}U	1E+1 Bone Surf	1E+0 Bone Surf	5E-10	-		-
			(2E+1)	(2E+0)	-	3E-12	3E-7	3E-6
		W, see ^{230}U	-	7E-1	3E-10	1E-12		
		Y, see ^{230}U	-	4E-2	2E-11	5E-14		
92	Uranium-234[3]	D, see ^{230}U	1E+1 Bone Surf	1E+0 Bone Surf	5E-10	-		
			(2E+1)	(2E+0)	-	3E-12	3E-7	3E-6
		W. see ^{230}U	-	7E-1	3E-10	1E-12		
		Y, see ^{230}U	-	4E-2	2E-11	5E-14		

			Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
			Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
			Oral Ingestion	Inhalation				Monthly Average Concentration
Atomic No.	Radionuclide	Class	ALI (μCi)	ALI (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	(μCi/ml)
92	Uranium-235[3]	D, see ^{230}U	1E+1 Bone Surf	1E+0 Bone Surf	6E-10	-		
			(2E+1)	(2E+0)	-	3E-12	3E-7	3E-6
		W, see ^{230}U		8E-1	3E-10	1E-12		
		Y, see ^{230}U	-	4E-2	2E-11	6E-14		
92	Uranium-236	D, see ^{230}U	1E+1 Bone Surf	1E+0 Bone Surf	5E-10	-		
			(2E+1)	(2E+0)	-	3E-12	3E-7	3E-6
		W, see ^{230}U	-	8E-1	3E-10	1E-12	-	
		Y, see ^{230}U	-	4E-2	2E-11	6E-14		
92	Uranium-237	D, see ^{230}U	2E+3 LLI wall	3E+3	1E-6	4E-9		
			(2E+3)	-	-	-	3E-5	3E-4
		W, see ^{230}U	-	2E+3	7E-7	2E-9		
		Y, see ^{230}U		2E+3	6E-7	2E-9		
92	Uranium-238[3]	D, see ^{230}U	1E+1 Bone Surf	1E+0 Bone Surf	6E-10	-		
			(2E+1)	(2E+0)	-	3E-12	3E-7	3E-6
		W, see ^{230}U		8E-1	3E-10	1E-12		
		Y, see ^{230}U		4E-2	2E-11	6E-14		
92	Uranium-239[3]	D, see ^{230}U	7E+4	2E+5	8E-5	3E-7	9E-4	9E-3
		W, see ^{230}U		2E+5	7E-5	2E-7		
		Y, see ^{230}U		2E+5	6E-5	2E-7		
92	Uranium-240	D, see ^{230}U	1E+3	4E+3	2E-6	5E-9	2E-5	2E-4
		W, see ^{230}U		3E+3	1E-6	4E-9		
		Y, see ^{230}U		2E+3	1E-6	3E-9		

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Col. 2 Inhalation ALI (μCi)	Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
92	Uranium-natural[3]	D, see ^{230}U	1E+1 Bone Surf	1E+0 Bone Surf	5E-10	-	-	-
			(2E+1)	(2E+0)	-	3E-12	3E-7	3E-6
		W, see ^{230}U	-	8E-1	3E-10	9E-13	-	-
		Y, see ^{230}U	-	5E-2	2E-11	9E-14	-	-
93	Neptunium-232[2]	W, all compounds	1E+5	2E+3 Bone Surf	7E-7	-	2E-3	2E-2
			-	(5E+2)	-	6E-9	-	-
93	Neptunium-233[2]	W, all compounds	8E+5	3E+6	1E-3	4E-6	1E-2	1E-1
93	Neptunium-234	W, all compounds	2E+3	3E+3	1E-6	4E-9	3E-5	3E-4
93	Neptunium-235	W, all compounds	2E+4 LLI wall	8E+2 Bone Surf	3E-7	-	-	-
			(2E+4)	(1E+3)	-	2E-9	3E-4	3E-3
93	Neptunium-236 (1.15E+5y)	W, all compounds	3E+0 Bone Surf	2E-2 Bone Surf	9E-12	-	-	-
			(6E+0)	(5E-5)	-	8E-14	9E-8	9E-7
93	Neptunium-236m (22.531)	W, all compounds	3E+3 Bone Surf	3E+1 Bone Surf	1E-8	-	-	-
			(4E+3)	(7E+1)	-	1E-10	5E-5	5E-4
93	Neptunium-237	W, all compounds	5E-1 Bone Surf	4E-3 Bone Surf	2E-12	-	-	-
			(1E+0)	(1E-2)	-	1E-14	2E-8	2E-7
93	Neptunium-238	W, all compounds	1E+3	6E+1 Bone Surf	3E-8	-	2E-5	2E-4
			-	(2E+2)	-	2E-10	-	-
93	Neptunium-239	W, all compounds	2E+3 LLI wall	2E+3	9E-7	3E-9	-	
			(2E+3)	-			2E-5	2E-4
93	Neptunium-240[2]	W, all compounds	2E+4	8E+4	3E-5	1E-7	3E-4	3E-3

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
94	Plutonium-234	W, all compounds except PuO_2	8E+3	2E+2	9E-8	3E-10	1E-4	1E-3
		Y, PuO_2	-	2E+2	8E-8	3E-10	-	-
94	Plutonium-235[(2)]	W, see ^{234}Pu	9E+5	3E+6	1E-3	4E-6	1E-2	1E-1
		Y, see ^{234}Pu	-	3E+6	1E-3	3E-6	-	-
94	Plutonium-236	W, see ^{234}Pu	2E+0 Bone Surf	2E-2 Bone Surf	8E-12	-	-	-
			(4E+0)	(4E-2)	-	5E-14	6E-8	6E-7
		Y, see ^{234}Pu	-	4E-2	2E-11	6E-14	-	-
94	Plutonium-237	W, see ^{234}Pu	1E+4	3E+3	1E-6	5E-9	2E-4	2E-3
		Y, see ^{234}Pu	-	3E+3	1E-6	4E-9	-	-
94	Plutonium-238	W, see ^{234}Pu	9E-1 Bone Surf	7E-3 Bone Surf	3E-12	-	-	-
			(2E+0)	(1E-2)	-	2E-14	2E-8	2E-7
		Y, see ^{234}Pu	-	2E-2	8E-12	2E-14	-	-
94	Plutonium-239	W, see ^{234}Pu	8E-1 Bone Surf	6E-3 Bone Surf	3E-12	-	-	-
			(1E+0)	(1E-2)	-	2E-14	2E-8	2E-7
		Y, see ^{234}Pu	-	2E-2 Bone Surf	7E-12	-	-	-
			-	(2E-2)	-	2E-14	-	-
94	Plutonium-240	W, see ^{234}Pu	8E-1 Bone Surf	6E-3 Bone Surf	3E-12	-	-	-
			(1E+0)	(1E-2)	-	2E-14	2E-8	2E-7
		Y, see ^{234}Pu	-	2E-2 Bone Surf	7E-12	-	-	-
			-	(2E-2)	-	2E-14	-	-

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Col. 2 Inhalation ALI (μCi)	Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
94	Plutonium-241	W, see ^{234}Pu	4E+1 Bone Surf	3E-1 Bone Surf	1E-10	-	-	
			(7E+1)	(6E-1)	-	8E-13	1E-6	1E-5
		Y, see ^{234}Pu		8E-1 Bone Surf	3E-10	-	-	
				(1E+0)	-	1E-12	-	
94	Plutonium-242	W, see ^{234}Pu	SE-1 Bone Surf	7E-3 Bone Surf	3E-12	-		
			(1E+0)	(1E-2)	-	2E-14	2E-8	2E-7
		Y, see ^{234}Pu	-	2E-2 Bone Surf	7E-12	-	-	
			-	(2E-2)	-	2E-14	-	
94	Plutonium-243	W, see ^{234}Pu	2E+4	4E+4	2E-5	5E-8	2E-4	2E-3
		Y, see ^{234}Pu	-	4E+4	2E-5	5E-8	-	
94	Plutonium-244	W, see ^{234}Pu	8E-1 Bone Surf	7E-3 Bone Surf	3E-12	-	-	
			(2E+0)	(1E-2)	-	2E-14	2E-8	2E-7
		Y, see ^{234}Pu		2E-2 Bone Surf	7E-12	-	-	
				(2E-2)	-	2E-14	-	
94	Plutonium-245	W, see ^{234}Pu	2E+3	5E+3	2E-6	6E-9	3E-5	3E-4
		Y, see ^{234}Pu		4E+3	2E-6	6E-9	-	
94	Plutonium-246	W, see ^{234}Pu	4E+2 LLI wall	3E+2	1E-7	4E-10	-	
			(4E+2)	-	-	-	6E-6	6E-5
		Y, see ^{234}Pu	-	3E+2	1E-7	4E-10	-	
95	Americium-237’’’	W, all compounds	8E+4	3E+5	1E-4	4E-7	1E-3	1E-2
95	Americium-238[2]	W, all compounds	4E+4	3E+3 Bone Surf	1E-6	-	5E-4	5E-3
				(6E+3)	-	9E-9	-	
95	Americium-239	W, all compounds	5E+3	1E+4	5E-6	2E-8	7E-5	7E-4

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
95	Americium-240	W, all compounds	2E+3	3E+3	1E-6	4E-9	3E-5	3E-4
95	Americium-241	W, all compounds	8E-1 Bone Surf	6E-3 Bone Surf	3E-12	-	-	-
			(1E+0)	(1E-2)	-	2E-14	2E-8	2E-7
95	Americium-242m	W, all compounds	8E-1 Bone Surf	6E-3 Bone Surf	3E-12	-	-	-
			(1E+0)	(1E-2)	-	2E-14	2E-8	2E-7
95	Americium-242	W, all compounds	4E+3	8E+1 Bone Surf	4E-8	-	5E-5	5E-4
			-	(9E+1)	-	1E-10	-	-
95	Americium-243	W, all compounds	8E-1 Bone Surf	6E-3 Bone Surf	3E-12	-	-	-
			(1E+0)	(1E-2)	-	2E-14	2E-8	2E-7
95	Americium-244m(2)	W, all compounds	6E+4 St. wall	4E+3 Bone Surf	2E-6	-	-	-
			(8E+4)	(7E+3)	-	1E-8	1E-3	1E-2
95	Americium-244	W, all compounds	3E+3	2E+2 Bone Surf	8E-8	-	4E-5	4E-4
			-	(3E+2)	-	4E-10	-	-
95	Americium-245	W, all compounds	3E+4	8E+4	3E-5	1E-7	4E-4	4E-3
95	Americium-246m(2)	W, all compounds	5E+4 St. wall	2E+5	8E-5	3E-7	-	-
			(6E+4)	-	-	-	8E-4	8E-3
95	Americium-246(2)	W, all compounds	3E+4	1E+5	4E-5	1E-7	4E-4	4E-3
96	Curium-238	W, all compounds	2E+4	1E+3	5E-7	2E-9	2E-4	2E-3
96	Curium-240	W, all compounds	6E+1 Bone Surf	6E-1 Bone Surf	2E-10	-		
			(8E+1)	(6E-1)	-	9E-13	1E-6	1E-5
96	Curium-241	W, all compounds	1E+3	3E+1 Bone Surf	1E-8		2E-5	2E-4
				(4E+1)	-	5E-11	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
96	Curium-242	W, all compounds	3E+1 Bone Surf	3E-1 Bone Surf	1E-10		-	-
			(5E+1)	(3E-1)	-	4E-13	7E-7	7E-6
96	Curium-243	W, all compounds	1E+0 Bone Surf	9E-3 Bone Surf	4E-12	-	-	-
			(2E+0)	(2E-2)	-	2E-14	3E-8	3E-7
96	Curium-244	W, all compounds	1E+0 Bone Surf	1E-2 Bone Surf	5E-12			-
			(3E+0)	(2E-2)	-	3E-14	3E-8	3E-7
96	Curium-245	W, all compounds	7E-1 Bone Surf	6E-3 Bone Surf	3E-12	-		
			(1E+0)	(1E-2)	-	2E-14	2E-8	2E-7
96	Curium-246	W, all compounds	7E-1 Bone Surf	6E-3 Bone Surf	3E-12	-	-	-
			(1E+0)	(1E-2)	-	2E-14	2E-8	2E-7
96	Curium-247	W, all compounds	8E-1 Bone Surf	6E-3 Bone Surf	3E-12		-	-
			(1E+0)	(1E-2)	-	2E-14	2E-8	2E-7
96	Curium-248	W, all compounds	2E-1 Bone Surf	2E-3 Bone Surf	7E-13			
			(4E-1)	(3E-3)	-	4E-15	5E-9	5E-8
96	Zufium-249'''	W, all compounds	5E+4	2E+4 Bone Surf	7E-6		7E-4	7E-3
			-	(3E+4)	-	4E-8	-	-
96	Curium-250	W, all compounds	4E-2 Bone Surf	3E-4 Bone Surf	1E-13		-	-
			(6E-2)	(5E-4)	-	8E-16	9E-10	9E-9
97	Berkelium-245	W, all compounds	2E+3	1E+3	5E-7	2E-9	3E-5	3E-4
97	Berkelium-246	W, all compounds	3E+3	3E+3	1E-6	4E-9	4E-5	4E-4
97	Berkelium-247	W, all compounds	5E-1 Bone Surf	4E-3 Bone Surf	2E-12		-	-
			(1E+0)	(9E-3)	-	1E-14	2E-8	2E-7

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
97	Berkelium-249	W, all compounds	2E+2 Bone Surf	2E+0 Bone Surf	7E-10	-	-	-
			(5E+2)	(4E+0)	-	5E-12	6E-6	6E-5
97	Berkelium-250	W, all compounds	9E+3	3E+2 Bone Surf	1E-7	-	1E-4	1E-3
			-	(7E+2)	-	1E-9	-	-
98	Californium-244[2]	W, all compounds except those given for Y	3E+4 St. wall	6E+2	2E-7	8E-10	-	-
			(3E+4)	-	-	-	4E-4	4E-3
		Y, oxides and hydroxides	-	6E+2	2E-7	8E-10	-	-
98	Californium-246	W, see ^{244}Cf	4E+2	9E+0	4E-9	1E-11	5E-6	5E-5
		Y, see ^{244}Cf	-	9E+0	4E-9	1E-11	-	-
98	Californium-248	W, see ^{244}Cf	8E+0 Bone Surf	6E-2 Bone Surf	3E-11	-	-	-
			(2E+1)	(1E-1)	-	2E-13	2E-7	2E-6
		Y, see ^{244}Cf	-	1E-1	4E-11	1E-13	-	-
98	Californium-249	W, see ^{244}Cf	5E-1 Bone Surf	4E-3 Bone Surf	2E-12	-	-	-
			(1E+0)	(9E-3)	-	1E-14	2E-8	2E-7
		Y, see ^{244}Cf	-	1E-2 Bone Surf	4E-12	-	-	-
			-	(1E-2)	-	2E-14	-	-
98	Californium-250	W, see ^{244}Cf	1E+0 Bone Surf	9E-3 Bone Surf	4E-12	-	-	-
			(2E+0)	(2E-2)	-	3E-14	3E-8	3E-7
		Y, see ^{244}Cf	-	3E-2	1E-11	4E-14	-	-
98	Californium-251	W, see ^{244}Cf	5E-1 Bone Surf	4E-3 Bone Surf	2E-12	-		
			(1E+0)	(9E-3)	-	1E-14	2E-8	2E-7
		Y, see ^{244}Cf		1E-2 Bone Surf	4E-12	-		
				(1E-2)	-	2E-14	-	

Atomic No.	Radionuclide	Class	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
98	Californium-252	W, see ^{244}Cf	2E+0 Bone Surf	2E-2 Bone Surf	8E-12	-	-	-
			(5E+0)	(4E-2)	-	5E-14	7E-8	7E-7
		Y, see ^{244}Cf	-	3E-2	1E-11	5E-14	-	-
98	Californium-253	W, see ^{244}Cf	2E+2 Bone Surf	2E+0	8E-10	3E-12	-	-
			(4E+2)	-	-	-	5E-6	5E-5
		Y, see ^{244}Cf	-	2E+0	7E-10	2E-12	-	-
98	Californium-254	W, see ^{244}Cf	2E+0	2E-2	9E-12	3E-14	3E-8	3E-7
		Y, see ^{244}Cf	-	2E-2	7E-12	2E-14	-	-
99	Einsteinium-250	W, all compounds	4E+4	5E+2 Bone Surf	2E-7	-	6E-4	6E-3
			-	(1E+3)	-	2E-9	-	-
99	Einsteinium-251	W, all compounds	7E+3	9E+2 Bone Surf	4E-7	-	1E-4	1E-3
			-	(1E+3)	-	2E-9	-	-
99	Einsteinium-253	W, all compounds	2E+2	1E+0	6E-10	2E-12	2E-6	2E-5
99	Einsteinium-254m	W. all compounds	3E+2 LLI wall	1E+1	4E-9	1E-11	-	-
			(3E+2)	-	-	-	4E-6	4E-5
99	Einsteinium-254	W, all compounds	8E+0 Bone Surf	7E-2 Bone Surf	3E-11	-	-	-
			(2E+1)	(1E-1)	-	2E-13	2E-7	2E-6
100	Fermium-252	W, all compounds	5E+2	1E+1	5E-9	2E-11	6E-6	6E-5
100	Fermium-253	W, all compounds	1E+3	1E+1	4E-9	1E-11	1E-5	1E-4
100	Fermium-254	W. all compounds	3E+3	9E+1	4E-8	1E-10	4E-5	4E-4
100	Fermium-255	W, all compounds	5E+2	2E+1	9E-9	3E-11	7E-6	7E-5
100	Fermium-257	W, all compounds	2E+1 Bone Surf	2E-1 Bone Surf	7E-11	-		
			(4E+1)	(2E-1)	-	3E-13	5E-7	5E-6

Atomic No.	Radionuclide	Class	Table 1 Occupational Values, Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations, Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers, Monthly Average Concentration (μCi/ml)
101	Mendelevium-257	W, all compounds	7E+3	8E+1 Bone Surf	4E-8		1E-4	1E-3
				(9E+1)	-	1E-10		
101	Mendelevium-258	W, all compounds	3E+1 Bone Surf	2E-1 Bone Surf	1E-10			
			(5E+1)	(3E-1)		5E-13	6E-7	6E-6
	Any single radionuclide not listed above with decay mode other than alpha emission or spontaneous fission and with radioactive half-life less than 2 hours	Submersion"		2E+2	IE-7	1E-9		-
	Any single radionuclide not listed above with decay mode other than alpha emission or spontaneous fission and with radioactive half-life greater than 2 hours			2E-1	1E-10	1E-12	1E-8	I E-7
	Any single radionuclide not listed above that decays by alpha emission or spontaneous fission, or any mixture for which either the identity or the concentration of any radionuclide in the mixture is not known			4E-4	2E-13	1E-15	2E-9	2E-8

FOOTNOTES:

(1) "Submersion" means that values given are for submission in a hemispherical semi-infinite cloud of airborne material.

(2) These radionuclides have radiological half-lives of less than 2 hours. The total effective dose equivalent received during operations with these radionuclides might include a significant contribution from external exposure. The DAC values for all radionuclides, other than those designated Class "Submersion," are based upon the committed effective dose equivalent due to the intake of the radionuclide into the body and do NOT include potentially significant contributions to dose equivalent from external exposures. The licensee may substitute 1E-7μCi/ml for the listed DAC to account for the submersion dose prospectively, but should use individual monitoring devices or other radiation measuring instruments that measure external exposure to demonstrate compliance with the limits. (See section 20.1203.)

(3) For soluble mixtures of U-238, U-234. and U-235 in air, chemical toxicity may be the limiting factor (see section 20.1201(e)). If the percent by weight (enrichment) of U-235 is not greater than 5, the concentration value for a 40-hour workweek is 0.2 milligrams uranium per cubic meter of air average. For any enrichment, the product of the average concentration and time of exposure during a 40-hour workweek shall not exceed 8E-3 (SA) μCi-hr/ml, where SA is the specific activity of the uranium inhaled. The specific activity for natural uranium is 6.77E-7 curies per gram U. The specific activity for other mixtures of U-236, U-235, and U-234, if not known, shall be:

SA = 3.6E-7 curies/gram U U-depleted

SA = [0.4 + 0.38 (enrichment) + 0.0034 $(\text{enrichment})^2$] E-6, enrichment ≥ 0.72

where enrichment is the percentage by weight of U-235, expressed in percent.

NOTE:

1. If the identity of each radionuclide in a mixture is known but the concentration of one or more of the radionuclides in the mixture is not known, the DAC for the mixture shall be the most restrictive DAC of any radionuclide in the mixture.
2. If the identity of each radionuclide in the mixture is not known, but it is known that certain radionuclides specified in the appendix are not present in the mixture, the inhalation ALI, DAC, and effluent and sewage concentration for the mixture are the lowest values specified in this appendix for any radionuclide that is not known to be absent from the mixture; or

	Table 1 Occupational Values			Table 2 Effluent Concentrations		Table 3 Releases to Sewers
	Col. 1	Col. 2	Col. 3	Col. 1	Col. 2	
	Oral Ingestion	Inhalation				Monthly Average Concentration (μCi/ml)
Radionuclide	AL1 (μCi)	AL1 (μCi)	DAC (μCi/ml)	Air (μCi/ml)	Water (μCi/ml)	
If it is known that Ac-227-D and Cm-250-W are not present	-	7E-4	3E-13	-	-	
If, in addition, it is known that Ac-227-W, Y, Th-229-W, Y, Th-230-W, Th-232-W, Y, Pa-231-W, Y, Np-237-W, Pu-239-W, Pu-240-W, Pu-242-W. Am-241-W, Am-242m-W, Am-243-W. Cm-245-W, Cm-246-W, Cm-247-W, Cm-248-W, Bk-247-W. Cf-249-W, and Cf-25 1-W are not present	-	7E-3	3E-12	-	-	
If, in addition, it is known that Sm-146-W, Sm-147-W, Cd-148-D, W, Gd-152-D, W, Th-228-W, Y, Th-230-Y. U-232-Y. U-233-Y, U-234-Y, U-235-Y, U-236-Y. U-238-Y, Np-236-W, Pu-236-W, Y, Pu-238-W, Y, Pu-239-Y, Pu-240-Y, Pu-242-Y, Pu-244-W, Y, Cm-243-W, Cm-244-W, Cf-248-W, Cf-249-Y, Cf-250-W, Y, Cf-251-Y. Cf-252-W, Y, and Cf-254-W, Y are not present	-	7E-2	3E-11	-	-	-
If, in addition, it is known that Pb-210-D, Bi-210m-W, Po-210-D, W, Ra-223-W, Ra-225-W, Ra-226-W, Ac-225-D. W, Y, Th-227-W, Y, U-230-D. W, Y, U-232-D, W, Pu-241 -W, Cm-240-W, Cm-242-W, Cf-248-Y, Es-254-W, Fm-257-W, and Md-258-W are not present	-	7E-1	3E-10	-	-	
If, in addition, it is known that Si-32-Y. Ti-44-Y, Fe-60-D, Sr-90-Y. Zr-93-D. Cd-l 13m-D, Cd-l 13-D, In-l IS-D, W, La-138-D, Lu-176-W, Hf-178m-D, W, Hf-182-D, W, Bi-210m-D, Ra-224-W, Ra-228-W, Ac-226-D. W, Y, Pa-230-W, Y, U-233-D, W; U-234-D, W, U-235-D. W, U-236-D, W, U-238-D, W, Pu-241-Y, Bk-249-W, Cf-253-W, Y, and Es-253-W are not present	-	7E+0	3E-9	-	-	

Radionuclide	Table 1 Occupational Values Col. 1 Oral Ingestion ALI (μCi)	Table 1 Col. 2 Inhalation ALI (μCi)	Table 1 Col. 3 Inhalation DAC (μCi/ml)	Table 2 Effluent Concentrations Col. 1 Air (μCi/ml)	Table 2 Col. 2 Water (μCi/ml)	Table 3 Releases to Sewers Monthly Average Concentration (μCi/ml)
If it is known that Ac-227-D, W, Y, Th-229-W, Y, Th-232-W, Y, Pa-231-W, Y, Cm-248-W, and Cm-250-W are not present	-	-	-	1E-14		
If, in addition, it is known that Sm-146-W, Gd-148-D, W, Gd-152-D, Th-228-W, Y, Th-230-W, Y, U-232-Y, U-233-Y, U-234-Y, U-235-Y, U-236-Y, U-238-Y, U-Nat-Y, Np-236-W, Np-237-W, Pu-236-W, Y, Pu-238-W, Y, Pu-239-W, Y, Pu-240-W, Y, Pu-242-W, Y, Pu-244-W, Y, Am-241-W, Am-242m-W, Am-243-W, Cm-243-W, Cm-244-W, Cm-245-W, Cm-246-W, Cm-247-W, Bk-247-W, Cf-249-W, Y, Cf-250-W, Y, Cf-251-W, Y, Cf-252-W, Y, and Cf-254-W, Y are not present	-	-	-	1E-13	-	-
If, in addition, it is known that Sm-147-W, Gd-152-W, Pb-210-D, Bi-210m-W, Po-210-D, W, Ra-223-W, Ra-225-W, Ra-226-W, Ac-225-D, W, Y, Th-227-W, Y, U-230-D, W, Y, U-232-D, W, U-Nat-W, Pu-241-W, Cm-240-W, Cm-242-W, Cf-248-W, Y, Es-254-W, Fm-257-W, and Md-258-W are not present	-	-	-	1E-12		
If, in addition, it is known that Fe-60, Sr-90, Cd-113m, Cd-113, In-115, I-129, Cs-134, Sm-145, Sm-147, Gd-148, Gd-152, Hg-194 (organic), Bi-210m, Ra-223, Ra-224, Ra-225, Ac-225, Th-228, Th-230, U-233, U-234, U-235, U-236, U-238, U-Nat, Cm-242, Cf-248, Es-254, Fm-257, and Md-258 are not present	-	-	-		1E-6	1E-5

3. If a mixture of radionuclides consists of uranium and its daughters in ore dust (10μm AMAD particle distribution assumed) prior to chemical separation of the uranium from the ore, the following values may be used for the DAC of the mixture; 6E-11μCi of gross alpha activity from uranium-238, uranium-234, thorium-230, and radium-226 per milliliter of air; 3E-11 μCi of natural uranium per milliliter of air; or 45 micrograms of natural uranium per cubic meter of air.

4. If the identity and concentration of each radionuclide in a mixture are known, the limiting values should be derived as follows: determine, for each radionuclide in the mixture, the ratio between the concentration present in the mixture and the concentration otherwise established in Appendix B for the specific radionuclide when not in a mixture. The sum of such ratios for all of the radionuclides in the mixture may not exceed "1" (i.e., "unity").

   Example: If radionuclides "A," "B," and "C" are present in concentrations C_A, C_B, and C_C, and if the applicable DACs are DAC_A, DAC_B, and DAC_C, respectively, then the concentrations shall be limited so that the following relationship exists:

$$\frac{C_A}{DAC_A} + \frac{C_B}{DAC_B} + \frac{C_C}{DAC_C} < 1$$

Discussion:

The data tabulated in this appendix is intended to be used to show compliance with a number of sections of Part 20 that refer to one or more of the tables in the appendix. For example, monitoring of workers for intakes is required when annual intakes may exceed specified fractions of Columns (2) and (3) of Table (1). Licensees may show compliance, in part, with public dose limits for doses resulting from effluents by referring to the concentrations listed in Columns (1) and (2) of Table (2). Compliance with sewer release limits is shown, in part, by using the values in Table (3). In addition, certain requirements such as area posting, respiratory protection, and incident reporting use Appendix B values as triggers for these actions. Table (1) is based on occupational dose limits, Table (2) on dose limits to members of the public, and Table (3) is a special case of exposure to members of the public.

The values tabulated in Appendix B are all secondary limits or derived quantities, and each column in the appendix is based on a primary limit, which in this case is a dose. A secondary limit, such as the DAC, is derived from a primary limit. The difference is that the primary limit is absolute in that it is not to be exceeded in any routine situation. The secondary limit is not absolute in the sense that it is applicable only if certain conditions are met. It is not valid as a limit if these conditions are not met. For example, the ALI is a limit if there are no external exposures during the monitoring year and the only source of exposure is internal. If there are external exposures, limiting annual intakes to an AL1 will lead to exceeding the primary dose limit, and hence a violation of NRC requirements. The ALI is also not a limit if the ingested or inhaled radioactive material contains more than one type of radionuclide. In such cases, the ALIs of each of the components must be adjusted to take account of the presence of the other components.

A derived quantity, such as the DAC, is not a limit at all, and may be exceeded at any time provided certain restrictions apply. The DAC is tabulated for convenience and because it is an easily measured quantity. It is easily calculated from the ALI by assuming a suitable breathing rate and exposure time. Part 20 does not limit airborne concentrations at any given time to values below the DAC, and requirements in Part 20 that are specified in terms of airborne concentrations generally use only time-averaged concentrations and not instantaneous values.

In this appendix, the daughter products of the radionuclides listed were not included in the intake when calculating the tabulated values of ALI and DAC for the parent. However, the effects of the daughters that are produced in the body after intake of the parent are included in the calculations. For example, uranium decays in a long decay chain that includes many radioactive daughter products. When considering the inhalation of uranium, the calculations of ALIs and DACs for the tables assume that only the parent uranium isotope is inhaled, and no daughters are considered with the uranium inhalation. The daughters produced in the body after the uranium is taken into the body are included in estimating the dose resulting from the intake. To properly account for the dose from a parent that produces one or more daughters, the parent and each of the daughters must be treated as separately inhaled or ingested radionuclides, and the dose from each added to produce a total dose. The parent and its daughters that may be in the inhaled air or

in the ingested material are considered as a mixture of radionuclides and not as members of one chain.

One exception to this rule is the case in which the half-life of the daughter is very short (usually less than about 20 minutes) and much shorter than that of the parent. In this case, the tabulated values assume that the daughter is in secular equilibrium with the parent and that both are always inhaled or ingested together. The daughter in this case need not be considered separately in determining the ALI from the table, because it has been already included in calculating the ALI for the parent. In such situations, you will not find data in the table pertaining to short-lived daughters of tabulated radionuclides. In such cases, assume that the data for the missing daughter has been included in the data for the parent.

Column (1) in the appendix is the ALI for ingestion by occupationally exposed workers. Ingestion means taking in the material by mouth via food or drink or as solid or liquid contamination in the workplace. The values are based on an annual dose limit of 5 rem effective dose equivalent (called the stochastic dose limit) or 50 rem organ dose equivalent (called the non-stochastic, or deterministic, dose limit), whichever is more limiting. Internal dose models described in ICRP Publication 30 were used to calculate the effective and organ doses that would result from intake of unit activity of each of the radionuclides listed in the table. The intakes that would lead to an effective dose of 5 rem or an organ dose of 50 rem are then calculated. The highest intake that does not result in exceeding any organ limit or the effective dose equivalent limit is then selected as the AL1 and tabulated in Column (1). If the ALI is based on an organ dose, the organ is specified under the tabulated ALI, and the ALI that would result in an effective dose equivalent of 5 rem is listed in parentheses under that organ name. The stochastic ALI is specified in parentheses because it is sometimes needed to show compliance when several radionuclides are present in the ingested material. If the ALI is based on the effective dose equivalent, then only that value is listed, with no other information.

Column (2) in Table (1) is calculated in the same manner as that used for Column (1), except that the intake is by inhalation of airborne material rather than ingestion. The methods of calculation are the same, but the dosimetric models are those for inhalation rather than for ingestion. In addition, inhaled material is classified into one of three classes, called D, W, or Y, depending on how rapidly the material is cleared from the lungs after it is inhaled. Class D is cleared most rapidly, within a matter of days, and Class Y is cleared most slowly, within months or years. Class W is intermediate. The same radionuclide may exist in one or more classes depending on its chemical and physical characteristics. For example, uranium as a fluoride is a Class D material, but some of its oxide forms are Class W, and other oxide forms are Class Y. Licensees should make a concerted effort to accurately classify the airborne material present at their sites because such classification will determine the ALI and the dose received by a worker following an intake.

The first step in classification is to know the chemical form of the airborne radioactive material. With that knowledge, the licensee may refer to the listing in Appendix B, which specifies the classification of the most frequently encountered compounds of each radionuclide. If the specific

compound is not listed, then other references may be used, such as the tabulations by the ICRP. See Reference {1} listed below.

Note also that the values of the ALI are based on the assumption that the airborne radioactive material is in the form of particles with an activity median aerodynamic diameter of 1 micrometer, or micron. If the median particle size on site is known and is substantially different from 1 micron, for example 5 microns, then the ALIs may be adjusted accordingly, but only after obtaining approval from NRC.' The method of adjustment is described in References {1} and {2}, listed below. If the particle size is not known, then 1 micron is assumed.

The values in Column (3), Table (1), the derived air concentrations (DACs), are calculated directly from Column (2) by dividing the respective ALIs by the breathing rate of a standard person (1.2 m^3/hr) and the number of working hours per year, taken to be 2,000 hr/yr. Exceptions to this method are those airborne radionuclides that pose an external rather than an internal hazard, and for which ALIs are not given, such as the isotopes of xenon and krypton. In such cases, the DACs are calculated directly from the external doses, assuming immersion in a semi-infinite cloud of the gas.

The values in Table (2) differ from those in Table (1) in two major respects: they do not include values that are based on non-stochastic radiation effects, because the public dose limits are so low that such effects are no longer of concern; and they are based on a stochastic dose limit of 100 mrem/yr effective dose equivalent rather than 5 rem/yr. The concentrations in Column (1) of Table (2) were obtained by dividing the stochastic ALIs in Table (1) by the breathing rate of 2,400 m^3/yr, then dividing by 3 to take into account the fact that members of the public breathe the air 24 hours per day all year, rather than 8 hr/day during work days, as is assumed for occupational exposure, and also to adjust for differences in inhalation rates between persons at work and members of the public. The result is then divided by 50 to adjust the values from a dose limit of 5,000 mrem/yr to 100 mrem/yr. Because the occupational ALIs were calculated for healthy adult workers, but members of the public include groups that may be of varying health conditions as well as children, the results are again divided by a safety factor of 2 to allow for this effect.

In the case of radionuclides that pose an external hazard, the concentrations in Column (1) of Table (2) were obtained by adjusting the occupational DACs in Table (1) for the difference in dose limits, that is, by dividing by a factor of 50, and then adjusting for differences in exposure duration from 8 hours per day during work days to 24 hours per day every day.

The concentrations for liquid effluents in Column (2) of Table (2) were obtained by using the most restrictive value in Column (1) of Table (1) and then adjusting it in the same manner as that used to adjust the air values.

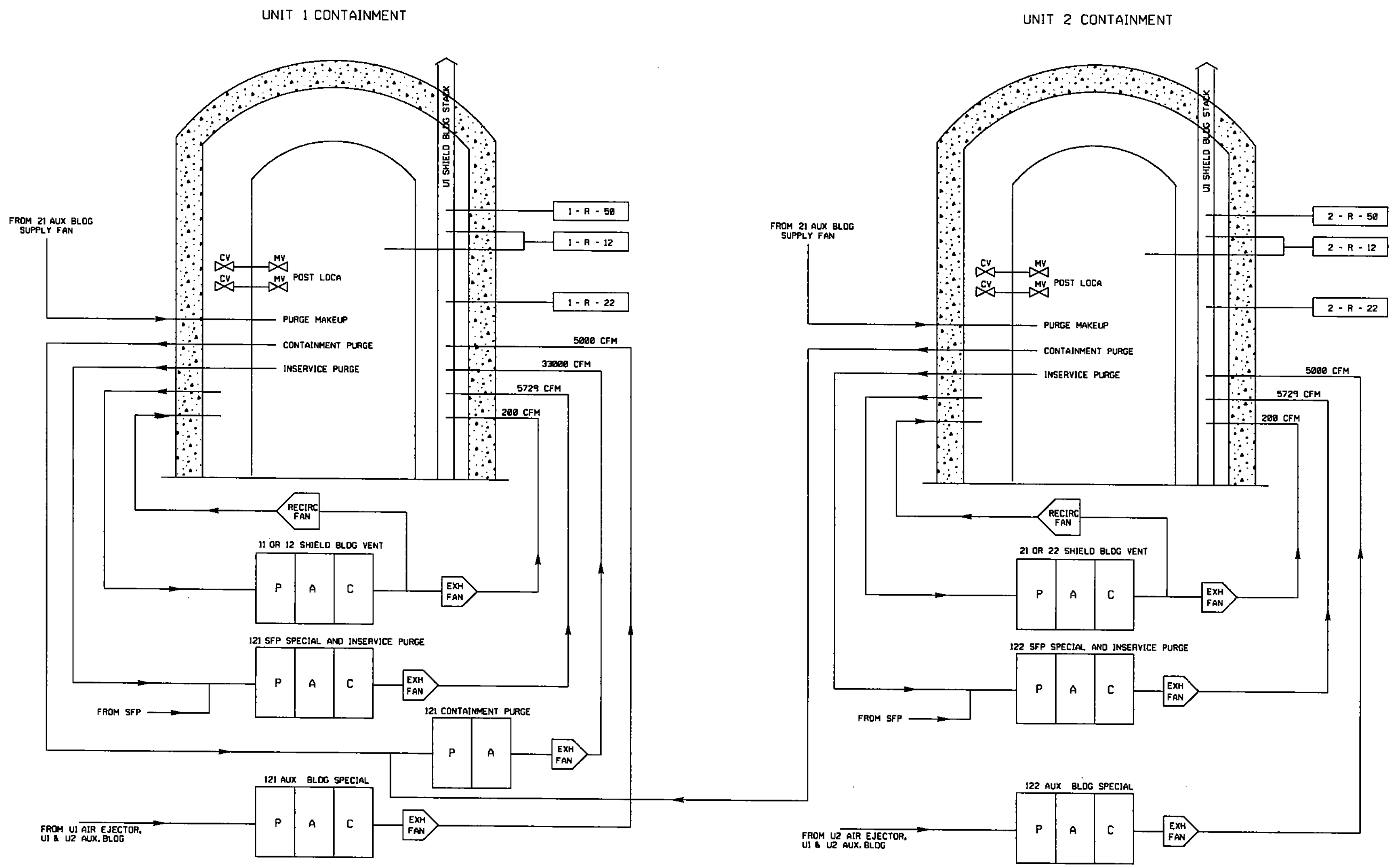
The monthly average concentrations in Table (3) were obtained by assuming that a person obtains all of his water from the licensee's sewer outfall, and then calculating the average concentration that would result in an annual ingestion dose of 500 mrem. Averaging the

concentrations over a month rather than a year avoids excessive short-term peak discharges by seasonal discharges.

References:

1. International Commission on Radiological Protection, Publication 30 and addenda, Pergamon Press, Fairview Park, Elmsford, NY 10523.
2. NRC Regulatory Guide 8.9, Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program.

SHIELD BUILDING STACK EFFLUENT RELEASE FLOW PATHS



TAB 8 Rev. 16

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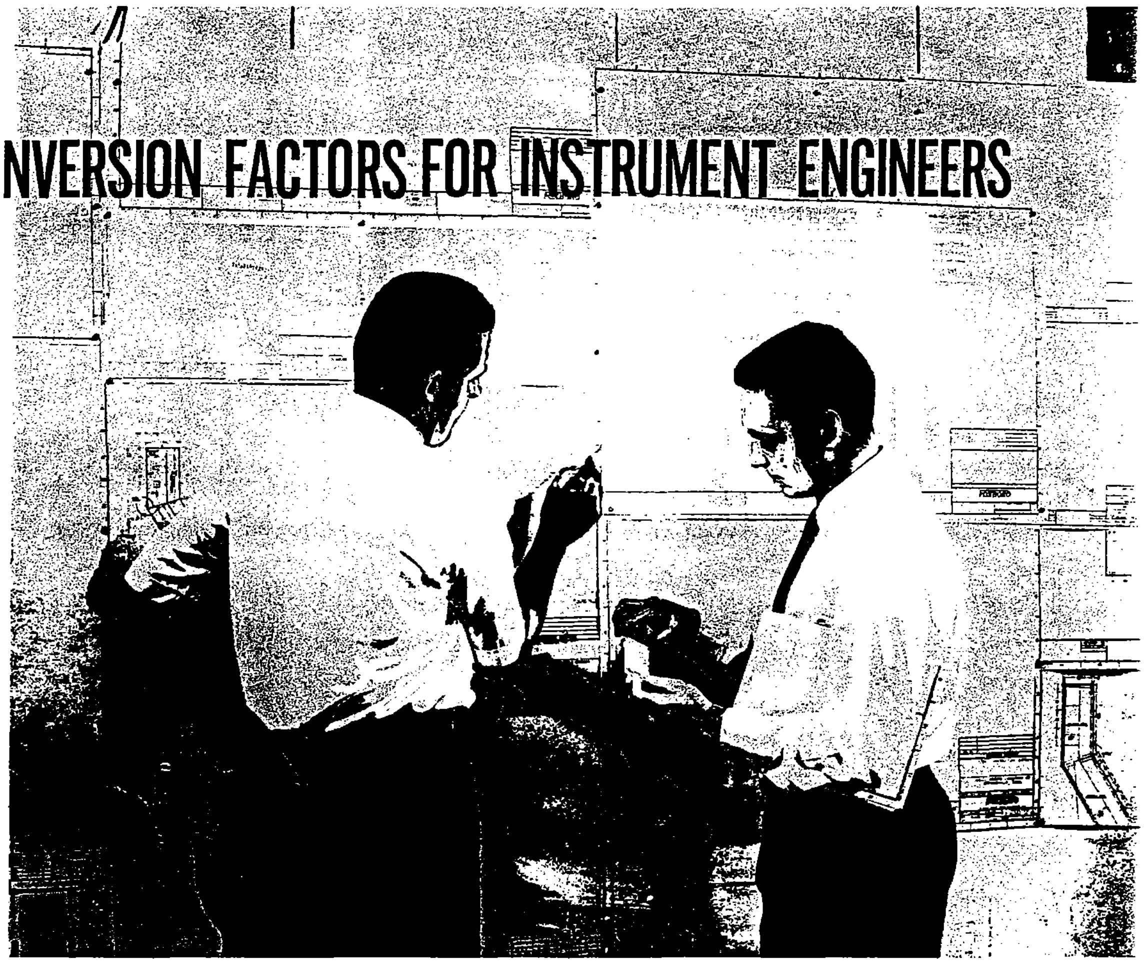
UNITS AND CONVERSION FACTORS FOR INSTRUMENT ENGINEERS



FOXBORO
REGISTERED TRADEMARK

TAB 9, Rev. 1

INTERNATIONAL SYSTEM OF UNITS

CONVERSION FACTORS (ACCURATE TO ± 0.1%)

The **Activity** of a radionuclide is expressed in becquerels, (Bq), equal to one disintegration per second.

	ACTIVITY			ACTIVITY	
1 terabecquerel	= 1 TBq	= 27 curies	1 kilocurie	=1 kCi	= 37 terabecquerels
1 gigabecuerel	= 1 GBq	= 27 millicuries	1 curie	=1Ci	= 37 gigabecquerels
1 megabecquerel	= 1 MBq	= 27 microcuries	1 millicurie	=1mCi	= 37 megabecquerels
1 kilobecquerel	= 1 kBq	= 27 nanocuries	1 microcurie	=1 µCi	= 37 kilobecquerels
1 becquerel	= 1 Bq	= 27 picocuries	1 nanocurie	=1 nCi	= 37 becquerels

The **Dose Equivalent** is expressed in the sievert, (Sv) equal to 1 biologically weighted joule per kg.

	DOSE EQUIVALENT			DOSE EQUIVALENT	
1 sievert	= 1 Sv	= 100 rem	1 kilorem	= 1 krem	= 10 sieverts
1 millisievert	= 1 mSv	= 100 millirem	1 rem	= 1 rem	= 10 millisieverts
1 microsievert	= 1 µSv	= 100 microrem	1 millirem	= 1 mrem	= 10 microsieverts
1 nanosievert	= 1 nSv	= 100 nanorem	1 microrem	= 1 µrem	= 10 nanosieverts

The **Absorbed Dose** is measured by the gray, (Gy), equal to 1 joule of energy absorbed per kilogram.

	ABSORBED DOSE			ABSORBED DOSE	
1 kilogray	= 1 kGy	= 100 krad	1 kilorad	= 1 krad	= 10 grays
1 gray	= 1 Gy	= 100 rad	1 rad	= 1 rad	= 10 milligrays
1 milligray	= 1 mGy	= 100 millirad	1 millirad	= 1 mrad	= 10 micrograys
1 microgray	= 1 µGy	= 100 nanorad	1 microrad	= 1 µrad	= 10 nanograys

The **Effective Dose Equivalent** for internal or partial body exposure, measured in sieverts, equals the sum of organ dose equivalents times weighting factors, $\omega\tau$ = 0.25 (gonads), 0.15 (breast), 0.12 (marrow), 0.12 (lung) 0.03 (thyroid), 0.03 (bone surf), and 0.06 (each of next 5 remainder organs.

The **Committed Dose Equivalent** is the dose equivalent, in sieverts, received over 50 years after an internal uptake.

UNITS AND CONVERSION FACTORS

These conversion factors are among those found most useful to instrument engineers. They are accurate to the four or five significant figures given and higher accuracy is almost never warranted in industrial instrumentation. However, high accuracy is sometimes required, and there are times, for example, when the variation in gravity at different locations and the distinction between weight and mass should not be ignored. The standard intensity of gravity at mean sea level, latitude 45°, is 980.7 gals (for Galileo). The actual intensity at the Bureau of Standards is 980.0, while it is 978.2 at the Canal Zone, 982.2 in northern Alaska, about 167.0 on the surface of the moon, and zero in the "weightless" condition of free fall or in artificial satellite orbit. The range between Alaska and the Canal Zone, about ±0.2 percent, is twice the accuracy limit of ±0.1 percent for the Type CXX Strain-Gauge-Actuated Load Cells. The conversion factor that grams x 980.7 = dynes must be used with care, as it applies only under standard conditions, such as exist near Portland, Oregon. At other locations the correct conversion factor is equal to the local intensity of gravity.

The International System of Units (designated MKSA for meter, kilogram, second, and ampere, or SI, for System International d' Unites) has been adopted by France as the only legal system of units, and by the National Bureau of Standards in this country. The kilogram is the unit of mass. The newton is the unit of force, the force required to give a mass of one kilogram an acceleration of one meter per second per second. (In contrast to the newton is the weight of a kilogram mass under standard gravity, a unit now called the kilopond, and formerly referred to as a "kilogram (weight)." In the cgs system, the unit of force is the dyne, so that the standard gravity weight force of a one-gram mass is 980.7 dynes.) The unit of pressure is the pascal, a pressure of one newton per square meter, or the bar, equal to 100,000 pascals.

Below is a set of prefixes, some of which are in common use, as in megatons, kilowatts, millivolts and microinches, to indicate decimal multiples and submultiples of units. These prefixes have been internationally accepted:

10^{12}	tera	10	deka	10^{-6}	micro
10^{9}	giga	10^{-1}	deci	10^{-9}	nano
10^{6}	mega	10^{-2}	centi	10^{-12}	pico
10^{3}	kilo	10^{-3}	milli	10^{-15}	femto
10^{2}	hecto			10^{-18}	atto

Atmospheres—atm (Standard sea-level pressure)

x 1.01325	= Bars
x 76.0	= Centimeters of mercury, at 0 C
x 29.92	= Inches of mercury, at 0 C
x 33.96	= Feet of water, at 68 F
x 1.0332	= Kilograms (kiloponds) per square centimeter
x 101325	= Pascals
x 14.696	= Pounds per square inch
x 1.0581	= Tons per square foot
x 760	= Torr

Barrels, Liquid—bbl

x 31.5	= Gallons (liquid)

Barrels, Oil—bbl

x 42	= Gallons (oil)

Bars

x 33.52	= Feet of water, at 68 F
x 29.53	= Inches of mercury, at 0 C
x 1.0197	= Kilograms (kiloponds) per square centimeter
x 100000	= Pascals
x 14.504	= Pounds per square inch
x 0.98692	= Standard atmospheres
x 1.0443	= Tons per square foot
x 750.06	= Torr

British Thermal Units—Btu or B

x 778	= Foot-pounds
x 0.252	= Kilogram-calories
x 107.6	= Kilogram-meters
x 3.929×10^{-4}	= Horsepower-hours
x 2.930×10^{-4}	= Kilowatt-hours

British Thermal Units Per Minute—Btu per min

x 12.96	= Foot-pounds per second
x 0.02357	= Horsepower
x 0.01758	= Kilowatts
x 17.58	= Watts

Centares

x 1	= Square meters

Centimeters—cm

x 0.3937	= Inches

Centimeters of Mercury, at 0 C

x 0.013332	= Bars
x 0.4468	= Feet of water, at 68 F
x 5.362	= Inches of water, at 68 F
x 0.013595	= Kilograms (kiloponds) per square centimeter
x 27.85	= Pounds per square foot
x 0.19337	= Pounds per square inch
x 0.013158	= Standard atmospheres
x 10	= Torr

Centimeters Per Second—cm per sec

x 1.9685	= Feet per minute
x 0.03281	= Feet per second
x 0.03600	= Kilometers per hour
x 0.6000	= Meters per minute
x 0.02237	= Miles per hour
x 3.728×10^{-4}	= Miles per minute

Cubic Centimeters—cu cm *or* cm^3

x 3.5315×10^{-5}	= Cubic feet
x 6.1024×10^{-2}	= Cubic inches
x 1.3080×10^{-6}	= Cubic yards
x 2.642×10^{-4}	= Gallons
x 2.200×10^{-4}	= Gallons (Imperial)
x 1.0000×10^{-3}	= Liters

Cubic Feet—cu ft

x 2.832×10^{4}	= Cubic centimeters
x 1728	= Cubic inches
x 0.02832	= Cubic meters
x 0.03704	= Cubic yards
x 7.481	= Gallons
x 6.229	= Gallons (Imperial)
x 28.32	= Liters

Cubic Feet Per Minute—cfm

x 472.0	= Cubic centimeters per second
x 0.1247	= Gallons per second
x 0.4720	= Liters per second
x 62.32	= Pounds of water per minute, at 68 F

Cubic Feet Per Second—cfs

x 0.6463	= Million gallons per day
x 448.8	= Gallons per minute

Cubic Inches—cu in

x 16.387	= Cubic centimeters
x 5.787×10^{-4}	= Cubic feet
x 1.6387×10^{-5}	= Cubic meters
x 2.143×10^{-5}	= Cubic yards
x 4.329×10^{-3}	= Gallons
x 3.605×10^{-3}	= Gallons (Imperial)
x 1.6387×10^{-2}	= Liters

Cubic Meters—cu m *or* m^3

x 35.31	= Cubic feet
x 61.024	= Cubic inches
x 1.308	= Cubic yards
x 264.2	= Gallons
x 220.0	= Gallons (Imperial)
x 1000.0	= Liters

Cubic Yards—cu yd

x 7.646×10^{5}	= Cubic centimeters
x 27	= Cubic feet
x 46,656	= Cubic inches
x 0.7646	= Cubic meters
x 201.97	= Gallons
x 168.17	= Gallons (Imperial)
x 764.6	= Liters

Degrees, Angular—deg *or* °

x 60	= Minutes
x 0.017453	= Radians
x 3600	= Seconds

Degrees Per Second, Angular—deg per sec

x 0.017453	= Radians per second
x 0.16667	= Revolutions per minute
x 2.7778×10^{-3}	= Revolutions per second

Drams—dr

x 27.344	= Grains
x 0.0625	= Ounces
x 1.7718	= Grams

Fathoms

x 6	= Feet
x 1.8288	= Meters

Feet—ft

x 30.480	= Centimeters
x 12	= Inches
x 0.3048	= Meters
x 0.3333	= Yards

Feet of Water, at 68 F

x 0.02984	= Bars
x 0.8811	= Inches of mercury, at 0 C
x 0.03042	= Kilograms (kiloponds) per square centimeter
x 62.32	= Pounds per square foot
x 0.4328	= Pounds per square inch
x 0.02945	= Standard atmospheres

Feet Per Minute—fpm

x 0.5080	= Centimeters per second
x 0.016667	= Feet per second
x 0.01829	= Kilometers per hour
x 0.3048	= Meters per minute
x 0.01136	= Miles per hour

Feet Per Second Per Second—fpsps

x 30.48	= Centimeters per second per second (gals)
x 0.3048	= Meters per second per second

Foot-Pounds—ft-lb

x 1.285×10^{-3}	= British Thermal Units
x 3.239×10^{-4}	= Kilogram-calories
x 0.13825	= Kilogram-meters
x 5.050×10^{-7}	= Horsepower-hours
x 3.766×10^{-7}	= Kilowatt-hours

Gallons—gal

x 8.322	= Pounds of water, at 68 F in air
x 8.330	= Pounds of water, at 68 F in vacuo
x 3785	= Cubic centimeters
x 0.13368	= Cubic feet
x 231	= Cubic inches
x 3.785×10^{-3}	= Cubic meters
x 4.951×10^{-3}	= Cubic yards
x 0.8327	= Gallons (Imperial)
x 8	= Pints (liquid)
x 4	= Quarts (liquid)
x 3.785	= Liters

Gallons, Imperial—gal

x 10.000	= Pounds of water, at 62 F in air
x 4546	= Cubic centimeters
x 0.16054	= Cubic feet
x 4.546×10^{-3}	= Cubic meters
x 5.946×10^{-3}	= Cubic yards
x 1.20094	= Gallons
x 4.546	= Liters

Gallons Per Minute—gpm

x 2.228×10^{-3}	= Cubic feet per second
x 8.021	= Cubic feet per hour
x 0.22715	= Cubic meters per hour
x 0.06308	= Liters per second

Grains, Avoirdupois *or* Troy

Grains avoirdupois	= Grains troy
x 0.0648	= Grams

Grains Per Gallon, at 68 F

x 17.17	= Parts per million by weight in water
x 142.9	= Pounds per million gallons

Grains Per Imperial Gallon at 62 F

x 14.29	= Parts per million by weight in water

Grams—g

x 980.7	= Dynes
x 15.432	= Grains
x 0.035274	= Ounces
x 0.032151	= Ounces (troy)
x 2.2046×10^{-3}	= Pounds

Grams Per Centimeter—g per cm

x 5.600×10^{-3}	= Pounds per inch

Grams Per Cubic Centimeter—g per cu cm

x 62.43	= Pounds per cubic foot
x 0.03613	= Pounds per cubic inch

Grams Per Liter—g per l

x 58.42	= Grains per gallon
x 8.345	= Pounds per 1000 gallons
x 0.06243	= Pounds per cubic foot
x 1003	= Parts per million by weight in water, at 68 F

Hectares—ha

x 1.0764×10^{5}	= Square feet

Horsepower—hp

x 33,000	= Foot-pounds per minute
x 550	= Foot-pounds per second
x 42.42	= British Thermal Units per minute
x 10.69	= Kilogram-calories per minute
x 1.0139	= Horsepower (metric)
x 0.7457	= Kilowatts
x 745.7	= Watts

Horsepower, Boiler—hp

x 33,479	= British Thermal Units per hour
x 9.803	= Kilowatts

Horsepower-Hours—hp-hr

x 1.98×10^{6}	= Foot-pounds
x 2545	= British Thermal Units
x 641.3	= Kilogram-calories
x 2.737×10^{5}	= Kilogram-meters
x 0.7457	= Kilowatt-hours

Inches—in.
x 2.540 = Centimeters

Inches of Mercury, at 0 C
x 0.03386 = Bars
x 1.135 = Feet of water, at 68 F
x 13.62 = Inches of water, at 68 F
x 0.03453 = Kilograms (kiloponds) per square centimeter
x 70.73 = Pounds per square foot
x 0.4912 = Pounds per square inch
x 0.03342 = Standard atmospheres

Inches of Water, at 68 F
x 2.486×10^{-3} = Bars
x 0.07342 = Inches of mercury, at 0 C
x 2.535×10^{-3} = Kilograms (kiloponds) per square centimeter
x 0.5770 = Ounces per square inch
x 5.193 = Pounds per square foot
x 0.03606 = Pounds per square inch
x 2.454×10^{-3} = Standard atmospheres

Kilograms—kg
x 2.2046 = Pounds
x 1.102×10^{-3} = Tons (short)

Kilograms Per Meter—kg per m
x 0.6720 = Pounds per foot

Kilograms (Kiloponds) Per Square Centimeter—kg (kp) per sq. cm.
x 0.9807 = Bars
x 32.87 = Feet of water, at 68 F
x 28.96 = Inches of mercury, at 0 C
x 2048 = Pounds per square foot
x 14.223 = Pounds per square inch
x 0.9678 = Standard atmospheres

Kilograms Per Square Millimeter—kg per sq mm
x 10^6 = Kilograms per square meter

Kilometers Per Hour—kmph
x 27.78 = Centimeters per second
x 0.9113 = Feet per second
x 54.68 = Feet per minute
x 16.667 = Meters per minute
x 0.5396 = Knots
x 0.6214 = Miles per hour

Kilometers Per Hour Per Second—kmphps
x 27.78 = Centimeters per second per second
x 0.9113 = Feet per second per second
x 0.2778 = Meters per second per second

Kilowatts—kw
x 4.425×10^4 = Foot-pounds per minute
x 737.6 = Foot-pounds per second
x 56.9 = British Thermal Units per minute
x 14.33 = Kilogram-calories per minute
x 1.3410 = Horsepower

Kilowatthours—kwhr
x 2.655×10^6 = Foot-pounds
x 3413 = British Thermal Units
x 860 = Kilogram-calories
x 3.671×10^5 = Kilogram-meters
x 1.3410 = Horsepower-hours

Liters—l
x 1000.0 = Cubic centimeters
x 0.035315 = Cubic feet
x 61.024 = Cubic inches
x 1.308×10^{-3} = Cubic yards
x 0.2642 = Gallons
x 0.2200 = Gallons (Imperial)

Liters Per Minute—l per min
x 5.885×10^{-4} = Cubic feet per second
x 4.403×10^{-3} = Gallons per second
x 3.666×10^{-3} = Gallons per second (Imperial)

Meters—m
x 3.281 = Feet
x 39.37 = Inches
x 1.0936 = Yards

Meters Per Minute—m per min
x 1.6667 = Centimeters per second
x 3.281 = Feet per minute
x 0.05468 = Feet per second
x 0.0600 = Kilometers per hour
x 0.03728 = Miles per hour

Meters Per Second—m per sec
x 196.8 = Feet per minute
x 3.281 = Feet per second
x 3.600 = Kilometers per hour
x 0.0600 = Kilometers per minute
x 2.237 = Miles per hour
x 0.03728 = Miles per minute

Microns—mu *or* μ
x 10^{-6} = Meters

Miles
x 1.6094×10^5 = Centimeters
x 5280 = Feet
x 1.6094 = Kilometers
x 1760 = Yards

Miles Per Hour—mph
x 44.70 = Centimeters per second
x 88 = Feet per minute
x 1.4667 = Feet per second
x 1.6094 = Kilometers per hour
x 0.8684 = Knots
x 26.82 = Meters per minute

Miles Per Minute
x 2682 = Centimeters per second
x 88 = Feet per second
x 1.6094 = Kilometers per minute
x 60 = Miles per hour

Milliers (Metric Ton, Tonne)
x 10^3 = Kilograms

Minutes, Angular
x 2.909×10^{-4} = Radians

Ounces—oz
x 16 = Drams
x 437.5 = Grains
x 0.06250 = Pounds
x 28.35 = Grams
x 0.9115 = Ounces (troy)
x 2.790×10^{-5} = Tons (long)
x 2.835×10^{-5} = Tons (metric)

Ounces, Troy—oz
x 480 = Grains
x 20 = Pennyweights (troy)
x 0.08333 = Pounds (troy)
x 31.103 = Grams
x 1.0971 = Ounces (avoirdupois)

Ounces, Fluid—oz
x 1.8046 = Cubic inches
x 0.02957 = Liters

Ounces Per Square Inch—oz per sq in.
x 0.06250 = Pounds per square inch
x 4.395 = Grams per square centimeter

Parts Per Million—ppm by weight in water
x 0.0582 = Grains per gallon, at 68 F
x 0.0700 = Grains per Imperial gallon, at 62 F
x 8.322 = Pounds per million gallons, at 68 F

Pennyweights, Troy—dwt

x 24	= Grains
x 1.5552	= Grams

Poise

x 100	= Centipoise
x 2.0886×10^{-3}	= Pound (weight) second per square foot
x .06721	= Pound (mass) per foot-second

Pounds—lb

x 16	= Ounces
x 256	= Drams
x 7000	= Grains
x 5×10^{-4}	= Tons (short)
x 453.6	= Grams
x 1.2153	= Pounds (troy)

Pounds, Troy—lb

x 5760	= Grains
x 240	= Pennyweights (troy)
x 12	= Ounces (troy)
x 373.2	= Grams
x 0.8229	= Pounds (avoirdupois)
x 13.166	= Ounces (avoirdupois)
x 3.6735×10^{-4}	= Tons (long)
x 4.1143×10^{-4}	= Tons (short)
x 3.7324×10^{-4}	= Tons (metric)

Pounds of Water, at 68 F

x 0.01604	= Cubic feet
x 27.72	= Cubic inches
x 0.1200	= Gallons

Pounds of Water Per Minute, at 68 F

x 2.673×10^{-4}	= Cubic feet per second

Pounds Per Cubic Foot—lb per cu ft

x 0.016018	= Grams per cubic centimeter
x 16.018	= Kilograms per cubic meter
x 5.787×10^{-4}	= Pounds per cubic inch

Pounds Per Cubic Inch—lb per cu in.

x 27.68	= Grams per cubic centimeter
x 2.768×10^{4}	= Kilograms per cubic meter
x 1728	= Pounds per cubic foot

Pounds Per Foot—lb per ft

x 1.488	= Kilograms per meter
x 14.88	= Grams per centimeter

Pounds Per Square Foot—psf

x 0.01605	= Feet of water, at 68 F
x 4.882×10^{-4}	= Kilograms per square centimeter
x 6.944×10^{-3}	= Pounds per square inch

Pounds Per Square Inch—psi

x 0.06805	= Atmospheres
x 2.311	= Feet of water, at 68 F
x 27.73	= Inches of water, at 68 F
x 2.036	= Inches of mercury, at 0 C
x 0.07031	= Kilograms per square centimeter

Quarts, Dry—qt

x 67.20	= Cubic inches

Quarts, Liquid—qt

x 57.75	= Cubic inches

Quintal (Metric)

x 100	= Kilograms
x 220.46	= Pounds
x 101.28	= Pounds (Argentina)
x 129.54	= Pounds (Brazil)
x 101.41	= Pounds (Chile)
x 101.47	= Pounds (Mexico)
x 101.43	= Pounds (Peru)

Stokes

x 1.076×10^{-3}	= Square feet per second

Temperature—temp

Degrees Celsius—C

C + 273.15	= K Kelvin
(C x 9/5) + 32	= F Fahrenheit
C x 4/5	= R Réaumur

Degrees Fahrenheit—F

F + 459.67	= Rankine
(F − 32) x 5/9	= C Celsius
(F − 32) x 4/9	= R Réaumur

Degrees Réaumur—R

R x 5/4	= C Celsius
(R x 9/4) +32	= F Fahrenheit

Tons, Long

x 1016	= Kilograms
x 2240	= Pounds
x 1.1200	= Tons (short)

Tons, Metric (Tonne, Millier)

x 1000	= Kilograms
x 2204.6	= Pounds

Tons, Short

x 2000	= Pounds
x 32000	= Ounces
x 907.2	= Kilograms
x 2430.6	= Pounds (troy)
x 0.8929	= Tons (long)
x 0.9072	= Tons (metric)

Tons of Water Per 24 Hours, at 68 F

x 83.33	= Pounds water per hour
x 0.1667	= Gallons per minute
x 1.337	= Cubic feet per hour

Watts—w

x 0.0569	= British Thermal Units per minute
x 44.25	= Foot-pounds per minute
x 0.7376	= Foot-pounds per second
x 1.341×10^{-3}	= Horsepower
x 0.01433	= Kilogram-calories per minute

Watthours—whr

x 3.413	= British Thermal Units
x 2655	= Foot-pounds
x 1.341×10^{-3}	= Horsepower-hours
x 0.860	= Kilogram-calories
x 367.1	= Kilogram-meters

TEMPERATURE CONVERSION TABLES

Fahrenheit and Celsius (Centigrade)

C	*	F	C	*	F	C	*	F	C	*	F	C	*	F
-273.15	-459.67		-17.2	1	33.8	10.6	51	123.8	43	110	230	266	510	950
-268	-450		-16.7	2	35.6	11.1	52	125.6	49	120	248	271	520	968
-262	-440		-16.1	3	37.4	11.7	53	127.4	54	130	266	277	530	986
-257	-430		-15.6	4	39.2	12.2	54	129.2	60	140	284	282	540	1004
-251	-420		-15.0	5	41.0	12.8	55	131.0	66	150	302	288	550	1022
-246	-410		-14.4	6	42.8	13.3	56	132.8	71	160	320	293	560	1040
-240	-400		-13.9	7	44.6	13.9	57	134.6	77	170	338	299	570	1058
-234	-390		-13.3	8	46.4	14.4	58	136.4	82	180	356	304	580	1076
-229	-380		-12.8	9	48.2	15.0	59	138.2	88	190	374	310	590	1094
-223	-370		-12.2	10	50.0	15.6	60	140.0	93	200	392	316	600	1112
-218	-360		-11.7	11	51.8	16.1	61	141.8	99	210	410	321	610	1130
-212	-350		-11.1	12	53.6	16.7	62	143.6				327	620	1148
-207	-340		-10.6	13	55.4	17.2	63	145.4				332	630	1166
-201	-330		-10.0	14	57.2	17.8	64	147.2				338	640	1184
-196	-320		-9.4	15	59.0	18.3	65	149.0				343	650	1202
-190	-310		-8.9	16	60.8	18.9	66	150.8	100	212	413	349	660	1220
-184	-300		-8.3	17	62.6	19.4	67	152.6				354	670	1238
-179	-290		-7.8	18	64.4	20.0	68	154.4				360	680	1256
-173	-280		-7.2	19	66.2	20.6	69	156.2				366	690	1274
-169	-273	-459.4	-6.7	20	68.0	21.1	70	158.0				371	700	1292
-168	-270	-454	-6.1	21	69.8	21.7	71	159.8				377	710	1310
-162	-260	-436	-5.6	22	71.6	22.2	72	161.6	104	220	428	382	720	1328
-157	-250	-418	-5.0	23	73.4	22.8	73	163.4	110	230	446	388	730	1346
-151	-240	-400	-4.4	24	75.2	23.3	74	165.2	116	240	464	393	740	1364
-146	-230	-382	-3.9	25	77.0	23.9	75	167.0	121	250	482	399	750	1382
-140	-220	-364	-3.3	26	78.8	24.4	76	168.8	127	260	500	404	760	1400
-134	-210	-346	-2.8	27	80.6	25.0	77	170.6	132	270	518	410	770	1418
-129	-200	-328	-2.2	28	82.4	25.6	78	172.4	138	280	536	416	780	1436
-123	-190	-310	-1.7	29	84.2	26.1	79	174.2	143	290	554	421	790	1454
-118	-180	-292	-1.1	30	86.0	26.7	80	176.0	149	300	572	427	800	1472
-112	-170	-274	-0.6	31	87.8	27.2	81	177.8	154	310	590	432	810	1490
-107	-160	-256	0	32	89.6	27.8	82	179.6	160	320	608	438	820	1508
-101	-150	-238	0.6	33	91.4	28.3	83	181.4	166	330	626	443	830	1526
-95.6	-140	-220	1.1	34	93.2	28.9	84	183.2	171	340	644	449	840	1544
-90.0	-130	-202	1.7	35	95.0	29.4	85	185.0	177	350	662	454	850	1562
-84.4	-120	-184	2.2	36	96.8	30.0	86	186.8	182	360	680	460	860	1580
-78.9	-110	-166	2.8	37	98.6	30.6	87	188.6	188	370	698	466	870	1598
-73.3	-100	-148	3.3	38	100.4	31.1	88	190.4	193	380	716	471	880	1616
-67.8	-90	-130	3.9	39	102.2	31.7	89	192.2	199	390	734	477	890	1634
-62.2	-80	-112	4.4	40	104.0	32.2	90	194.0	204	400	752	482	900	1652
-56.7	-70	-94	5.0	41	105.8	32.8	91	195.8	210	410	770	488	910	1670
-51.1	-60	-76	5.6	42	107.6	33.3	92	197.6	216	420	788	493	920	1688
-45.6	-50	-58	6.1	43	109.4	33.9	93	199.4	221	430	806	499	930	1706
-40.0	-40	-40	6.7	44	111.2	34.4	94	201.2	227	440	824	504	940	1724
-34.4	-30	-22	7.2	45	113.0	35.0	95	203.0	232	450	842	510	950	1742
-28.9	-20	-4	7.8	46	114.8	35.6	96	204.8	238	460	860	516	960	1760
-23.3	-10	14	8.3	47	116.6	36.1	97	206.6	243	470	878	521	970	1778
-17.8	0	32	8.9	48	118.4	36.7	98	208.4	249	480	896	527	980	1796
			9.4	49	120.2	37.2	99	210.2	254	490	914	532	990	1814
			10.0	50	122.0	37.8	100	212.0	260	500	932	538	1000	1832

*In the center (colored) column, find the temperature to be converted. The equivalent temperature is in the left column, if converting to Celsius, and in the right column, if converting to Fahrenheit.

INTERPOLATION VALUES

C	*	F	C	*	F
0.56	1	1.8	3.33	6	10.8
1.11	2	3.6	3.89	7	12.6
1.67	3	5.4	4.44	8	14.4
2.22	4	7.2	5.00	9	16.2
2.78	5	9.0	5.56	10	18.0

TEMPERATURE CONVERSION TABLES

Fahrenheit and Celsius (Centigrade) (continued)

C	*	F	C	*	F	C	*	F	C	*	F
543	1010	1850	821	1510	2750	1099	2010	3650	1377	2510	4550
549	1020	1868	827	1520	2768	1104	2020	3668	1382	2520	4568
554	1030	1886	832	1530	2786	1110	2030	3686	1388	2530	4586
560	1040	1904	838	1540	2804	1116	2040	3704	1393	2540	4604
566	1050	1922	843	1550	2822	1121	2050	3722	1399	2550	4622
571	1060	1940	849	1560	2840	1127	2060	3740	1404	2560	4640
577	1070	1958	854	1570	2858	1132	2070	3758	1410	2570	4658
582	1080	1976	860	1580	2876	1138	2080	3776	1416	2580	4676
588	1090	1994	866	1590	2894	1143	2090	3794	1421	2590	4694
593	1100	2012	871	1600	2912	1149	2100	3812	1427	2600	4712
599	1110	2030	877	1610	2930	1154	2110	3830	1432	2610	4730
604	1120	2048	882	1620	2948	1160	2120	3848	1438	2620	4748
610	1130	2066	888	1630	2966	1166	2130	3866	1443	2630	4766
616	1140	2084	893	1640	2984	1171	2140	3884	1449	2640	4784
621	1150	2102	899	1650	3002	1177	2150	3902	1454	2650	4802
627	1160	2120	904	1660	3020	1182	2160	3920	1460	2660	4820
632	1170	2138	910	1670	3038	1188	2170	3938	1466	2670	4838
638	1180	2156	916	1680	3056	1193	2180	3956	1471	2680	4856
643	1190	2174	921	1690	3074	1199	2190	3974	1477	2690	4874
649	1200	2192	927	1700	3092	1204	2200	3992	1482	2700	4892
654	1210	2210	932	1710	3110	1210	2210	4010	1488	2710	4910
660	1220	2228	938	1720	3128	1216	2220	4028	1493	2720	4928
666	1230	2246	943	1730	3146	1221	2230	4046	1499	2730	4946
671	1240	2264	949	1740	3164	1227	2240	4064	1504	2740	4964
677	1250	2282	954	1750	3182	1232	2250	4082	1510	2750	4982
682	1260	2300	960	1760	3200	1238	2260	4100	1516	2760	5000
688	1270	2318	966	1770	3218	1243	2270	4118	1521	2770	5018
693	1280	2336	971	1780	3236	1249	2280	4136	1527	2780	5036
699	1290	2354	977	1790	3254	1254	2290	4154	1532	2790	5054
704	1300	2372	982	1800	3272	1260	2300	4172	1538	2800	5072
710	1310	2390	988	1810	3290	1266	2310	4190	1543	2810	5090
716	1320	2408	993	1820	3308	1271	2320	4208	1549	2820	5108
721	1330	2426	999	1830	3326	1277	2330	4226	1554	2830	5126
727	1340	2444	1004	1840	3344	1282	2340	4244	1560	2840	5144
732	1350	2462	1010	1850	3362	1288	2350	4262	1566	2850	5162
738	1360	2480	1016	1860	3380	1293	2360	4280	1571	2860	5180
743	1370	2498	1021	1870	3398	1299	2370	4298	1577	2870	5198
749	1380	2516	1027	1880	3416	1304	2380	4316	1582	2880	5216
754	1390	2534	1032	1890	3434	1310	2390	4334	1588	2890	5234
760	1400	2552	1038	1900	3452	1316	2400	4352	1593	2900	5252
766	1410	2570	1043	1910	3470	1321	2410	4370	1599	2910	5270
771	1420	2588	1049	1920	3488	1327	2420	4388	1604	2920	5288
777	1430	2606	1054	1930	3506	1332	2430	4406	1610	2930	5306
782	1440	2624	1060	1940	3524	1338	2440	4424	1616	2940	5324
788	1450	2642	1066	1950	3542	1343	2450	4442	1621	2950	5342
793	1460	2660	1071	1960	3560	1349	2460	4460	1627	2960	5360
799	1470	2678	1077	1970	3578	1354	2470	4478	1632	2970	5378
804	1480	2696	1082	1980	3596	1360	2480	4496	1638	2980	5396
810	1490	2714	1088	1990	3614	1366	2490	4514	1643	2990	5414
816	1500	2732	1093	2000	3632	1371	2500	4532	1649	3000	5432

Temperature Conversion Formulas

Degrees Celsius (formerly Centigrade) C

$C + 273.15$ = K Kelvin

$(C \times 9/5) + 32$ = F Fahrenheit

$C \times 4/5$ = R Réaumur

Degrees Fahrenheit—F

$F + 459.67$ = Rankine

$(F - 32) \times 5/9$ = C Celsius

$(F - 32) \times 4/9$ = R Réaumur

Degrees Réaumur—R

$R \times 5/4$ = C Celsius

$(R \times 9/4) + 32$ = F Fahrenheit

PROPERTIES OF SATURATED STEAM

Dry Saturated Steam: Pressure Table*

Abs Press., Lb Sq In.	Temp F	Specific Volume		Enthalpy			Entropy			Internal Energy		Abs Press., Lb Sq In.
		Sat. Liquid	Sat. Vapor	Sat. Liquid	Evap	Sat. Vapor	Sat. Liquid	Evap	Sat. Vapor	Sat. Liquid	Sat. Vapor	
p	t	v_f	v_g	h	h_{fg}	h_g	s_f	s_{fg}	s_g	u_f	u_g	p
1.0	101.74	0.01614	333.6	69.70	1036.3	1106.0	0.1326	1.8456	1.9782	69.70	1044.3	1.0
2.0	126.08	0.01623	173.73	93.99	1022.2	1116.2	0.1749	1.7451	1.9200	93.98	1051.9	2.0
3.0	141.48	0.01630	118.71	109.37	1013.2	1122.6	0.2008	1.6855	1.8863	109.36	1056.7	3.0
4.0	152.97	0.01636	90.63	120.86	1006.4	1127.3	0.2198	1.6427	1.8625	120.85	1060.2	4.0
5.0	162.24	0.01640	73.52	130.13	1001.0	1131.1	0.2347	1.6094	1.8441	130.12	1063.1	5.0
6.0	170.06	0.01645	61.98	137.96	996.2	1134.2	0.2472	1.5820	1.8292	137.94	1065.4	6.0
7.0	176.85	0.01649	53.64	144.76	992.1	1136.9	0.2581	1.5586	1.8167	144.74	1067.4	7.0
8.0	182.86	0.01653	47.34	150.79	988.5	1139.3	0.2674	1.5383	1.8057	150.77	1069.2	8.0
9.0	188.28	0.01656	42.40	156.22	985.2	1141.4	0.2759	1.5203	1.7962	156.19	1070.8	9.0
10	193.21	0.01659	38.42	161.17	982.1	1143.3	0.2835	1.5041	1.7876	161.14	1072.2	10
14.696	212.00	0.01672	26.80	180.07	970.3	1150.4	0.3120	1.4446	1.7566	180.02	1077.5	14.696
15	213.03	0.01672	26.29	181.11	969.7	1150.8	0.3135	1.4415	1.7549	181.06	1077.8	15
20	227.96	0.01683	20.089	196.16	960.1	1156.3	0.3356	1.3962	1.7319	196.10	1081.9	20
25	240.07	0.01692	16.303	208.42	952.1	1160.6	0.3533	1.3606	1.7139	208.34	1085.1	25
30	250.33	0.01701	13.746	218.82	945.3	1164.1	0.3680	1.3313	1.6993	218.73	1087.8	30
35	259.28	0.01708	11.898	227.91	939.2	1167.1	0.3807	1.3063	1.6870	227.80	1090.1	35
40	267.25	0.01715	10.498	236.03	933.7	1169.7	0.3919	1.2844	1.6763	235.90	1092.0	40
45	274.44	0.01721	9.401	243.36	928.6	1172.0	0.4019	1.2650	1.6669	243.22	1093.7	45
50	281.01	0.01727	8.515	250.09	924.0	1174.1	0.4110	1.2474	1.6585	249.93	1095.3	50
55	287.07	0.01732	7.787	256.30	919.6	1175.9	0.4193	1.2316	1.6509	256.12	1096.7	55
60	292.71	0.01738	7.175	262.09	915.5	1177.6	0.4270	1.2168	1.6438	261.90	1097.9	60
65	297.97	0.01743	6.655	267.50	911.6	1179.1	0.4342	1.2032	1.6374	267.29	1099.1	65
70	302.92	0.01748	6.206	272.61	907.9	1180.6	0.4409	1.1906	1.6315	272.38	1100.2	70
75	307.60	0.01753	5.816	277.43	904.5	1181.9	0.4472	1.1787	1.6259	277.19	1101.2	75
80	312.03	0.01757	5.472	282.02	901.1	1183.1	0.4531	1.1676	1.6207	281.76	1102.1	80
85	316.25	0.01761	5.168	286.39	897.8	1184.2	0.4587	1.1571	1.6158	286.11	1102.9	85
90	320.27	0.01766	4.896	290.56	894.7	1185.3	0.4641	1.1471	1.6112	290.27	1103.7	90
95	324.12	0.01770	4.652	294.56	891.7	1186.2	0.4692	1.1376	1.6068	294.25	1104.5	95
100	327.81	0.01774	4.432	298.40	888.8	1187.2	0.4740	1.1286	1.6026	298.08	1105.2	100
110	334.77	0.01782	4.049	305.66	883.2	1188.9	0.4832	1.1117	1.5948	305.30	1106.5	110
120	341.25	0.01789	3.728	312.44	877.9	1190.4	0.4916	1.0962	1.5878	312.05	1107.6	120
130	347.32	0.01796	3.455	318.81	872.9	1191.7	0.4995	1.0817	1.5812	318.38	1108.6	130
140	353.02	0.01802	3.220	324.82	868.2	1193.0	0.5069	1.0682	1.5751	324.35	1109.6	140
150	358.42	0.01809	3.015	330.51	863.6	1194.1	0.5138	1.0556	1.5694	330.01	1110.5	150
160	363.53	0.01815	2.834	335.93	859.2	1195.1	0.5204	1.0436	1.5640	335.39	1111.2	160
170	368.41	0.01822	2.675	341.09	854.9	1196.0	0.5266	1.0324	1.5590	340.52	1111.9	170
180	373.06	0.01827	2.532	346.03	850.8	1196.9	0.5325	1.0217	1.5542	345.42	1112.5	180
190	377.51	0.01833	2.404	350.79	846.8	1197.6	0.5381	1.0116	1.5497	350.15	1113.1	190
200	381.79	0.01839	2.288	355.36	843.0	1198.4	0.5435	1.0018	1.5453	354.68	1113.7	200
250	400.95	0.01865	1.8438	376.00	825.1	1201.1	0.5675	0.9588	1.5263	375.14	1115.8	250
300	417.33	0.01890	1.5433	393.84	809.0	1202.8	0.5879	0.9225	1.5104	392.79	1117.1	300
350	431.72	0.01913	1.3260	409.69	794.2	1203.9	0.6056	0.8910	1.4966	408.45	1118.0	350
400	444.59	0.0193	1.1613	424.0	780.5	1204.5	0.6214	0.8630	1.4844	422.6	1118.5	400
450	456.28	0.0195	1.0320	437.2	767.4	1204.6	0.6356	0.8378	1.4734	435.5	1118.7	450
500	467.01	0.0197	0.9278	449.4	755.0	1204.4	0.6487	0.8147	1.4634	447.6	1118.6	500
550	476.94	0.0199	0.8424	460.8	743.1	1203.9	0.6608	0.7934	1.4542	458.8	1118.2	550
600	486.21	0.0201	0.7698	471.6	731.6	1203.2	0.6720	0.7734	1.4454	469.4	1117.7	600
650	494.90	0.0203	0.7083	481.8	720.5	1202.3	0.6826	0.7548	1.4374	479.4	1117.1	650
700	503.10	0.0205	0.6554	491.5	709.7	1201.2	0.6925	0.7371	1.4296	488.8	1116.3	700
750	510.86	0.0207	0.6092	500.8	699.2	1200.0	0.7019	0.7204	1.4223	598.0	1115.4	750
800	518.23	0.0209	0.5687	509.7	688.9	1198.6	0.7108	0.7045	1.4153	506.6	1114.4	800
850	525.26	0.0210	0.5327	518.3	678.8	1197.1	0.7194	0.6891	1.4085	515.0	1113.3	850
900	531.98	0.0212	0.5006	526.6	668.8	1195.4	0.7275	0.6744	1.4020	523.1	1112.1	900
950	538.43	0.0214	0.4717	534.6	659.1	1193.7	0.7355	0.6602	1.3957	530.9	1110.8	950
1000	544.61	0.0216	0.4456	542.4	649.4	1191.8	0.7430	0.6467	1.3897	538.4	1109.4	1000
1100	556.31	0.0220	0.4001	557.4	630.4	1187.8	0.7575	0.6205	1.3780	552.9	1106.4	1100
1200	567.22	0.0223	0.3619	571.7	611.7	1183.4	0.7711	0.5956	1.3667	566.7	1103.0	1200
1300	577.46	0.0227	0.3293	585.4	593.2	1178.6	0.7840	0.5719	1.3559	580.0	1099.4	1300
1400	587.10	0.0231	0.3012	598.7	574.7	1173.4	0.7963	0.5491	1.3454	592.7	1095.4	1400
1500	596.23	0.0235	0.2765	611.6	556.3	1167.9	0.8082	0.5269	1.3351	605.1	1091.2	1500
2000	635.82	0.0257	0.1878	671.7	463.4	1135.1	0.8619	0.4230	1.2849	662.2	1065.6	2000
2500	668.13	0.0287	0.1307	730.6	360.5	1091.1	0.9126	0.3197	1.2322	717.3	1030.6	2500
3000	695.36	0.0346	0.0858	802.5	217.8	1020.3	0.9731	0.1885	1.1615	783.4	972.7	3000
3206.2	705.40	0.0503	0.0503	902.7	0	902.7	1.0580	0	1.0580	872.9	872.9	3206.2

*Abridged from "Thermodynamic Properties of Steam" by Joseph H. Keenan and Frederick G. Keyes.
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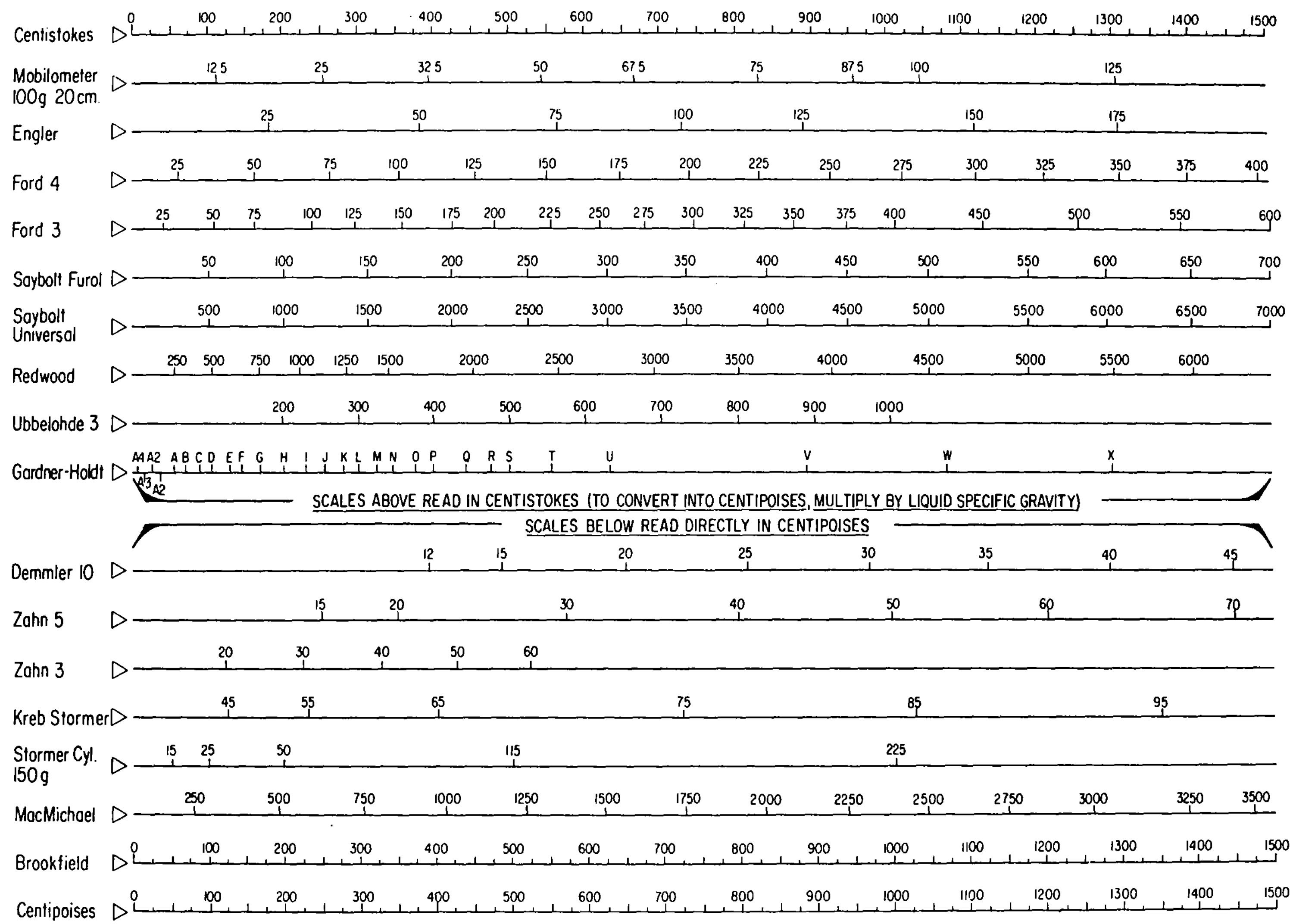
FOXBORO

VISCOSITY COMPARISON CHART



NOTE This chart is intended to be an aid in determining approximate comparative viscosities of Newtonian fluids.

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AR-15-1

LIQUID GRAVITY TABLES AND WEIGHT FACTORS

Liquid Lighter than Water

Sp Gr 60 F/60 F	°Bé	°API	Lb per gal at 60 F in vacuo	Lb per cu ft at 60 F in vacuo
0.600	103.33	104.33	5.0025	37.4219
.605	101.41	102.38	5.0442	37.7338
.610	99.51	100.47	5.0858	38.0456
.615	97.64	98.58	5.1275	38.3575
.620	95.80	96.73	5.1692	38.6693
.625	94.00	94.90	5.2109	38.9812
.630	92.22	93.10	5.2526	39.2930
.635	90.47	91.33	5.2943	39.6049
.640	88.75	89.59	5.3360	39.9167
.645	87.05	87.88	5.3776	40.2286
.650	85.38	86.19	5.4193	40.5404
.655	83.74	84.53	5.4610	40.8523
.660	82.12	82.89	5.5027	41.1641
.665	80.53	81.28	5.5444	41.4760
.670	78.95	79.69	5.5861	41.7878
.675	77.41	78.13	5.6278	42.0997
.680	75.88	76.59	5.6695	42.4115
.685	74.38	75.07	5.7111	42.7234
.690	72.90	73.57	5.7528	43.0352
.695	71.44	72.10	5.7945	43.3471
.700	70.00	70.64	5.8362	43.6589
.705	68.58	69.21	5.8779	43.9708
.710	67.18	67.80	5.9196	44.2826
.715	65.80	66.40	5.9613	44.5945
.720	64.44	65.03	6.0030	44.9063
.725	63.10	63.67	6.0446	45.2182
.730	61.78	62.34	6.0863	45.5300
.735	60.48	61.02	6.1280	45.8419
.740	59.19	59.72	6.1697	46.1537
.745	57.92	58.43	6.2114	46.4656
.750	56.67	57.17	6.2531	46.7774
.755	55.43	55.92	6.2948	47.0893
.760	54.21	54.68	6.3365	47.4011
.765	53.01	53.47	6.3781	47.7130
.770	51.81	52.27	6.4198	48.0248
.775	50.65	51.08	6.4615	48.3367
.780	49.49	49.91	6.5032	48.6485
.785	48.34	48.75	6.5449	48.9604
.790	47.22	47.61	6.5866	49.2722
.795	46.10	46.49	6.6283	49.5841
.800	45.00	45.38	6.6700	49.8959
.805	43.91	44.28	6.7116	50.2077
.810	42.84	43.19	6.7533	50.5196
.815	41.78	42.12	6.7950	50.8315
.820	40.73	41.06	6.8367	51.1433
.825	39.70	40.02	6.8784	51.4552
.830	38.67	38.98	6.9201	51.7670
.835	37.66	37.96	6.9618	52.0789
.840	36.67	36.95	7.0034	52.3907
.845	35.68	35.96	7.0451	52.7026
.850	34.71	34.97	7.0868	53.0144
.855	33.74	34.00	7.1285	53.3263
.860	32.79	33.03	7.1702	53.6381
.865	31.85	32.08	7.2119	53.9500
.870	30.92	31.14	7.2536	54.2618
.875	30.00	30.21	7.2953	54.5737
.880	29.09	29.30	7.3369	54.8855
.885	28.19	28.39	7.3786	55.1974
.890	27.30	27.49	7.4203	55.5092
.895	26.42	26.60	7.4620	55.8211
.900	25.56	25.72	7.5037	56.1329
.905	24.70	24.85	7.5454	56.4448
.910	23.85	23.99	7.5871	56.7566
.915	23.01	23.14	7.6288	57.0685
.920	22.17	22.30	7.6704	57.3803
.925	21.35	21.47	7.7121	57.6922
.930	20.54	20.65	7.7538	58.0040
.935	19.73	19.84	7.7955	58.3159
.940	18.93	19.03	7.8372	58.6277
.945	18.15	18.24	7.8789	58.9396
.950	17.37	17.45	7.9206	59.2514
.955	16.60	16.67	7.9623	59.5633
.960	15.83	15.90	8.0039	59.8751
.965	15.08	15.13	8.0456	60.1870
.970	14.33	14.38	8.0873	60.4988
.975	13.59	13.63	8.1290	60.8107
.980	12.86	12.89	8.1707	61.1225
.985	12.13	12.15	8.2124	61.4344
.990	11.41	11.43	8.2541	61.7462
.995	10.70	10.71	8.2958	62.0581
1.000	10.00	10.00	8.3374	62.3699

Liquid Heavier than Water

Sp Gr 60 F/60 F	°Bé	°Tw	Lb per gal at 60 F in vacuo	Lb per cu ft at 60 F in vacuo	Sp Gr 60 F/60 F	°Bé	°Tw	Lb per gal at 60 F in vacuo	Lb per cu ft at 60 F in vacuo
1.005	0.72	1	8.3791	62.6817	1.505	48.65	101	12.5478	93.8667
1.010	1.44	2	8.4208	62.9936	1.510	48.97	102	12.5895	94.1785
1.015	2.14	3	8.4625	63.3054	1.515	49.29	103	12.6312	94.4904
1.020	2.84	4	8.5042	63.6173	1.520	49.61	104	12.6729	94.8022
1.025	3.54	5	8.5459	63.9291	1.525	49.92	105	12.7146	95.1141
1.030	4.22	6	8.5876	64.2410	1.530	50.23	106	12.7563	95.4259
1.035	4.90	7	8.6293	64.5528	1.535	50.54	107	12.7980	95.7378
1.040	5.58	8	8.6709	64.8647	1.540	50.84	108	12.8397	96.0496
1.045	6.24	9	8.7126	65.1765	1.545	51.15	109	12.8813	96.3615
1.050	6.91	10	8.7543	65.4884	1.550	51.45	110	12.9230	96.6733
1.055	7.56	11	8.7960	65.8002	1.555	51.75	111	12.9647	96.9852
1.060	8.21	12	8.8377	66.1121	1.560	52.05	112	13.0064	97.2970
1.065	8.85	13	8.8794	66.4239	1.565	52.35	113	13.0481	97.6089
1.070	9.49	14	8.9211	66.7358	1.570	52.64	114	13.0898	97.9207
1.075	10.12	15	8.9627	67.0476	1.575	52.94	115	13.1315	98.2326
1.080	10.74	16	9.0044	67.3595	1.580	53.23	116	13.1732	98.5444
1.085	11.36	17	9.0461	67.6713	1.585	53.52	117	13.2148	98.8563
1.090	11.97	18	9.0878	67.9832	1.590	53.81	118	13.2565	99.1681
1.095	12.58	19	9.1295	68.2950	1.595	54.09	119	13.2982	99.4800
1.100	13.18	20	9.1712	68.6069	1.600	54.38	120	13.3399	99.7918
1.105	13.78	21	9.2129	68.9187	1.605	54.66	121	13.3816	100.1037
1.110	14.37	22	9.2546	69.2306	1.610	54.94	122	13.4233	100.4155
1.115	14.96	23	9.2962	69.5424	1.615	55.22	123	13.4650	100.7274
1.120	15.54	24	9.3380	69.8543	1.620	55.49	124	13.5067	101.0392
1.125	16.11	25	9.3796	70.1661	1.625	55.77	125	13.5483	101.3511
1.130	16.68	26	9.4213	70.4780	1.630	56.04	126	13.5900	101.6629
1.135	17.25	27	9.4630	70.7898	1.635	56.32	127	13.6317	101.9748
1.140	17.81	28	9.5047	71.1017	1.640	56.55	128	13.6734	102.2866
1.145	18.36	29	9.5464	71.4135	1.645	56.85	129	13.7151	102.5985
1.150	18.91	30	9.5881	71.7254	1.650	57.12	130	13.7568	102.9103
1.155	19.46	31	9.6297	72.0372	1.655	57.39	131	13.7985	103.2222
1.160	20.00	32	9.6714	72.3491	1.660	57.65	132	13.8402	103.5340
1.165	20.54	33	9.7131	72.6609	1.665	57.91	133	13.8818	103.8459
1.170	21.07	34	9.7548	72.9728	1.670	58.17	134	13.9235	104.1578
1.175	21.60	35	9.7965	73.2846	1.675	58.43	135	13.9652	104.4696
1.180	22.12	36	9.8382	73.5965	1.680	58.69	136	14.0069	104.7814
1.185	22.64	37	9.8799	73.9083	1.685	58.95	137	14.0486	105.0933
1.190	23.15	38	9.9216	74.2202	1.690	59.20	138	14.0903	105.4051
1.195	23.66	39	9.9632	74.5320	1.695	59.45	139	14.1320	105.7170
1.200	24.17	40	10.0049	74.8439	1.700	59.71	140	14.1736	106.0287
1.205	24.67	41	10.0466	75.1557	1.705	59.96	141	14.2153	106.3407
1.210	25.16	42	10.0883	75.4676	1.710	60.20	142	14.2570	106.6525
1.215	25.66	43	10.1300	75.7794	1.715	60.45	143	14.2987	106.9644
1.220	26.15	44	10.1717	76.0913	1.720	60.70	144	14.3404	107.2762
1.225	26.63	45	10.2134	76.4031	1.725	60.94	145	14.3821	107.5881
1.230	27.11	46	10.2551	76.7149	1.730	61.18	146	14.4238	107.8999
1.235	27.59	47	10.2967	77.0268	1.735	61.43	147	14.4655	108.2118
1.240	28.06	48	10.3384	77.3387	1.740	61.67	148	14.5071	108.5236
1.245	28.53	49	10.3801	77.6505	1.745	61.91	149	14.5488	108.8355
1.250	29.00	50	10.4218	77.9624	1.750	62.15	150	14.5905	109.1473
1.255	29.46	51	10.4635	78.2742	1.755	62.38	151	14.6322	109.4592
1.260	29.92	52	10.5052	78.5861	1.760	62.61	152	14.6739	109.7710
1.265	30.38	53	10.5469	78.8979	1.765	62.85	153	14.7156	110.0829
1.270	30.83	54	10.5885	79.2098	1.770	63.08	154	14.7573	110.3947
1.275	31.27	55	10.6302	79.5216	1.775	63.31	155	14.7990	110.7066
1.280	31.72	56	10.6719	79.8335	1.780	63.54	156	14.8406	111.0184
1.285	32.16	57	10.7136	80.1453	1.785	63.77	157	14.8823	111.3303
1.290	32.60	58	10.7553	80.4572	1.790	64.00	158	14.9240	111.6421
1.295	33.03	59	10.7970	80.7690	1.795	64.22	159	14.9657	111.9540
1.300	33.46	60	10.8387	81.0809	1.800	64.44	160	15.0074	112.2658
1.305	33.89	61	10.8804	81.3927	1.805	64.67	161	15.0491	112.5777
1.310	34.31	62	10.9220	81.7046	1.810	64.89	162	15.0908	112.8895
1.315	34.73	63	10.9637	82.0164	1.815	65.11	163	15.1325	113.2014
1.320	35.15	64	11.0054	82.3283	1.820	65.33	164	15.1741	113.5132
1.325	35.57	65	11.0471	82.6401	1.825	65.55	165	15.2158	113.8251
1.330	35.98	66	11.0888	82.9520	1.830	65.77	166	15.2575	114.1369
1.335	36.39	67	11.1305	83.2638	1.835	65.98	167	15.2992	114.4488
1.340	36.79	68	11.1722	83.5757	1.840	66.20	168	15.3409	114.7606
1.345	37.19	69	11.2139	83.8875	1.845	66.41	169	15.3826	115.0725
1.350	37.59	70	11.2555	84.1994	1.850	66.62	170	15.4243	115.3843
1.355	37.99	71	11.2972	84.5112	1.855	66.83	171	15.4660	115.6962
1.360	38.38	72	11.3389	84.8231	1.860	67.04	172	15.5076	116.0080
1.365	38.77	73	11.3806	85.1349	1.865	67.25	173	15.5493	116.3199
1.370	39.16	74	11.4223	85.4468	1.870	67.46	174	15.5910	116.6317
1.375	39.55	75	11.4640	85.7586	1.875	67.67	175	15.6327	116.9436
1.380	39.93	76	11.5057	86.0705	1.880	67.87	176	15.6744	117.2554
1.385	40.31	77	11.5474	86.3823	1.885	68.08	177	15.7161	117.5673
1.390	40.68	78	11.5890	86.6942	1.890	68.28	178	15.7578	117.8791
1.395	41.06	79	11.6307	87.0060	1.895	68.48	179	15.7994	118.1910
1.400	41.43	80	11.6724	87.3179	1.900	68.68	180	15.8411	118.5028
1.405	41.80	81	11.7141	87.6297	1.905	68.88	181	15.8828	118.8147
1.410	42.17	82	11.7558	87.9416	1.910	69.08	182	15.9245	119.1265
1.415	42.53	83	11.7975	88.2534	1.915	69.28	183	15.9662	119.4384
1.420	42.89	84	11.8392	88.5653	1.920	69.48	184	16.0079	119.7502
1.425	43.25	85	11.8809	88.8771	1.925	69.68	185	16.0496	120.0621
1.430	43.60	86	11.9225	89.1890	1.930	69.87	186	16.0913	120.3739
1.435	43.95	87	11.9642	89.5008	1.935	70.06	187	16.1329	120.6858
1.440	44.31	88	12.0059	89.8127	1.940	70.26	188	16.1746	120.9976
1.445	44.65	89	12.0476	90.1245	1.945	70.45	189	16.2163	121.3095
1.450	45.00	90	12.0893	90.4364	1.950	70.64	190	16.2580	121.6213
1.455	45.34	91	12.1310	90.7482	1.955	70.83	191	16.2997	121.9332
1.460	45.68	92	12.1727	91.0600	1.960	71.02	192	16.3414	122.2450
1.465	46.02	93	12.2143	91.3719	1.965	71.21	193	16.3831	122.5569
1.470	46.36	94	12.2560	91.6838	1.970	71.40	194	16.4248	122.8687
1.475	46.69	95	12.2977	91.9956	1.975	71.58	195	16.4664	123.1806
1.480	47.03	96	12.3394	92.3075	1.980	71.77	196	16.5081	123.4924
1.485	47.36	97	12.3811	92.6193	1.985	71.95	197	16.5498	123.8043
1.490	47.68	98	12.4228	92.9312	1.990	72.14	198	16.5915	124.1161
1.495	48.01	99	12.4645	93.2430	1.995	72.32	199	16.6332	124.4280
1.500	48.33	100	12.5062	93.5549	2.000	72.50	200	16.6749	124.7398

FOXBORO

PHYSICAL CONSTANTS OF PARAFFIN HYDROCARBONS

COMPOUND	METHANE	ETHANE	PROPANE	ISO-BUTANE	N-BUTANE	ISO-PENTANE	N-PENTANE	N-HEXANE	N-HEPTANE	N-OCTANE	N-NONANE	N-DECANE
MOLECULAR FORMULA	CH_4	C_2H_6	C_3H_8	C_4H_{10}	C_4H_{10}	C_5H_{12}	C_5H_{12}	C_6H_{14}	C_7H_{16}	C_8H_{18}	C_9H_{20}	$C_{10}H_{22}$
MOLECULAR WEIGHT	16.042	30.068	44.094	58.120	58.120	72.146	72.146	86.172	100.198	114.224	128.250	142.276
MELTING POINT at 14.696 psia F	−296.5[d]	−297.9	−305.8	−255.3	−217.0	−255.8	−201.5	−139.6	−131.1	−70.2	−64.4	−21.5
" " C	−182.5[d]	−183.3	−187.7	−159.5	−138.3	−159.9	−129.7	−95.3	−90.6	−56.8	−53.6	−29.7
BOILING POINT at 14.696 psia F	−258.7	−127.5	−43.7	10.9	31.1	82.1	96.9	155.7	209.2	258.2	303.4	345.2
" " C	−161.5	−88.6	−42.1	−11.7	−0.5	27.9	36.1	68.7	98.4	125.7	150.8	174.0
DENSITY OF LIQUID at 60 F and 14.696 psia Specific Gravity at 60/60 F[b]	0.3[g]	0.374	0.5077[h]	0.5631[h]	0.5844[h]	0.6248	0.6312	0.6640	0.6882	0.7068	0.7217	0.7341
° API[b]	340[g]	247	147.2	119.8	110.6	95.0	92.7	81.6	74.1	68.7	64.6	61.3
Lb per gal at 60 F[c]	2.5[g]	3.11	4.224[h]	4.685[h]	4.863[h]	5.200	5.253	5.527	5.728	5.883	6.008	6.114
Gal per lb mole at 60 F	6.4[g]	9.67	10.44[h]	12.40[h]	11.95[h]	13.88	13.74	15.59	17.49	19.41	21.35	23.27
DENSITY OF VAPOR at 60 F and 14.696 psia Specific gravity air = 1.00—ideal gas	0.554[e]	1.038[e]	1.522[e]	2.006[e]	2.006[e]	2.491	2.491	2.975	3.459	3.943	4.428	4.912
" " —actual (corrected)	0.555	1.046	1.546	2.066	2.070	—	—	—	—	—	—	—
Lb per M cu ft—ideal gas	42.27[e]	79.23[e]	116.19[e]	153.15[e]	153.15[e]	190.11	190.11	227.07	264.03	300.99	337.95	374.91
" " —actual (corrected)	42.35	79.86	118.0	157.7	158.1	—	—	—	—	—	—	—
Cu ft vapor per gal liq—ideal gas	59[e]	39.25[e]	36.35[e]	30.59[e]	31.75[e]	27.40	27.68	24.38	21.73	19.58	17.80	16.32
" " —actual (corrected)	—	—	35.78	29.70	30.77	—	—	—	—	—	—	—
RATIO, GAS VOL PER LIQ VOL—ideal gas	442[e]	293.6[e]	271.9[e]	228.8[e]	237.5[e]	205.0	207.1	182.4	162.6	146.5	133.2	122.1
" " —actual (corrected)	—	—	267.6	222.1	230.1	—	—	—	—	—	—	—
CRITICAL CONDITIONS Temperature—F	−116.5	90.1	206.3	275.0	305.6	370.0	385.9	454.5	512.6	565.2	613.0	655.0
" C	−82.5	32.3	96.8	135.0	152.0	187.8	196.6	234.7	267.0	296.2	322.8	346.1
Pressure—atmosphere	45.8	48.2	42.0	36.0	37.5	32.9	33.3	29.9	27.0	24.6	23.5	21.8
Pressure—psia	673	708	617	529	551	483	490	440	397	362	345	320
GROSS HEAT OF COMBUSTION at 60 F Btu per lb vapor—ideal gas	23,891	22,329	21,670	21,265	21,315	21,046	21,094	20,949	20,842	20,764	20,701	20,653
Btu per cu ft—ideal gas	1,010[e]	1,769[e]	2,518[e]	3,256[e]	3,264[e]	4,001	4,009	4,756	5,503	6,250	6,996	7,743
" " —actual (corrected)	1,012	1,783	2,558	3,354	3,368	—	—	—	—	—	—	—
Btu per gal liq at 60 F	—	69,433(f)	91,044	99,097	103,047	108,820	110,125	115,069	118,658	121,420	123,625	126,456
FLAMMABILITY LIMITS Lower percent in air	5.0	3.22	2.37	1.80	1.86	1.32	1.40	1.25	1.0	0.84	0.74	0.67
Upper percent in air	15.0	12.45	9.50	8.44	8.41	—	7.80	6.90	6.0	3.2	2.9	2.6
CU FT OF AIR to burn 1 cu ft gas	9.53	16.67	23.82	30.97	30.97	38.11	38.11	45.26	52.41	59.55	66.70	73.85
HEAT OF VAPORIZATION at 14.696 psia Btu per lb at boiling point	219.7	210.7	183.5	157.8	165.9	145.9	153.8	144.2	136.2	131.9	126.9	120.2
SPECIFIC HEAT at 60 F and 14.696 psia C_p vapor Btu per lb	0.5271	0.4097	0.3885	0.3872	0.3970	0.3880	0.3972	0.3984	0.3992	0.3998	0.4003	0.4006
C_v vapor Btu per lb	0.402	0.343	0.342	0.352	0.363	0.361	0.370	0.375	0.379	0.382	0.385	0.387
N C_p/C_v Btu per lb	1.308	1.193	1.133	1.097	1.094	1.076	1.074	1.062	1.052	1.046	1.040	1.034
C_p liquid Btu per lb	—	—	0.534 at −43 F	0.537 at 14 F	0.548 at 0 F	0.533	0.536	0.552	0.528	0.523	0.522	0.520
VAPOR PRESSURE at 100 F, psia	—	780[j]	190	72.2	51.6	20.4	15.6	4.96	1.62	0.54	0.18	0.07
ANILINE POINT—F	—	—	—	225.7	181.2	168.3	159.3	155.5[i]	159.1	161.1	166.8	171.5
REFRACTIVE INDEX[a] $N\frac{20C}{D}$ at 68 F	—	—	—	—	—	1.3537	1.3575	1.3749	1.3876	1.3974	1.4054	1.4120
DIELECTRIC CONSTANT* at 20 C	—	—	1.61 at 0 C	—	—	1.843	1.844	1.890	1.924	1.948	1.972	1.991

[a] Air saturated hydrocarbons
[b] Absolute values from weights in vacuum
[c] Apparent values from weights in air
[d] At saturation pressure (triple point)
[e] Based on "perfect gas"
[f] Calculated
[g] Apparent value for dissolved Methane at 60 F
[h] Saturation pressure
[i] Critical solution temperatures
[j] Extrapolated value
*Dielectric Constants from NBS Circular 514; other data from NGAA Publication 2145—Revised 1957.

CONVERSION DATA FOR HYDROCARBON CALCULATIONS

Atomic weights: Carbon—12.01; Hydrogen—1.008.

Molecular weight of air—28.966.

Perfect gas at 32 F and 14.696 psia = 22.414 liter per gram mole.

Perfect gas at 60 F and 14.696 psia = 379.498 cu ft per mole.

Specific gravity at 60/60 F x 0.999015 = density at 60 F in grams per cc.

Density of water at 60 F = 8.337 lb per gal = 0.999015 grams per cc.

$$\text{Degrees API} = \frac{141.5}{\text{Sp Gr at } 60/60\text{ F}} - 131.5$$

Degrees Fahrenheit = 459.67 Rankine.

760 mm Hg = 14.696 psia.

1 pound = 453.6 grams.

1 cu ft = 28.32 liter.

1 cu ft = 7.481 gal.

1 gal = 3,785 ml.

STAINLESS STEEL COMPOSITION (Percent)

AISI* Type	Cr	Ni	C	Mn	Si	P	S	Other	REMARKS
201	16–18	3.5–5.5	.15 max	5.5–7.5	1 max	.06 max	.03 max	N .25 max	Low-nickel Equivalent of Type 301
202	17–19	4–6	.15 max	7.5–10	1 max	.06 max	.03 max	N .25 max	Low-nickel Equivalent of Type 302
301	16–18	6–8	.15 max	2 max	1 max	.045 max	.03 max		High Work-hardening
302	17–19	8–10	.15 max	2 max	1 max	.045 max	.03 max		General-Purpose "18-8"
302B	17–19	8–10	.15 max	2 max	2–3	.045 max	.03 max		More Scaling Resistance than Type 302
303	17–19	8–10	.15 max	2 max	1 max	.2 max	.15 min	.6 Mo or .6 Zr optional	Free Machining "18-8"—Heavy Cuts
303Se	17–19	8–10	.15 max	2 max	1 max	.2 max	.06 max	Se .15 min	Free Machining "18-8"—Light Cuts
304	18–20	8–12	.08 max	2 max	1 max	.045 max	.03 max		Low-Carbon—For Welding
304L	18–20	8–12	.03 max	2 max	1 max	.045 max	.03 max		Lower-Carbon—For Welding
305	17–19	10–13	.12 max	2 max	1 max	.045 max	.03 max		Lower Work-hardening Rate
308	19–21	10–12	.08 max	2 max	1 max	.045 max	.03 max		Welding Rod—For Ductility
309	22–24	12–15	.2 max	2 max	1 max	.045 max	.03 max		High-temp Strength and Scaling Resistance
309S	22–24	12–15	.08 max	2 max	1 max	.045 max	.03 max		Low-Carbon Type 309—For Welding
310	24–26	19–22	.25 max	2 max	1.5 max	.045 max	.03 max		Better High-temp Strength and Scaling Res.
310S	24–26	19–22	.08 max	2 max	1.5 max	.045 max	.03 max		Low-Carbon Type 310—For Welding
314	23–26	19–22	.25 max	2 max	1.5–3	.045 max	.03 max		Most Scaling Resistant
316	16–18	10–14	.08 max	2 max	1 max	.045 max	.03 max	Mo 2–3	Increased Corrosion Resistance
316L	16–18	10–14	.03 max	2 max	1 max	.045 max	.03 max	Mo 2–3	Low-carbon Type 316—For Welding
317	18–20	11–15	.08 max	2 max	1 max	.045 max	.03 max	Mo 3–4	More Corrosion Resistance than Type 316
321	17–19	9–12	.08 max	2 max	1 max	.045 max	.03 max	Ti 5xC min	Stabilized Against Carbide Precipitation
347	17–19	9–13	.08 max	2 max	1 max	.045 max	.03 max	Cb+Ta 10XC min	Stabilized Against Carbide Precipitation
348	17–19	9–13	.08 max	2 max	1 max	.045 max	.03 max	Cb+Ta 10XC min but Ta .1 max	Stabilized Against Carbide Precipitation
403	11.5–13	—	.15 max	1 max	.5 max	.04 max	.03 max		Like Type 410—"Turbine Quality"
405	11.5–14.5	—	.08 max	1 max	1 max	.04 max	.03 max	Al .1–.3	Reduced Heat-treat-hardenability
410	11.5–13.5	—	.15 max	1 max	1 max	.04 max	.03 max		General-Purpose "12 Cr"
414	11.5–13.5	1.25–2.5	.15 max	1 max	1 max	.04 max	.03 max		Better Strength than Type 410
416	12–14	—	.15 max	1.25 max	1 max	.06 max	.15 min	.6 Mo or .6 Zr optional	Free Machining "12 Cr"—Heavy Cuts
416Se	12–14	—	.15 max	1.25 max	1 max	.06 max	.06 max	Se .15 min	Free Machining "12 Cr"—Light Cuts
420	12–14	—	over .15	1 max	1 max	.04 max	.03 max		Like Type 410—For Higher Hardness
430	14–18	—	.12 max	1 max	1 max	.04 max	.03 max		General-Purpose "17 Cr"
430F	14–18	—	.12 max	1.25 max	1 max	.06 max	.15 min	.6 Mo or .6 Zr optional	Free Machining "17 Cr"—Heavy Cuts
430F Se	14–18	—	.12 max	1.25 max	1 max	.06 max	.06 max	Se .15 min	Free Machining "17 Cr"—Light Cuts
431	15–17	1.25–2.5	.2 max	1 max	1 max	.04 max	.03 max		Hardenable—High Impact Strength
440A	16–18	—	.6–.75	1 max	1 max	.04 max	.03 max	Mo .75 max	Higher Hardness than Type 420
440B	16–18	—	.75–.95	1 max	1 max	.04 max	.03 max	Mo .75 max	Higher Hardness than Type 440A
440C	16–18	—	.95–1.2	1 max	1 max	.04 max	.03 max	Mo .75 max	Higher Hardness than Type 440B
446	23–27	—	.2 max	1.5 max	1 max	.04 max	.03 max	N .25 max	Scaling Resistance at Elev Temp
501	4–6	—	over .10	1 max	1 max	.04 max	.03 max	Mo 0.40–0.65	Heat Resistant, Good Mechanical Properties at Elev Temp
502	4–6	—	.10 max	1 max	1 max	.04 max	.03 max	Mo 0.40–0.65	When Annealed, Greater Ductility, Lower Tensile Strength than Type 501

*American Iron & Steel Institute.

STANDARD DIMENSIONS FOR WELDED OR SEAMLESS STEEL PIPE—Schedule 10, 40, 80

(English and Metric Units)

Nominal Pipe Size Inches	OD Inches mm	Internal Diameter Inches mm Sch 10	Internal Diameter Inches mm Sch 40	Internal Diameter Inches mm Sch 80	Threads Per Inch	Nominal Weight lb/ft
1/8	.405 10.29	.307 7.80	.269 6.83	.215 5.46	27	0.244
1/4	.540 13.72	.410 10.41	.364 9.25	.302 7.67	18	0.424
3/8	.675 17.15	.545 13.84	.493 12.52	.423 10.74	18	0.567
1/2	.840 21.34	.674 17.12	.622 15.80	.546 13.87	14	0.850
3/4	1.050 26.67	.884 22.45	.824 20.93	.742 18.85	14	1.130
1	1.315 33.40	1.097 27.86	1.049 26.64	.957 24.31	11-1/2	1.678
1-1/4	1.660 42.16	1.442 36.63	1.380 35.05	1.278 32.46	11-1/2	2.272
1-1/2	1.900 48.26	1.682 42.72	1.610 40.89	1.500 38.10	11-1/2	2.717
2	2.375 60.33	2.157 54.79	2.067 52.50	1.939 49.25	11-1/2	3.652
2-1/2	2.875 73.03	2.635 66.93	2.469 62.71	2.323 59.00	8	5.793
3	3.500 88.90	3.260 82.80	3.068 77.93	2.900 73.66	8	7.575
4	4.500 114.3	4.260 108.2	4.026 102.3	3.826 97.18	8	10.790
6	6.625 168.3	6.357 161.5	6.065 154.1	5.761 146.3	8	18.974
8	8.625 219.1	8.329 211.6	7.981 202.7	7.625 193.7	8	28.554
10	10.750 273.1	10.420 264.7	10.020 254.5	9.564 242.9	8	40.483
12	12.750 323.9	12.390 314.7	11.938 303.2	11.376 289.0	8	43.773

WIRE TABLE, STANDARD ANNEALED COPPER*

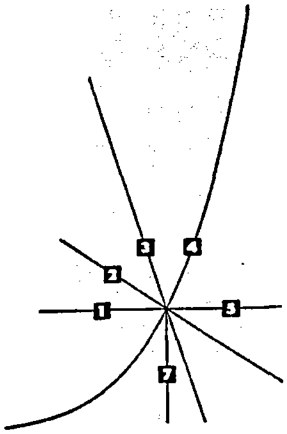
American Wire Gauge (B. & S.)
Metric and English Units

Cross Section mm²	Diameter mm	Gauge No.	Diameter Inches	Resistance Ohms/1000 ft
107.2	11.68	0000	.4600	0.04901
85.03	10.40	000	.4096	.06180
67.43	9.266	00	.3648	.07793
53.48	8.252	0	.3249	.09827
42.41	7.348	1	.2893	.1239
33.63	6.544	2	.2576	.1563
26.67	5.827	3	.2294	.1970
21.15	5.189	4	.2043	.2485
16.77	4.621	5	.1819	.3133
13.30	4.115	6	.1620	.3951
10.55	3.665	7	.1443	.4982
8.366	3.264	8	.1285	.6282
6.634	2.906	9	.1144	.7921
5.261	2.588	10	.1019	.9989
4.172	2.305	11	.09074	1.260
3.309	2.053	12	.08081	1.588
2.624	1.828	13	.07196	2.003
2.081	1.628	14	.06408	2.525
1.650	1.450	15	.05707	3.184
1.309	1.291	16	.05082	4.016
1.038	1.150	17	.04526	5.064
0.8231	1.024	18	.04030	6.385
.6527	0.9116	19	.03589	8.051
.5176	.8118	20	.03196	10.15
.4105	.7230	21	.02846	12.80
.3255	.6438	22	.02535	16.14
.2582	.5733	23	.02257	20.36
.2047	.5106	24	.02010	25.67
.1624	.4547	25	.01790	32.37
.1288	.4049	26	.01594	40.81
.1021	.3606	27	.01419	51.47
.08098	.3211	28	.01264	64.90
.06422	.2859	29	.01126	81.83
.05093	.2546	30	.01003	103.2
.04039	.2268	31	.008928	130.1
.03203	.2019	32	.007950	164.1
.02540	.1798	33	.007080	206.9
.02014	.1601	34	.006304	260.9
.01597	.1426	35	.005614	329.0
.01267	.1270	36	.005000	414.8
.01005	.1131	37	.004453	523.1
.007967	.1007	38	.003965	659.6
.006318	.08969	39	.003531	831.8
.005010	.07987	40	.003145	1049

*Dimensions and resistance at 20 C (68 F).

DECIMAL EQUIVALENTS

		1/64	.015625			17/64	.265625			33/64	.515625			49/64	.765625
	1/32		.03125		9/32		.28125		17/32		.53125		25/32		.78125
		3/64	.046875			19/64	.296875			35/64	.546875			51/64	.796875
1/16			.0625	5/16			.3125	9/16			.5625	13/16			.8125
		5/64	.078125			21/64	.328125			37/64	.578125			53/64	.828125
	3/32		.09375		11/32		.34375		19/32		.59375		27/32		.84375
		7/64	.109375			23/64	.359375			39/64	.609375			55/64	.859375
1/8			.125	3/8			.375	5/8			.625	7/8			.875
		9/64	.140625			25/64	.390625			41/64	.640625			57/64	.890625
	5/32		.15625		13/32		.40625		21/32		.65625		29/32		.90625
		11/64	.171875			27/64	.421875			43/64	.671875			59/64	.921875
3/16			.1875	7/16			.4375	11/16			.6875	15/16			.9375
		13/64	.203125			29/64	.453125			45/64	.703125			61/64	.953125
	7/32		.21875		15/32		.46875		23/32		.71875		31/32		.96875
		15/64	.234375			31/64	.484375			47/64	.734375			63/64	.984375
1/4			.25	1/2			.5	3/4			.75	1			1.



LEGEND

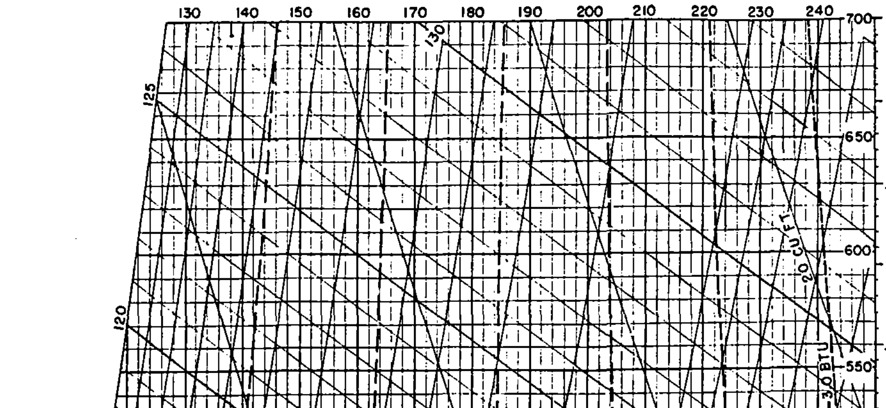
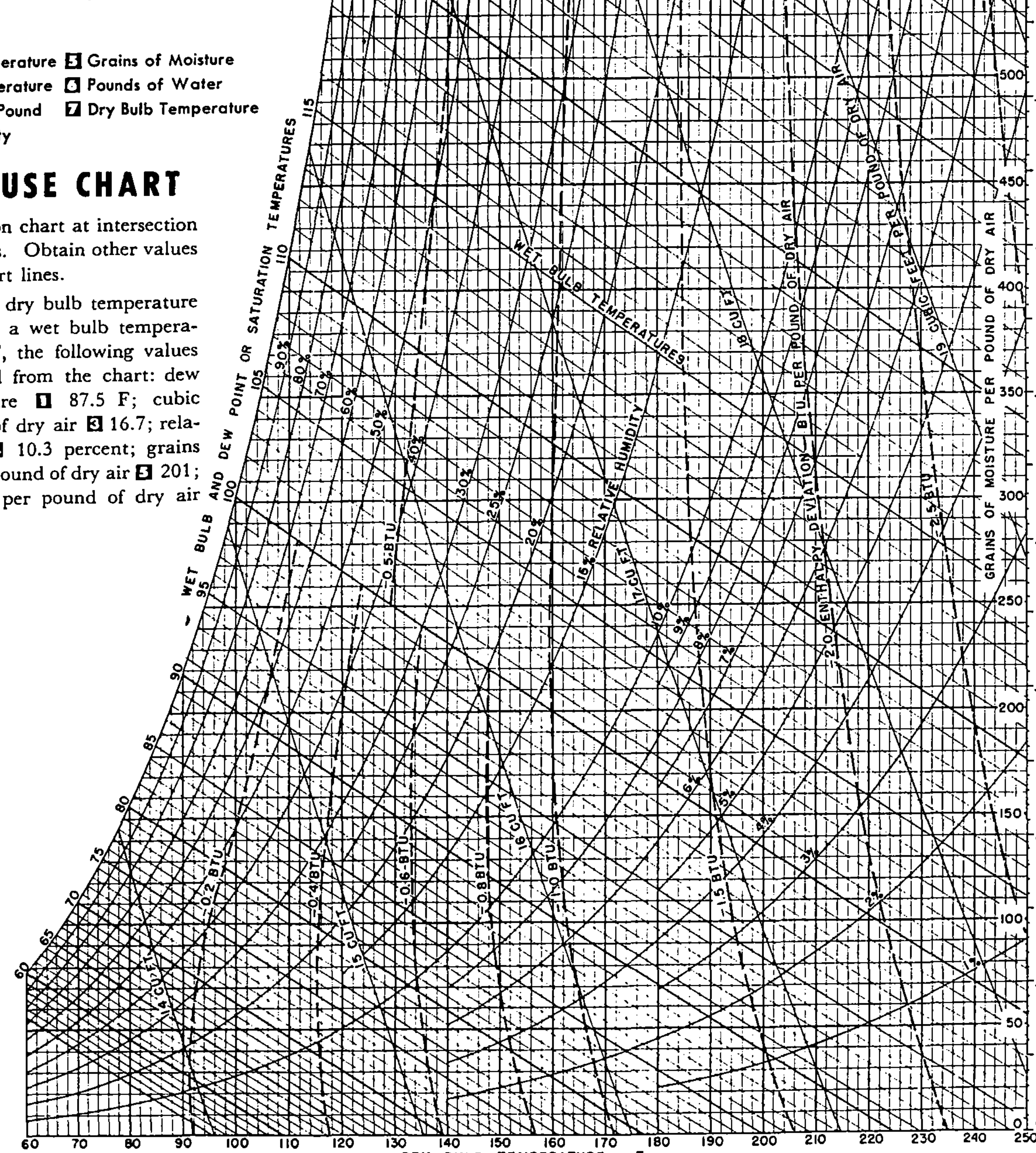
- 1 Dew Point Temperature
- 2 Wet Bulb Temperature
- 3 Cubic Feet Per Pound
- 4 Relative Humidity
- 5 Grains of Moisture
- 6 Pounds of Water
- 7 Dry Bulb Temperature

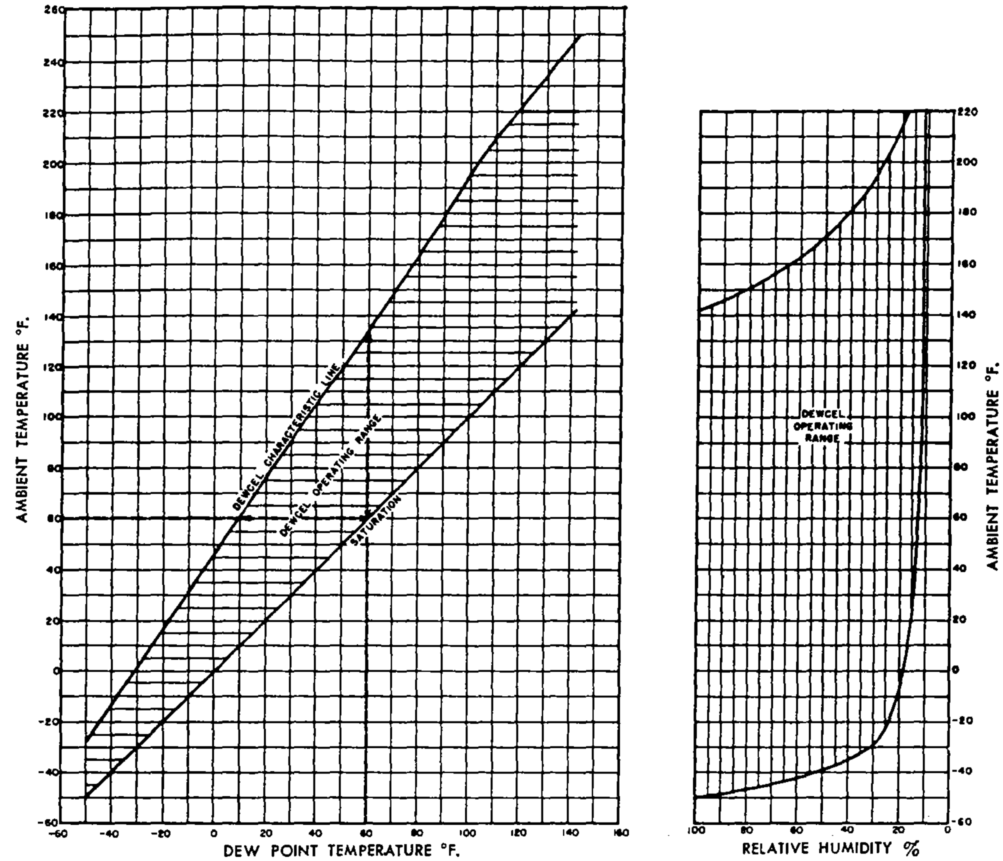
HOW TO USE CHART

Locate point on chart at intersection of any two values. Obtain other values by following chart lines.

Example: at a dry bulb temperature 7 of 172 F and a wet bulb temperature 2 of 102 F, the following values may be obtained from the chart: dew point temperature 1 87.5 F; cubic feet per pound of dry air 3 16.7; relative humidity 4 10.3 percent; grains of moisture per pound of dry air 5 201; pounds of water per pound of dry air 6 .0287.

BAROMETRIC PRESSURE 29.92 in. Hg



The Foxboro Dew Point Measuring System measures dew point temperatures between –50 F and +142 F within the limits of ambient temperature shown.

The operating range is shown in the left chart in terms of *dew point temperature*. For example, at an ambient temperature of 60 F, dew point temperatures from 60 F down to 10 F may be measured.

For convenience, the same operating range is shown in the right chart in terms of *relative humidity*. At the same ambient temperature of 60 F, used in the example above, it can be seen that the corresponding relative humidity range is 100 percent down to 13 percent.

Another example illustrates the use of the chart in determining a working range of ambient temperatures. To measure a dew point temperature of 60 F, it will be seen that the range of ambient temperatures must be between 60 F and 133 F.

Dew points at higher operating temperatures are measured by cooling a sample to a temperature within the operating range shown.

For example, with a furnace atmosphere at 1600 F temperature and 2 F dew point, the sample is cooled in a refrigeration unit to 35 F. Its dew point is then readily measured by the Dewcel element.

Blast furnace air at 300 F temperature and 70 F dew point, may be measured by passing it through an uninsulated pipe leading to a sampling chamber. Here, it is cooled to 120–130 F, enabling accurate dew point measurement.

Both the Sampling Chamber and Refrigeration Unit can be furnished by The Foxboro Company for most applications.

Readings are easily converted to:

ABSOLUTE HUMIDITY. The most accurate method of determining the moisture content of air is to measure its dew point. Dew point values are readily converted to absolute humidity units or vapor pressure units by reference to standard tables. Instruments reading absolute humidity directly are also available.

RELATIVE HUMIDITY. A two-pen Recorder can be furnished if a relative humidity measurement is desired. One pen records dew point; the second pen records dry-bulb temperature. A table or chart is used to convert these readings to relative humidity.

Foxboro Wet- and Dry-Bulb instruments, and direct-reading Relative Humidity instruments of the hair type are also available.

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TAB 9, Rev. 1

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CONTAINMENT VOLUME
REV. 0

CONTAINMENT VOLUME = 3.74 E10 cm^3

= 1.32 E6 ft^3

Reference: USAR 5.2.1.1

Introduction to Health Physics

BY

HERMAN CEMBER
Northwestern University

PERGAMON PRESS
OXFORD · LONDON · EDINBURGH · NEW YORK
TORONTO · SYDNEY · PARIS · BRAUNSCHWEIG

Effective Half-life

The total dose absorbed during any given time interval after the uptake of the iodine in the thyroid may be calculated by integrating the dose rate over the required time interval. In making this calculation, two factors must be considered, viz.

1. *In situ* radioactive decay of the isotope.
2. Biological elimination of the isotope.

In most instances, biological elimination follows first-order kinetics. In this case, the equation for the quantity of radioisotope within an organ at any time t after uptake of a quantity Q_0 is given by

$$Q = (Q_0\, e^{-\lambda_R t})\,(e^{-\lambda_B t}), \tag{6.13}$$

where λ_R is the radioactive decay constant, and λ_B is the biological elimination constant. The two exponentials in equation (6.13) may be combined

$$Q = Q_0\, e^{-(\lambda_R+\lambda_B)t}, \tag{6.14}$$

and, if $\lambda_E = \lambda_R + \lambda_B$, we have (6.15)

$$Q = Q_0\, e^{-\lambda_E t}, \tag{6.16}$$

where λ_E is called the *effective* elimination constant. The effective half-life then is

$$T_E = \frac{0.693}{\lambda_E}. \tag{6.17}$$

From the relationship among λ_E, λ_R, and λ_B, we have

$$\frac{1}{T_E} = \frac{1}{T_R} + \frac{1}{T_B}, \tag{6.18}$$

or

$$T_E = \frac{T_R \times T_B}{T_R + T_B}. \tag{6.19}$$

For ^{131}I, $T_R = 8$ days and T_B, the biological half-life in the thyroid, is 180 days. The effective half-life, therefore, is

$$T_E = \frac{8 \times 180}{8 + 180} = 7.7 \text{ days},$$

and the effective elimination constant is

$$\lambda_E = \frac{0.693}{7.7} = 0.09 \text{ day}^{-1}.$$

TAB 11, Rev. 0

Dose Due to Total Decay

The dose, dD, during an infinitesimally small time period, dt, at a time interval t after an initial dose rate D_0 is

$$\begin{aligned} dD &= \text{instantaneous dose rate} \times dt \\ &= D_0\, e^{-\lambda_E t}\, dt, \end{aligned} \tag{6.20}$$

where D_0 is the dose rate at time $t = 0$, the instant when the isotope was taken up by the tissue; and t is the elapsed time after uptake. The total dose during the time interval T after uptake of the isotope is

$$D = D_0 \int_0^T e^{-\lambda_E t}\, dt, \tag{6.21}$$

which, when integrated, yields

$$D = \frac{D_0}{\lambda_E}(1 - e^{-\lambda_E t}). \tag{6.22}$$

For an infinitely long time, that is, when the isotope is completely gone, equation (6.22) reduces to

$$D = \frac{D_0}{\lambda_E}. \tag{6.23}$$

It should be noted that the dose due to total decay is merely equal to the product of the initial dose rate, D_0, and the average life of the radioisotope within the organ, $1/\lambda_E$. For the case in Example 6.5, the total absorbed dose during the first 5 days after deposition of the radioiodine in the thyroid is, according to equation (6.22), and after substituting $\lambda_E = 0.693/T_E$,

$$\begin{aligned} D &= \frac{31.9 \text{ rad/hr} \times 24 \text{ hr/day} \times 7.7 \text{ day}}{0.693}\left(1 - \exp - \frac{0.693}{7.7} \times 5\right) \\ &= 3090 \text{ rads}, \end{aligned}$$

and the dose due to complete decay is, from equation (6.22),

$$D = \frac{31.9 \text{ rad/hr} \times 24 \text{ hr/day} \times 7.7 \text{ day}}{0.693} = 8520 \text{ rads}.$$

Gamma Emitters

For a uniformly distributed gamma emitting isotope, the dose rate at any point P due to the isotope in the infinitesimal volume dV at any other point at a distance r from point p, as shown in Fig. 6.7, is

$$dD = C\,\Gamma \frac{e^{-\mu r}}{r^2}\, dV \text{ rad/hr}, \tag{6.24}$$

using the half-face respirator, the wearer must try to eliminate possible leaks around the face piece. Half-face respirators are considered suitable for air-borne dust concentrations up to ten (10) times the recommended maximum atmospheric concentration; respirators with full face masks are considered suitable up to fifty (50) times the recommended maximum atmospheric concentration.

2. Supplied air masks that may be used either against dusts or gases or both. In this category we have two subdivisions: (a) air line hoods, which utilize uncontaminated air, under positive (with respect to the atmosphere) pressure supplied from a remote source, and (b) self-contained breathing apparatus, in which breathing air is supplied either from a bottle carried by the man, or from a cannister containing oxygen generating chemicals. The advantage of the supplied air device is that the pressure in the breathing zone is higher than atmospheric pressure. As a consequence, all leakage is from the inside out. When using a supplied air device, it is imperative to know the time limitation on the supply of air.

A third type of respiratory protective device is the gas-mask. In this device, contaminated air is cleared by chemicals in a cannister through which the air passes. Because of the specific action of the chemical agents on the contaminant, different cannisters must be used for different gases. For this reason, as well as for the fact that air may leak into the face-piece of a gas-mask, gas-masks are not recommended for use against radioactive gas.

Surface Contamination Limits

Contamination of personnel and/or equipment may occur either from normal operations or as a result of the breakdown of protective measures. An exact quantitative definition of contamination that would be applicable in all situations cannot be given. Generally, contamination means the presence of undesirable radioactivity—undesirable either in the context of health or for technical reasons, such as increased background, interference with tracer studies, etc. In this discussion, only the health aspects of contamination are considered.

Surface contamination falls into two categories, fixed and loose. In the case of fixed contamination, the radioactivity cannot be transmitted to personnel, and the hazard, consequently, is that of external radiation. For fixed contamination, therefore, the degree of acceptable contamination is directly related to the external radiation dose rate. Setting a maximum limit for fixed surface contamination thus becomes a relatively simple matter. The hazard from loose surface contamination arises mainly from the possibility of tactile transmission of the radioactive contaminant to the mouth or to the skin, or of resuspending the contaminant and then inhaling it. It follows that the

TAB 11, Rev. 0

degree of hazard from surface contamination is strongly dependent on the degree to which the contaminant is fixed to the surface.

Dealing with loose surface contamination limits is not as straightforward as dealing with contamination of air and water. In the case of air and water contamination, safety standards can be easily set—at least in theory—on the basis of recommended maximum absorbed doses (Dose = 5 (N-18), etc.). Using these criteria, we can calculate maximum permissible body burdens for each of the radioisotopes. From the calculated body burden we go one step further from the basic radiation safety criteria, and compute maximum concentrations in air and water which, if continuously inhaled or ingested, would result in a body burden less than the calculated maximum. For the case of surface contamination, we go one more step away from the basic criteria; we try to estimate the surface contamination which, if it should be dispersed into the environment, would result in concentrations that may lead to an excessive body burden. Thus, specification of limits for loose surface contamination is three steps removed from the basic safety requirements.

From the foregoing discussion, it is clear that maximum limits for surface contamination cannot be fixed in the same sense as limits for the concentration of radionuclides in air and water. Nevertheless, it is useful to compute a number that may serve as a guide in the evaluation of the hazard to workers from surface contamination, and to assist the health physicist in deciding whether or not to require the use of special protective measures for workers in contaminated areas.

On the basis of per-unit-quantity of radioactivity, inhalation is considered the most serious route of exposure. Surface contamination, therefore, is usually limited by the inhalation hazard that may arise from resuspension of the contaminant. The quantitative relationship between the concentration of loose surface contamination and consequent atmospheric concentration above the contaminated surface due to stirring up the surface is called the *resuspension factor*, f_r, and is defined by

$$f_r = \frac{\text{atmospheric concentration, } \mu\text{Ci/cm}^3}{\text{surface concentration } \mu\text{Ci/cm}^2}. \qquad (11.2)$$

Experimental investigation of the resuspension of loose surface contamination shows that the resuspension factor varies from about 10^{-4} to 10^{-8}, depending on the conditions under which the studies were conducted. A value of 10^{-6} is reasonable for the purpose of estimating the hazard from surface contamination.

Example 11.2

Estimate the maximum surface contamination of ^{90}Sr dust that may be allowed before taking special safety measures to protect personnel against a contamination hazard.

The maximum atmospheric concentration of ^{90}Sr recommended by the ICRP is 2×10^{-10} μCi/cm³. Using a value of 10^{-6} cm^{-1} for the resuspension factor in equation (11.2), we have

$$10^{-6}\ \text{cm}^{-1} = \frac{2 \times 10^{-10}\ \mu\text{Ci/cm}^3}{\text{surface concentration}}$$

$$\therefore \text{surface concentration} = 2 \times 10^{-4}\ \mu\text{Ci/cm}^2.$$

It should be emphasized that a figure for loose surface contamination calculated by the method of Example 11.2 is intended only as a guide. In any particular case, the health physicist may, at his discretion, and depending on the nature of the operation, the degree of ventilation, and other relevant factors, insist on more or less stringent requirements for surface contamination before requiring the use of protective devices for the worker.

TABLE 11.2. UNITED KINGDOM ATOMIC ENERGY AUTHORITY MAXIMUM PERMISSIBLE LEVELS OF SURFACE CONTAMINATION, μCi/cm²(a)

Type of surface	Principal alpha emitters(b)	Low toxicity alpha emitters(c)	Beta emitters
Inactive and low activity areas	10^{-5}	10^{-4}	10^{-4}
Active areas	10^{-4}	10^{-3}	10^{-3}
Personal clothing	10^{-5}	10^{-4}	10^{-4}
Authority clothing not normally worn in active areas	10^{-4}	10^{-3}	10^{-3}
Skin	10^{-6}	10^{-5}	10^{-4}

The contamination of surfaces by radioactive materials may give rise to external radiation or to intake of radioactive materials by persons. The control of surface contamination is, therefore, an essential part of the safe handling of radioactive materials. Surface contamination should be controlled to the lowest practicable levels, and in any case, within the maximum permissible levels specified above, unless relaxations (e.g. for firmly fixed contamination or low toxicity contaminants) have been permitted on the advice of the health physicist. The requirements of the maximum permissible doses from external radiation must be observed.

(a) Averaging is permitted over inanimate areas of up to 300 cm² or, for floors, walls, and ceilings, 1000 cm². Averaging is permitted over 100 cm² for skin; for the hands, over the whole area of one hand, nominally 300 cm².

(b) All alpha emitters other than those listed in note (c).

(c) Uranium isotopes.
Enriched and depleted uranium.
Natural uranium.
Natural thorium.
Short-lived radionuclides, such as radon daughters.
Thorium-232.
Thorium-228 and thorium-230 when diluted to a specific activity of the same order as that of natural uranium and natural thorium.

(From the U.K.A.E.A. Health and Safety Code, *Maximum Permissible Doses from Inhaled and Ingested Radioactive Materials*, Authority Code No. E.1.2, Issue No. 1, London, June 1961.)

Various laboratories and nuclear installations have set their own limits for contamination of personnel, equipment, and protective clothing. Tables 11.2 and 11.3 are given to illustrate some of the contamination standards maintained by several large users of radioisotopes.

TABLE 11.3. U.S.S.R. SURFACE CONTAMINATION LIMITS

Object of contamination	Contamination from 150 cm² in 1 min			
	Alpha particles		Beta particles	
	Before cleaning	After cleaning	Before cleaning	After cleaning
Hands	75	bg	5000	bg
Special linens and towels	75	bg	5000	bg
Cotton special work clothes	500	100	25,000	5000
"Pellicular" clothing	500	200	25,000	10,000
Gloves, outside	500	100	25,000	5000
Special shoes, outside	500	200	25,000	5000
Work surfaces and equipment	500	200	25,000	5000

Note. No contamination of the body is permitted.

(From *Sanitary Regulations for Work with Radioactive Substances and Sources of Ionizing Radiation*, Ministry of Health, U.S.S.R., Moscow, 1960.)

Waste Disposal

Proper collection and disposal of radioactive waste is an inherent part of contamination control and internal radiation protection. In one sense, we cannot dispose of radioactive waste. All other types (non-radioactive) of toxic wastes can be treated either chemically, physically, or biologically in such a manner as to reduce their toxicity. In the case of radioactive wastes, on the other hand, nothing can be done to decrease the radioactivity, and hence, the inherent toxicity of the waste. The only means of ultimate disposal is time—to allow for the decay of the radioactivity.

Radioactive wastes originate from any operation in which radioisotopes are used or produced. For purposes of management and treatment, wastes may be classified as high, intermediate, and low level. Low-level wastes are defined as those that must be diluted by a factor of no more than 10^3 before discharge into the environment, for intermediate levels, $10^3 < DF \leq 10^5$, and for high-level waste, the required dilution factor would be greater than 10^5. High-level wastes are associated with the inventory of fission products in the burned-up fuel of nuclear reactors and with the chemical and metallurgical processes involved in the separation of the fission products from the unspent uranium or plutonium in the burned-up fuel.

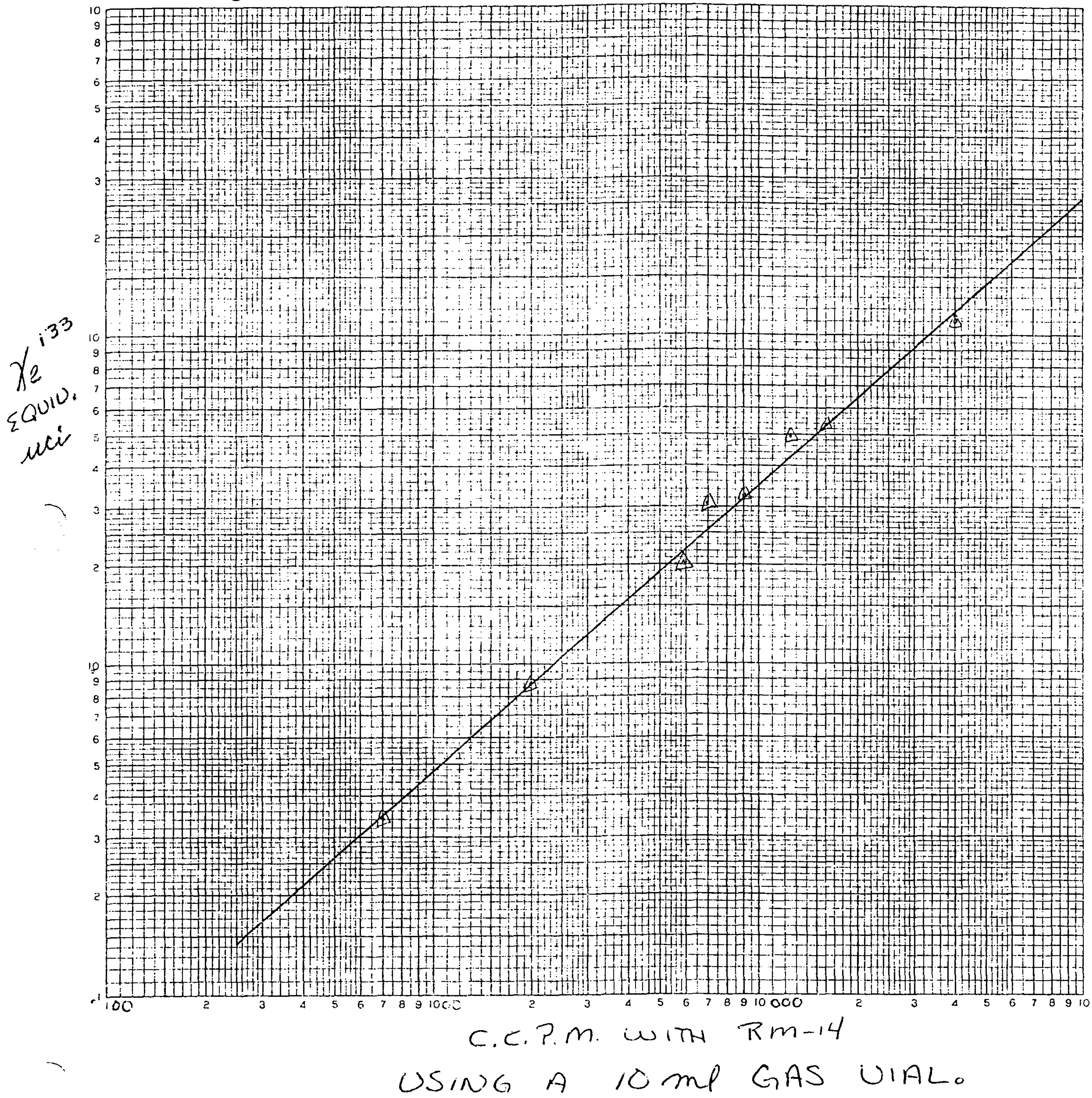
APPENDIX C EFFECTIVE HALF-LIVES

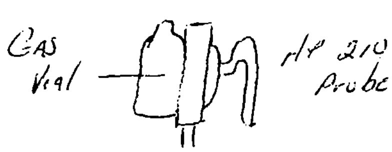
Radionuclide	T_b	T_r	T_e
^{3}H	19 d	12.26 y	19 d
^{7}Be	400 d[b]	53 d	47 d
^{14}C	35 d	5770 y	35 d
^{18}F	140 d[b]	1.85 h	1.8 h
^{24}Na	19 d	15 h	15 h
^{32}P	1200 d	14.3 d	14 d
^{40}K	37 d	1.3×10^{9} y	37 d
^{45}Ca	50 y	165 d	165 d
^{51}Cr	110 d	27.8 d	22 d
$^{48}V^{V}$	50 d[b]	16.1 d	12 d
^{65}Zn (23 d[b])	933 d	245 d	111 d
^{72}Ga	8.3 y[b]	14.1 h	14 h
^{90}Sr	11 y[b]	28 y	7.9 y
^{90}Y	500 d[b]	64.2 h	64 h
^{95}Zr	180 d[b]	65 d	48 d
^{95}Nb	50 d[b]	35 d	21 d
^{99}Mo	150 d[b]	2.75 d	2.7 d
^{106}Ru	20 d	1.0 y	19 d
^{113}Sn	149 d[b]	118 d	66 d
^{131}I	138 d[b]	8.05 d	7.6 d
^{137}Cs	17 d	30 y	17 d
^{140}Ba	200 d	12.8 d	12 d
^{140}La	35 d[b]	40.2 h	1.6 d
^{144}Ce	500 d[b]	285 d	180 d
^{143}Pr	50 d[b]	13.7 d	11 d
^{147}Nd	35 d[b]	11.1 d	8.4 d
^{147}Pm	100 d[b]	2.5 y	90 d
^{151}Sm	110 y[b]	90 y	50 y
^{154}Eu	4 y[b]	16 y	3.2 y
^{166}Ho	37 d[b]	27 h	1.1 d
^{170}Tm	110 d[b]	127 d	59 d
^{177}Lu	6 d	6.8 d	3.2 d
^{181}W	5 d[b]	130 d	4.8 d
^{210}Pb	531 d	21 y	1.4 y
^{210}Po	57 d	138.4 d	40 d
^{226}Ra + 55% da	55 y	1620 y	53 y
^{227}Ac	33 y[b]	21.2 y	13 y
^{234}Th–^{234}Pa	110 y[b]	24.1 d	24 d
^{233}U (30 d)	300 d	1.62×10^{5} y	300 d
^{239}Pu			
Sol./bone	120 y	2.44×10^{4} y	119 y
Insol./lungs	1.0 y	2.44×10^{4} y	1.0 y
^{241}Am	890 d[b]	458 y	890 d
^{242}Cm	600 d[b]	163 d	130 d

[a] From M. Errera and A. Forssberg, eds., *Mechanisms in Radiobiology*, Vol. 2, Academic Press, New York, 1960; and K. Z. Morgan and J. E. Turner, *Principles of Radiation Protection*, John Wiley & Sons, Inc., New York, 1967.

[b] Critical organ.

Xe^{133} EQUIVILANT ACTIVITY CHART





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DAVID E. HAMERSKI, PH.D.

PHYSICIST

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RR #2
Welch, MN 55089

11/11/81

From: D. Hamerski

Subject: CONTAINMENT HIGH-RANGE RADIATION MONITOR DOSE ASSESSMENT FACTORS BASED UPON Xe-133 AND I-131 EQUIVALENTS

Background: The calculated data presented here are based upon the following assumptions:

(1) That the monitor views a large, uniform fraction of the containment volume.

(2) That the monitor is not placed in an area which is protected by massive shielding.

(3) The source terms consist of the Owners Group source terms-- 100% noble gases, 50% iodines.

(4) That the monitor is basically a 3 inch diameter ion chamber.

(5) That the net free containment volume is 3.73×10^{10} cc.

This report supercedes all prior reports.

DAVID E. HAMERSKI, PH.D.
PHYSICIST

Results: I. Xe-133 Equivalent Data

The dose-factor for Xe-133 equivalent was found to be:

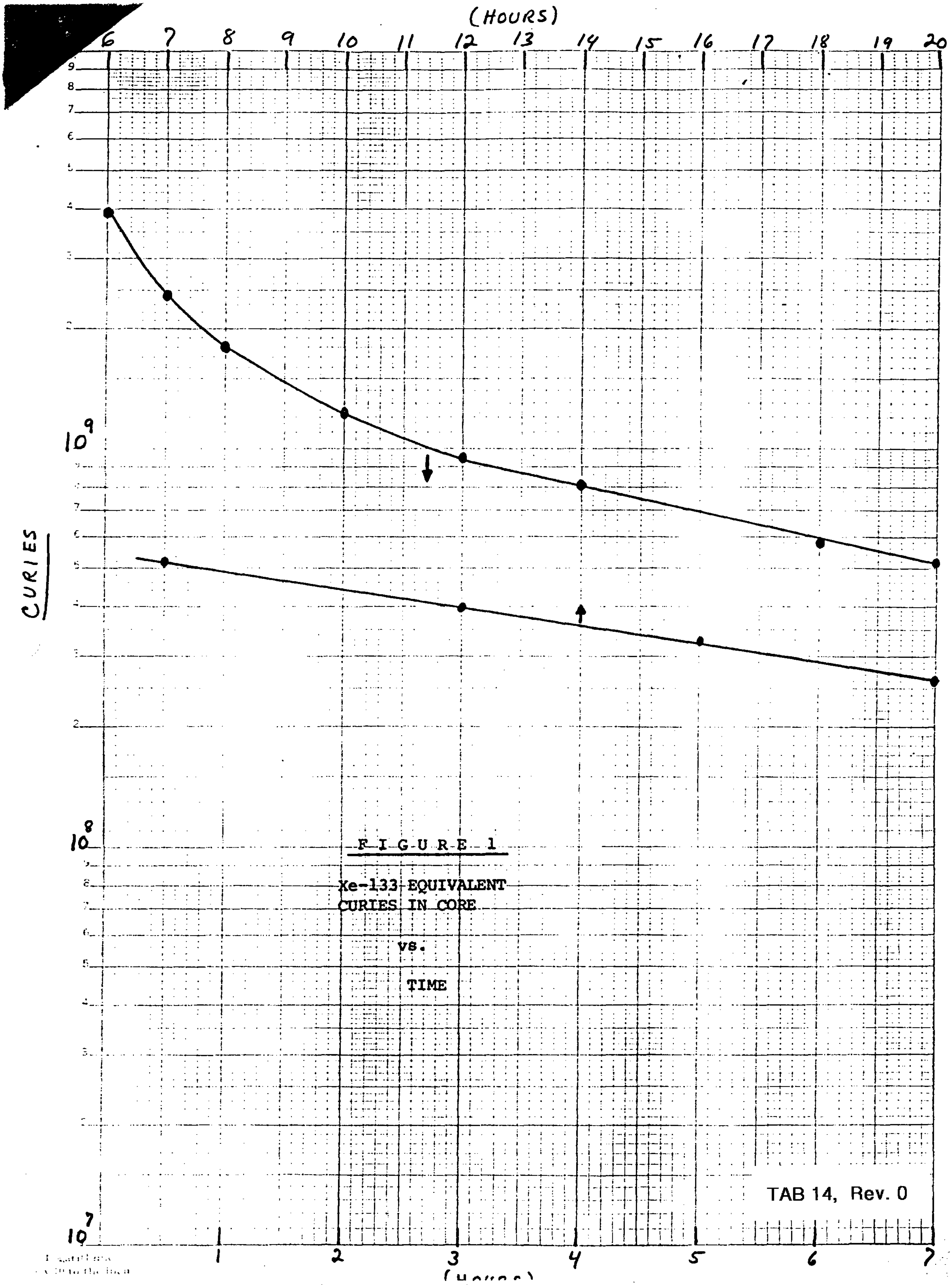
$$DF = 1.2 \times 10^4 \ (mr/hr)/(\mu Ci/cc)$$

The curve of Xe-133 equivalent versus time is given as Figure 1 and Figure 2.

II. I-131 Equivalent Data

(a) The dose-factor for I-131 equivalent versus time is given as Figure 3.

(b) The curve of I-131 equivalent versus time is given as Figure 4.



F I G U R E 1

Xe-133 EQUIVALENT CURIES IN CORE

vs.

TIME

TAB 14, Rev. 0

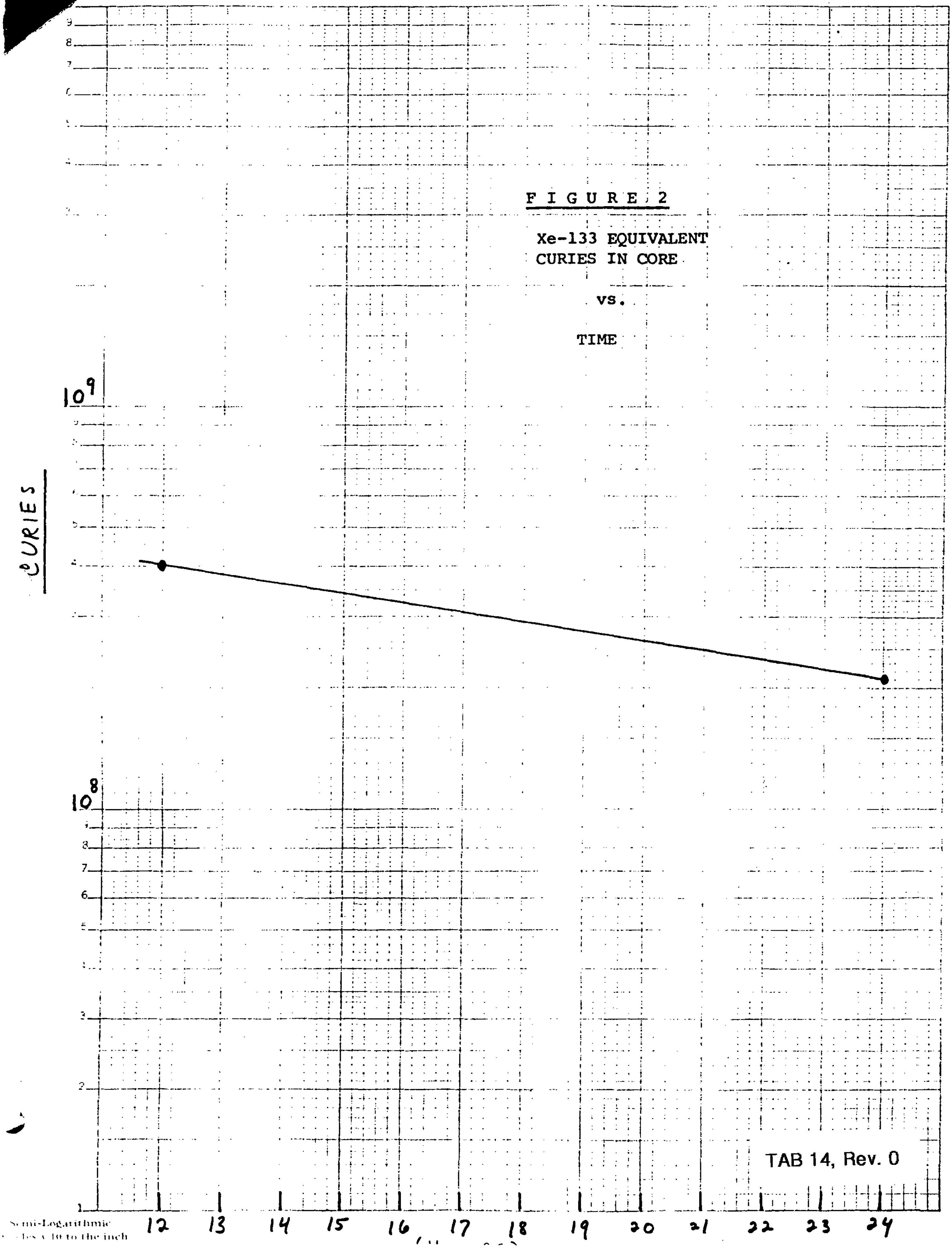


FIGURE 2
Xe-133 EQUIVALENT
CURIES IN CORE
vs.
TIME
CURIES
10^9
10^8
12
13
14
15
16
17
18
19
20
21
22
23
24
Semi-Logarithmic

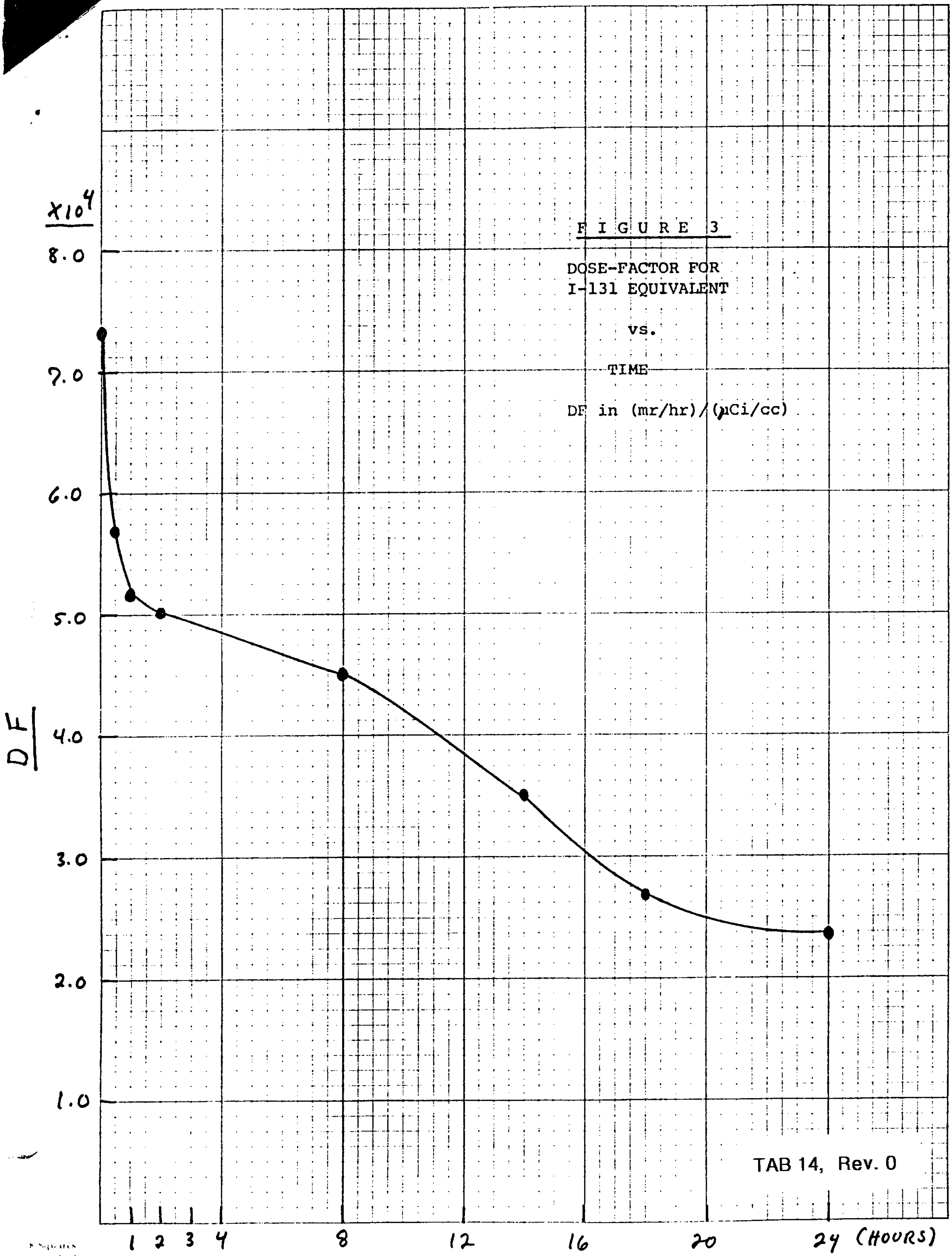


FIGURE 3
DOSE-FACTOR FOR
I-131 EQUIVALENT
vs.
TIME
DF in (mr/hr)/(μCi/cc)
x10^4
8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
DF
1
2
3
4
8
12
16
20
24
(HOURS)

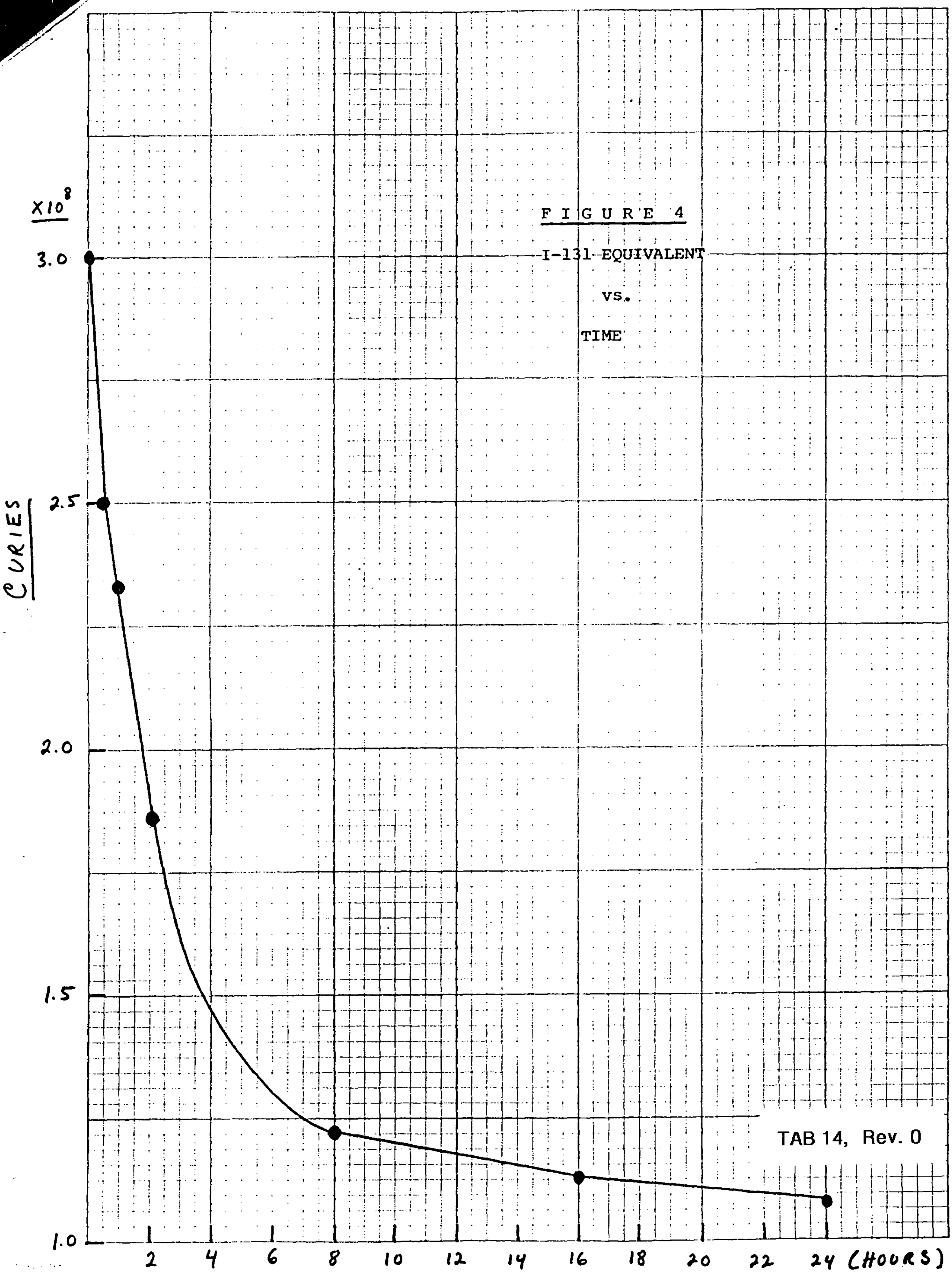


FIGURE 4
I-131 EQUIVALENT
vs.
TIME
X10^8
CURIES
3.0
2.5
2.0
1.5
1.0
2
4
6
8
10
12
14
16
18
20
22
24 (HOURS)

TAB 14, Rev. 0

DELETED

TAB 15

DELETED

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Rev. 26

U.S. NUCLEAR REGULATORY COMMISSION

Revision 1
February 1996

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 8.29

(Draft was issued as DG-8012)

INSTRUCTION CONCERNING RISKS FROM OCCUPATIONAL RADIATION EXPOSURE

A. INTRODUCTION

Section 19.12 of 10 CFR Part 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations," requires that all individuals who in the course of their employment are likely to receive in a year an occupational dose in excess of 100 mrem (1 mSv) be instructed in the health protection issues associated with exposure to radioactive materials or radiation. Section 20.1206 of 10 CFR Part 20, "Standards for Protection Against Radiation," requires that before a planned special exposure occurs the individuals involved are, among other things, to be informed of the estimated doses and associated risks.

This regulatory guide describes the information that should be provided to workers by licensees about health risks from occupational exposure. This revision conforms to the revision of 10 CFR Part 20 that became effective on June 20, 1991, to be implemented by licensees no later than January 1, 1994. The revision of 10 CFR Part 20 establishes new dose limits based on the effective dose equivalent (EDE), requires the summing of internal and external dose, establishes a requirement that licensees use procedures and engineering controls to the extent practicable to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA), provides for planned special exposures, establishes a dose limit for the embryo/fetus of an occupationally exposed declared pregnant woman, and explicitly states that Part 20 is not to be construed as limiting action that may be necessary to protect health and safety during emergencies.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 19 or 10 CFR Part 20. These regulations provide the regulatory bases for this guide. The information collection requirements in 10 CFR Parts 19 and 20 have been cleared under OMB Clearance Nos. 3150-0044 and 3150-0014, respectively.

B. DISCUSSION

It is important to qualify the material presented in this guide with the following considerations.

The coefficient used in this guide for occupational radiation risk estimates, 4 x 10^{-4} health effects per rem, is based on data obtained at much higher doses and dose rates than those encountered by workers. The risk coefficient obtained at high doses and dose rates was reduced to account for the reduced effectiveness of lower doses and dose rates in producing the stochastic effects observed in studies of exposed humans.

The assumption of a linear extrapolation from the lowest doses at which effects are observable down to

the occupational range has considerable uncertainty. The report of the Committee on the Biological Effects of Ionizing Radiation (Ref. 1) states that

> "... departure from linearity cannot be excluded at low doses below the range of observation. Such departures could be in the direction of either an increased or decreased risk. Moreover, epidemiologic data cannot rigorously exclude the existence of a threshold in the 100 mrem dose range. Thus, the possibility that there may be no risk from exposures comparable to external natural background radiation cannot be ruled out. At such low doses and dose rates, it must be acknowledged that the lower limit of the range of uncertainty in the risk estimates extends to zero."

The issue of beneficial effects from low doses, or hormesis, in cellular systems is addressed by the United Nations Scientific Committee on the Effects of Atomic Radiation (Ref. 2). UNSCEAR states that "... it would be premature to conclude that cellular adaptive responses could convey possible beneficial effects to the organism that would outweigh the detrimental effects of exposures to low doses of low-LET radiation."

In the absence of scientific certainty regarding the relationship between low doses and health effects, and as a conservative assumption for radiation protection purposes, the scientific community generally assumes that any exposure to ionizing radiation can cause biological effects that may be harmful to the exposed person and that the magnitude or probability of these effects is directly proportional to the dose. These effects may be classified into three categories:

Somatic Effects: Physical effects occurring in the exposed person. These effects may be observable after a large or acute dose (e.g., 100 rems[1] (1 Sv) or more to the whole body in a few hours); or they may be effects such as cancer that may occur years after exposure to radiation.

Genetic Effects: Abnormalities that may occur in the future children of exposed individuals and in subsequent generations (genetic effects exceeding normal incidence have not been observed in any of the studies of human populations).

Teratogenic Effects: Effects such as cancer or congenital malformation that may be observed in children who were exposed during the fetal and embryonic stages of development (these effects have been observed from high, i.e., above 20 rems (0.2 Sv), acute exposures).

The normal incidence of effects from natural and manmade causes is significant. For example, approximately 20% of people die from various forms of cancer whether or not they ever receive occupational exposure to radiation. To avoid increasing the incidence of such biological effects, regulatory controls are imposed on occupational doses to adults and minors and on doses to the embryo/fetus from occupational exposures of declared pregnant women.

Radiation protection training for workers who are occupationally exposed to ionizing radiation is an essential component of any program designed to ensure compliance with NRC regulations. A clear understanding of what is presently known about the biological risks associated with exposure to radiation will result in more effective radiation protection training and should generate more interest on the part of the workers in complying with radiation protection standards. In addition, pregnant women and other occupationally exposed workers should have available to them relevant information on radiation risks to enable them to make informed decisions regarding the acceptance of these risks. It is intended that workers who receive this instruction will develop respect for the risks involved, rather than excessive fear or indifference.

C. REGULATORY POSITION

Instruction to workers performed in compliance with 10 CFR 19.12 should be given prior to occupational exposure and periodically thereafter. The frequency of retraining might range from annually for licensees with complex operations such as nuclear power plants, to every three years for licensees who possess, for example, only low-activity sealed sources. If a worker is to participate in a planned special exposure, the worker should be informed of the associated risks in compliance with 10 CFR 20.1206.

In providing instruction concerning health protection problems associated with exposure to radiation, all occupationally exposed workers and their supervisors should be given specific instruction on the risk of biological effects resulting from exposure to radiation. The extent of these instructions should be commensurate with the radiological risks present in the workplace.

The instruction should be presented orally, in printed form, or in any other effective communication media to workers and supervisors. The appendix to this guide provides useful information for demonstrating compliance with the training requirements in 10 CFR Parts 19 and 20. Individuals should be given an opportunity to discuss the information and to ask questions. Testing is recommended, and each trainee should be asked to acknowledge in writing that the instruction has been received and understood.

[1] In the International System of Units (SI), the rem is replaced by the sievert; 100 rems is equal to 1 sievert (Sv).

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

Except in those cases in which an applicant or licensee proposes acceptable alternative methods for complying with specified portions of the Commission's regulations, the guidance and instructional materials in this guide will be used in the evaluation of applications for new licenses, license renewals, and license amendments and for evaluating compliance with 10 CFR 19.12 and 10 CFR Part 20.

REFERENCES

1. National Research Council, *Health Effects of Exposure to Low Levels of Ionizing Radiation*, Report of the Committee on the Biological Effects of Ionizing Radiation (BEIR V), National Academy Press, Washington, DC, 1990.

2. United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), *Sources and Effects of Ionizing Radiation*, United Nations, New York, 1993.

APPENDIX

INSTRUCTION CONCERNING RISKS FROM OCCUPATIONAL RADIATION EXPOSURE

This instructional material is intended to provide the user with the best available information about the health risks from occupational exposure to ionizing radiation. Ionizing radiation consists of energy or small particles, such as gamma rays and beta and alpha particles, emitted from radioactive materials, which can cause chemical or physical damage when they deposit energy in living tissue. A question and answer format is used. Many of the questions or subjects were developed by the NRC staff in consultation with workers, union representatives, and licensee representatives experienced in radiation protection training.

This Revision 1 to Regulatory Guide 8.29 updates the material in the original guide on biological effects and risks and on typical occupational exposure. Additionally, it conforms to the revised 10 CFR Part 20, "Standards for Protection Against Radiation," which was required to be implemented by licensees no later than January 1, 1994. The information in this appendix is intended to help develop respect by workers for the risks associated with radiation, rather than unjustified fear or lack of concern. Additional guidance concerning other topics in radiation protection training is provided in other NRC regulatory guides.

1. What is meant by health risk?

A health risk is generally thought of as something that may endanger health. Scientists consider health risk to be the statistical probability or mathematical chance that personal injury, illness, or death may result from some action. Most people do not think about health risks in terms of mathematics. Instead, most of us consider the health risk of a particular action in terms of whether we believe that particular action will, or will not, cause us some harm. The intent of this appendix is to provide estimates of, and explain the bases for, the risk of injury, illness, or death from occupational radiation exposure. Risk can be quantified in terms of the probability of a health effect per unit of dose received.

When x-rays, gamma rays, and ionizing particles interact with living materials such as our bodies, they may deposit enough energy to cause biological damage. Radiation can cause several different types of events such as the very small physical displacement of molecules, changing a molecule to a different form, or ionization, which is the removal of electrons from atoms and molecules. When the quantity of radiation energy deposited in living tissue is high enough, biological damage can occur as a result of chemical bonds being broken and cells being damaged or killed. These effects can result in observable clinical symptoms.

The basic unit for measuring absorbed radiation is the rad. One rad (0.01 gray in the International System of units) equals the absorption of 100 ergs (a small but measurable amount of energy) in a gram of material such as tissue exposed to radiation. To reflect biological risk, rads must be converted to rems. The new international unit is the sievert (100 rems = 1 Sv). This conversion accounts for the differences in the effectiveness of different types of radiation in causing damage. The rem is used to estimate biological risk. For beta and gamma radiation, a rem is considered equal to a rad.

2. What are the possible health effects of exposure to radiation?

Health effects from exposure to radiation range from no effect at all to death, including diseases such as leukemia or bone, breast, and lung cancer. Very high (100s of rads), short-term doses of radiation have been known to cause prompt (or early) effects, such as vomiting and diarrhea,[1] skin burns, cataracts, and even death. It is suspected that radiation exposure may be linked to the potential for genetic effects in the children of exposed parents. Also, children who were exposed to high doses (20 or more rads) of radiation prior to birth (as an embryo/fetus) have shown an increased risk of mental retardation and other congenital malformations. These effects (with the exception of genetic effects) have been observed in various studies of medical radiologists, uranium miners, radium workers, radiotherapy patients, and the people exposed to radiation from atomic bombs dropped on Japan. In addition, radiation effects studies with laboratory animals, in which the animals were given relatively high doses, have provided extensive data on radiation-induced health effects, including genetic effects.

It is important to note that these kinds of health effects result from high doses, compared to occupational levels, delivered over a relatively short period of time.

Although studies have not shown a consistent cause-and-effect relationship between current levels of occupational radiation exposure and biological effects, it is prudent from a worker protection perspective to assume that some effects may occur.

[1]These symptoms are early indicators of what is referred to as the acute radiation syndrome, caused by high doses delivered over a short time period, which includes damage to the blood-forming organs such as bone marrow, damage to the gastrointestinal system, and, at very high doses, can include damage to the central nervous system.

3. What is meant by early effects and delayed or late effects?

EARLY EFFECTS

Early effects, which are also called immediate or prompt effects, are those that occur shortly after a large exposure that is delivered within hours to a few days. They are observable after receiving a very large dose in a short period of time, for example, 300 rads (3 Gy) received within a few minutes to a few days. Early effects are not caused at the levels of radiation exposure allowed under the NRC's occupational limits.

Early effects occur when the radiation dose is large enough to cause extensive biological damage to cells so that large numbers of cells are killed. For early effects to occur, this radiation dose must be received within a short time period. This type of dose is called an acute dose or acute exposure. The same dose received over a long time period would not cause the same effect. Our body's natural biological processes are constantly repairing damaged cells and replacing dead cells; if the cell damage is spread over time, our body is capable of repairing or replacing some of the damaged cells, reducing the observable adverse conditions.

For example, a dose to the whole body of about 300–500 rads (3–5 Gy), more than 60 times the annual occupational dose limit, if received within a short time period (e.g., a few hours) will cause vomiting and diarrhea within a few hours; loss of hair, fever, and weight loss within a few weeks; and about a 50 percent chance of death if medical treatment is not provided. These effects would not occur if the same dose were accumulated gradually over many weeks or months (Refs. 1 and 2). Thus, one of the justifications for establishing annual dose limits is to ensure that occupational dose is spread out in time.

It is important to distinguish between whole body and partial body exposure. A localized dose to a small volume of the body would not produce the same effect as a whole body dose of the same magnitude. For example, if only the hand were exposed, the effect would mainly be limited to the skin and underlying tissue of the hand. An acute dose of 400 to 600 rads (4–6 Gy) to the hand would cause skin reddening; recovery would occur over the following months and no long-term damage would be expected. An acute dose of this magnitude to the whole body could cause death within a short time without medical treatment. Medical treatment would lessen the magnitude of the effects and the chance of death; however, it would not totally eliminate the effects or the chance of death.

DELAYED EFFECTS

Delayed effects may occur years after exposure. These effects are caused indirectly when the radiation changes parts of the cells in the body, which causes the normal function of the cell to change, for example, normal healthy cells turn into cancer cells. The potential for these delayed health effects is one of the main concerns addressed when setting limits on occupational doses.

A delayed effect of special interest is genetic effects. Genetic effects may occur if there is radiation damage to the cells of the gonads (sperm or eggs). These effects may show up as genetic defects in the children of the exposed individual and succeeding generations. However, if any genetic effects (i.e., effects in addition to the normal expected number) have been caused by radiation, the numbers are too small to have been observed in human populations exposed to radiation. For example, the atomic bomb survivors (from Hiroshima and Nagasaki) have not shown any significant radiation-related increases in genetic defects (Ref. 3). Effects have been observed in animal studies conducted at very high levels of exposure and it is known that radiation can cause changes in the genes in cells of the human body. However, it is believed that by maintaining worker exposures below the NRC limits and consistent with ALARA, a margin of safety is provided such that the risk of genetic effects is almost eliminated.

4. What is the difference between acute and chronic radiation dose?

Acute radiation dose usually refers to a large dose of radiation received in a short period of time. Chronic dose refers to the sum of small doses received repeatedly over long time periods, for example, 20 mrem (or millirem, which is 1-thousandth of a rem) (0.2 mSv) per week every week for several years. It is assumed for radiation protection purposes that any radiation dose, either acute or chronic, may cause delayed effects. However, only large acute doses cause early effects; chronic doses within the occupational dose limits do not cause early effects. Since the NRC limits do not permit large acute doses, concern with occupational radiation risk is primarily focused on controlling chronic exposure for which possible delayed effects, such as cancer, are of concern.

The difference between acute and chronic radiation exposure can be shown by using exposure to the sun's rays as an example. An intense exposure to the sun can result in painful burning, peeling, and growing of new skin. However, repeated short exposures provide time for the skin to be repaired between exposures. Whether exposure to the sun's rays is long term or spread over short periods, some of the injury may not be repaired and may eventually result in skin cancer.

Cataracts are an interesting case because they can be caused by both acute and chronic radiation. A certain threshold level of dose to the lens of the eye is required before there is any observable visual impairment, and the impairment remains after the exposure is stopped. The threshold for cataract development

from acute exposure is an acute dose on the order of 100 rads (1 Gy). Further, a cumulative dose of 800 rads (8 Gy) from protracted exposures over many years to the lens of the eye has been linked to some level of visual impairment (Refs. 1 and 4). These doses exceed the amount that may be accumulated by the lens from normal occupational exposure under the current regulations.

5. What is meant by external and internal exposure?

A worker's occupational dose may be caused by exposure to radiation that originates outside the body, called "external exposure," or by exposure to radiation from radioactive material that has been taken into the body, called "internal exposure." Most NRC-licensed activities involve little, if any, internal exposure. It is the current scientific consensus that a rem of radiation dose has the same biological risk regardless of whether it is from an external or an internal source. The NRC requires that dose from external exposure and dose from internal exposure be added together, if each exceeds 10% of the annual limit, and that the total be within occupational limits. The sum of external and internal dose is called the total effective dose equivalent (TEDE) and is expressed in units of rems (Sv).

Although unlikely, radioactive materials may enter the body through breathing, eating, drinking, or open wounds, or they may be absorbed through the skin. The intake of radioactive materials by workers is generally due to breathing contaminated air. Radioactive materials may be present as fine dust or gases in the workplace atmosphere. The surfaces of equipment and workbenches may be contaminated, and these materials can be resuspended in air during work activities.

If any radioactive material enters the body, the material goes to various organs or is excreted, depending on the biochemistry of the material. Most radioisotopes are excreted from the body in a few days. For example, a fraction of any uranium taken into the body will deposit in the bones, where it remains for a longer time. Uranium is slowly eliminated from the body, mostly by way of the kidneys. Most workers are not exposed to uranium. Radioactive iodine is preferentially deposited in the thyroid gland, which is located in the neck.

To limit risk to specific organs and the total body, an annual limit on intake (ALI) has been established for each radionuclide. When more than one radionuclide is involved, the intake amount of each radionuclide is reduced proportionally. NRC regulations specify the concentrations of radioactive material in the air to which a worker may be exposed for 2,000 working hours in a year. These concentrations are termed the derived air concentrations (DACs). These limits are the total amounts allowed if no external radiation is received. The resulting dose from the internal radiation sources (from breathing air at 1 DAC) is the maximum allowed to an organ or to the worker's whole body.

6. How does radiation cause cancer?

The mechanisms of radiation-induced cancer are not completely understood. When radiation interacts with the cells of our bodies, a number of events can occur. The damaged cells can repair themselves and permanent damage is not caused. The cells can die, much like the large numbers of cells that die every day in our bodies, and be replaced through the normal biological processes. Or a change can occur in the cell's reproductive structure, the cells can mutate and subsequently be repaired without effect, or they can form precancerous cells, which may become cancerous. Radiation is only one of many agents with the potential for causing cancer, and cancer caused by radiation cannot be distinguished from cancer attributable to any other cause.

Radiobiologists have studied the relationship between large doses of radiation and cancer (Refs. 5 and 6). These studies indicate that damage or change to genes in the cell nucleus is the main cause of radiation-induced cancer. This damage may occur directly through the interaction of the ionizing radiation in the cell or indirectly through the actions of chemical products produced by radiation interactions within cells. Cells are able to repair most damage within hours; however, some cells may not be repaired properly. Such misrepaired damage is thought to be the origin of cancer, but misrepair does not always cause cancer. Some cell changes are benign or the cell may die; these changes do not lead to cancer.

Many factors such as age, general health, inherited traits, sex, as well as exposure to other cancer-causing agents such as cigarette smoke can affect susceptibility to the cancer-causing effects of radiation. Many diseases are caused by the interaction of several factors, and these interactions appear to increase the susceptibility to cancer.

7. Who developed radiation risk estimates?

Radiation risk estimates were developed by several national and international scientific organizations over the last 40 years. These organizations include the National Academy of Sciences (which has issued several reports from the Committee on the Biological Effects of Ionizing Radiations, BEIR), the National Council on Radiation Protection and Measurements (NCRP), the International Commission on Radiological Protection (ICRP), and the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). Each of these organizations continues to review new research findings on radiation health risks.

Several reports from these organizations present new findings on radiation risks based upon revised estimates of radiation dose to survivors of the atomic bombing at Hiroshima and Nagasaki. For example, UNSCEAR published risk estimates in 1988 and 1993 (Refs. 5 and 6). The NCRP also published a report in 1988, "New Dosimetry at Hiroshima and Nagasaki and Its Implications for Risk Estimates" (Ref. 7). In January 1990, the National Academy of Sciences released the fifth report of the BEIR Committee, "Health Effects of Exposure to Low Levels of Ionizing Radiation" (Ref. 4). Each of these publications also provides extensive bibliographies on other published studies concerning radiation health effects for those who may wish to read further on this subject.

8. What are the estimates of the risk of fatal cancer from radiation exposure?

We don't know exactly what the chances are of getting cancer from a low-level radiation dose, primarily because the few effects that may occur cannot be distinguished from normally occurring cancers. However, we can make estimates based on extrapolation from extensive knowledge from scientific research on high dose effects. The estimates of radiation effects at high doses are better known than are those of most chemical carcinogens (Ref. 8).

From currently available data, the NRC has adopted a risk value for an occupational dose of 1 rem (0.01 Sv) Total Effective Dose Equivalent (TEDE) of 4 in 10,000 of developing a fatal cancer, or approximately 1 chance in 2,500 of fatal cancer per rem of TEDE received. The uncertainty associated with this risk estimate does not rule out the possibility of higher risk, or the possibility that the risk may even be zero at low occupational doses and dose rates.

The radiation risk incurred by a worker depends on the amount of dose received. Under the linear model explained above, a worker who receives 5 rems (0.05 Sv) in a year incurs 10 times as much risk as another worker who receives only 0.5 rem (0.005 Sv). Only a very few workers receive doses near 5 rems (0.05 Sv) per year (Ref. 9).

According to the BEIR V report (Ref. 4), approximately one in five adults normally will die from cancer from all possible causes such as smoking, food, alcohol, drugs, air pollutants, natural background radiation, and inherited traits. Thus, in any group of 10,000 workers, we can estimate that about 2,000 (20%) will die from cancer without any occupational radiation exposure.

To explain the significance of these estimates, we will use as an example a group of 10,000 people, each exposed to 1 rem (0.01 Sv) of ionizing radiation. Using the risk factor of 4 effects per 10,000 rem of dose, we estimate that 4 of the 10,000 people might die from delayed cancer because of that 1-rem dose (although the actual number could be more or less than 4) in addition to the 2,000 normal cancer fatalities expected to occur in that group from all other causes. This means that a 1-rem (0.01 Sv) dose may increase an individual worker's chances of dying from cancer from 20 percent to 20.04 percent. If one's lifetime occupational dose is 10 rems, we could raise the estimate to 20.4 percent. A lifetime dose of 100 rems may increase chances of dying from cancer from 20 to 24 percent. The average measurable dose for radiation workers reported to the NRC was 0.31 rem (0.0031 Sv) for 1993 (Ref. 9). Today, very few workers ever accumulate 100 rems (1 Sv) in a working lifetime, and the average career dose of workers at NRC-licensed facilities is 1.5 rems (0.015 Sv), which represents an estimated increase from 20 to about 20.06 percent in the risk of dying from cancer.

It is important to understand the probability factors here. A similar question would be, "If you select one card from a full deck of cards, will you get the ace of spades?" This question cannot be answered with a simple yes or no. The best answer is that your chance is 1 in 52. However, if 1000 people each select one card from full decks, we can predict that about 20 of them will get an ace of spades. Each person will have 1 chance in 52 of drawing the ace of spades, but there is no way we can predict which persons will get that card. The issue is further complicated by the fact that in a drawing by 1000 people, we might get only 15 successes, and in another, perhaps 25 correct cards in 1000 draws. We can say that if you receive a radiation dose, you will have increased your chances of eventually developing cancer. It is assumed that the more radiation exposure you get, the more you increase your chances of cancer.

The normal chance of dying from cancer is about one in five for persons who have not received any occupational radiation dose. The additional chance of developing fatal cancer from an occupational exposure of 1 rem (0.01 Sv) is about the same as the chance of drawing any ace from a full deck of cards three times in a row. The additional chance of dying from cancer from an occupational exposure of 10 rem (0.1 Sv) is about equal to your chance of drawing two aces successively on the first two draws from a full deck of cards.

It is important to realize that these risk numbers are only estimates based on data for people and research animals exposed to high levels of radiation in short periods of time. There is still uncertainty with regard to estimates of radiation risk from low levels of exposure. Many difficulties are involved in designing research studies that can accurately measure the projected small increases in cancer cases that might be caused by low exposures to radiation as compared to the normal rate of cancer.

These estimates are considered by the NRC staff to be the best available for the worker to use to make an informed decision concerning acceptance of the risks associated with exposure to radiation. A worker who decides to accept this risk should try to keep exposure to radiation as low as is reasonably achievable (ALARA) to avoid unnecessary risk.

9. If I receive a radiation dose that is within occupational limits, will it cause me to get cancer?

Probably not. Based on the risk estimates previously discussed, the risk of cancer from doses below the occupational limits is believed to be small. Assessment of the cancer risks that may be associated with low doses of radiation are projected from data available at doses larger than 10 rems (0.1 Sv) (Ref. 3). For radiation protection purposes, these estimates are made using the straight line portion of the linear quadratic model (Curve 2 in Figure 1). We have data on cancer probabilities only for high doses, as shown by the solid line in Figure 1. Only in studies involving radiation doses above occupational limits are there dependable determinations of the risk of cancer, primarily because below the limits the effect is small compared to differences in the normal cancer incidence from year to year and place to place. The ICRP, NCRP, and other standards-setting organizations assume for radiation protection purposes that there is some risk, no matter how small the dose (Curves 1 and 2). Some scientists believe that the risk drops off to zero at some low dose (Curve 3), the threshold effect. The ICRP and NCRP endorse the linear quadratic model as a conservative means of assuring safety (Curve 2).

For regulatory purposes, the NRC uses the straight line portion of Curve 2, which shows the number of effects decreasing linearly as the dose decreases. Because the scientific evidence does not conclusively demonstrate whether there is or is not an effect at low doses, the NRC assumes for radiation protection purposes, that even small doses have some chance of causing cancer. Thus, a principle of radiation protection is to do more than merely meet the allowed regulatory limits; doses should be kept as low as is reasonably achievable (ALARA). This is as true for natural carcinogens such as sunlight and natural radiation as it is for those that are manmade, such as cigarette smoke, smog, and x-rays.

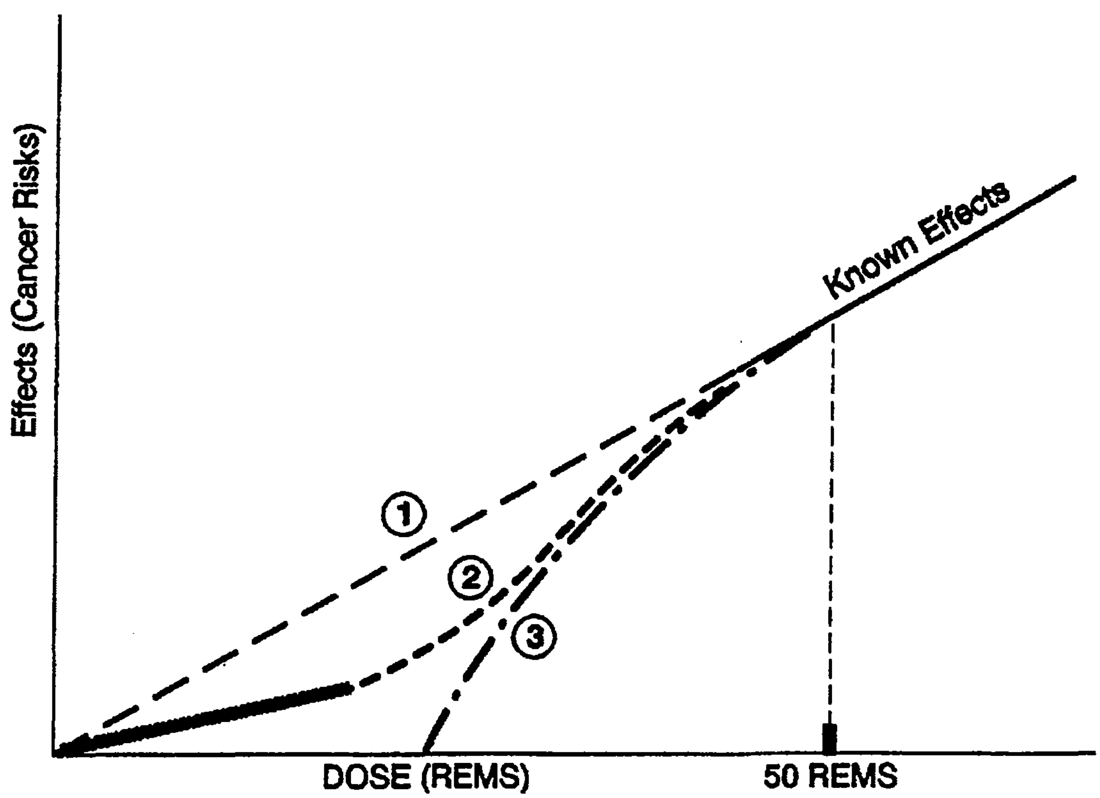

Figure 1. Some Proposed Models for How the Effects of Radiation Vary With Doses at Low Levels

10. How can we compare the risk of cancer from radiation to other kinds of health risks?

One way to make these comparisons is to compare the average number of days of life expectancy lost because of the effects associated with each particular health risk. Estimates are calculated by looking at a large number of persons, recording the age when death occurs from specific causes, and estimating the average number of days of life lost as a result of these early deaths. The total number of days of life lost is then averaged over the total observed group.

Several studies have compared the average days of life lost from exposure to radiation with the number of days lost as a result of being exposed to other health risks. The word "average" is important because an individual who gets cancer loses about 15 years of life expectancy, while his or her coworkers do not suffer any loss.

Some representative numbers are presented in Table 1. For categories of NRC-regulated industries with larger doses, the average measurable occupational dose in 1993 was 0.31 rem (0.0031 Sv). A simple calculation based on the article by Cohen and Lee (Ref. 10) shows that 0.3 rem (0.003 Sv) per year from age 18 to 65 results in an average loss of 15 days. These estimates indicate that the health risks from occupational radiation exposure are smaller than the risks associated with many other events or activities we encounter and accept in normal day-to-day activities.

It is also useful to compare the estimated average number of days of life lost from occupational exposure to radiation with the number of days lost as a result of working in several types of industries. Table 2 shows average days of life expectancy lost as a result of fatal work-related accidents. Table 2 does not include non-accident types of occupational risks such as occupational disease and stress because the data are not available.

These comparisons are not ideal because we are comparing the possible effects of chronic exposure to radiation to different kinds of risk such as accidental death, in which death is inevitable if the event occurs. This is the best we can do because good data are not available on chronic exposure to other workplace carcinogens. Also, the estimates of loss of life expectancy for workers from radiation-induced cancer do not take into consideration the competing effect on the life expectancy of the workers from industrial accidents.

11. What are the health risks from radiation exposure to the embryo/fetus?

During certain stages of development, the embryo/fetus is believed to be more sensitive to radiation damage than adults. Studies of atomic bomb survivors exposed to acute radiation doses exceeding 20 rads (0.2 Gy) during pregnancy show that children born after receiving these doses have a higher risk of mental retardation. Other studies suggest that an association exists between exposure to diagnostic x-rays before birth and carcinogenic effects in childhood and in adult life. Scientists are uncertain about the magnitude of the risk. Some studies show the embryo/fetus to be more sensitive to radiation-induced cancer than adults, but other studies do not. In recognition of the possibility of increased radiation sensitivity, and because dose to the

Table 1 Estimated Loss of Life Expectancy from Health Risks[a]

Health Risk	*Estimate of Life Expectancy Lost (average)*	
Smoking 20 cigarettes a day		6 years
Overweight (by 15%)		2 years
Alcohol consumption (U.S. average)		1 year
All accidents combined		1 year
Motor vehicle accidents	207 days	
Home accidents	74 days	
Drowning	24 days	
All natural hazards (earthquake, lightning, flood, etc.)		7 days
Medical radiation		6 days
Occupational Exposure		
0.3 rem/y from age 18 to 65	15 days	
1 rem/y from age 18 to 65	51 days	

[a]Adapted from Reference 10.

Table 2 Estimated Loss of Life Expectancy from Industrial Accidents[a]	
Industry Type	*Estimated Days of Life Expectancy Lost (Average)*
All industries	60
Agriculture	320
Construction	227
Mining and Quarrying	167
Transportation and Public Utilities	160
Government	60
Manufacturing	40
Trade	27
Services	27

[a]Adapted from Reference 10.

embryo/fetus is involuntary on the part of the embryo/fetus, a more restrictive dose limit has been established for the embryo/fetus of a declared pregnant radiation worker. See Regulatory Guide 8.13, "Instruction Concerning Prenatal Radiation Exposure."

If an occupationally exposed woman declares her pregnancy in writing, she is subject to the more restrictive dose limits for the embryo/fetus during the remainder of the pregnancy. The dose limit of 500 mrems (5 mSv) for the total gestation period applies to the embryo/fetus and is controlled by restricting the exposure to the declared pregnant woman. Restricting the woman's occupational exposure, if she declares her pregnancy, raises questions about individual privacy rights, equal employment opportunities, and the possible loss of income. Because of these concerns, the declaration of pregnancy by a female radiation worker is voluntary. Also, the declaration of pregnancy can be withdrawn for any reason, for example, if the woman believes that her benefits from receiving the occupational exposure would outweigh the risk to her embryo/fetus from the radiation exposure.

12. Can a worker become sterile or impotent from normal occupational radiation exposure?

No. Temporary or permanent sterility cannot be caused by radiation at the levels allowed under NRC's occupational limits. There is a threshold below which these effects do not occur. Acute doses on the order of 10 rems (0.1 Sv) to the testes can result in a measurable but temporary reduction in sperm count. Temporary sterility (suppression of ovulation) has been observed in women who have received acute doses of 150 rads (1.5 Gy). The estimated threshold (acute) radiation dose for induction of permanent sterility is about 200 rads (2 Gy) for men and about 350 rads (3.5 Gy) for women (Refs. 1 and 4). These doses are far greater than the NRC s occupational dose limits for workers.

Although acute doses can affect fertility by reducing sperm count or suppressing ovulation, they do not have any direct effect on one's ability to function sexually. No evidence exists to suggest that exposures within the NRC's occupational limits have any effect on the ability to function sexually.

13. What are the NRC occupational dose limits?

For adults, an annual limit that does not exceed:

- 5 rems (0.05 Sv) for the total effective dose equivalent (TEDE), which is the sum of the deep dose equivalent (DDE) from external exposure to the whole body and the committed effective dose equivalent (CEDE) from intakes of radioactive material.
- 50 rems (0.5 Sv) for the total organ dose equivalent (TODE), which is the sum of the DDE from external exposure to the whole body and the committed dose equivalent (CDE) from intakes of radioactive material to any individual organ or tissue, other than the lens of the eye.
- 15 rems (0.15 Sv) for the lens dose equivalent (LDE), which is the external dose to the lens of the eye.
- 50 rems (0.5 Sv) for the shallow dose equivalent (SDE), which is the external dose to the skin or to any extremity.

For minor workers, the annual occupational dose limits are 10 percent of the dose limits for adult workers.

For protection of the embryo/fetus of a declared pregnant woman, the dose limit is 0.5 rem (5 mSv) during the entire pregnancy.

The occupational dose limit for adult workers of 5 rems (0.05 Sv) TEDE is based on consideration of the potential for delayed biological effects. The 5-rem (0.05 Sv) limit, together with application of the concept of keeping occupational doses ALARA, provides a level of risk of delayed effects considered acceptable by the NRC. The limits for individual organs are below the dose levels at which early biological effects are observed in the individual organs.

The dose limit for the embryo/fetus of a declared pregnant woman is based on a consideration of the possibility of greater sensitivity to radiation of the embryo/fetus and the involuntary nature of the exposure.

14. What is meant by ALARA?

ALARA means "as low as is reasonably achievable." In addition to providing an upper limit on an individual's permissible radiation dose, the NRC requires that its licensees establish radiation protection

programs and use procedures and engineering controls to achieve occupational doses, and doses to the public, as far below the limits as is reasonably achievable. "Reasonably achievable" also means "to the extent practicable." What is practicable depends on the purpose of the job, the state of technology, the costs for averting doses, and the benefits. Although implementation of the ALARA principle is a required integral part of each licensee's radiation protection program, it does not mean that each radiation exposure must be kept to an absolute minimum, but rather that "reasonable" efforts must be made to avert dose. In practice, ALARA includes planning tasks involving radiation exposure so as to reduce dose to individual workers and the work group.

There are several ways to control radiation doses, e.g., limiting the time in radiation areas, maintaining distance from sources of radiation, and providing shielding of radiation sources to reduce dose. The use of engineering controls, from the design of facilities and equipment to the actual set-up and conduct of work activities, is also an important element of the ALARA concept.

An ALARA analysis should be used in determining whether the use of respiratory protection is advisable. In evaluating whether or not to use respirators, the goal should be to achieve the optimal sum of external and internal doses. For example, the use of respirators can lead to increased work time within radiation areas, which increases external dose. The advantage of using respirators to reduce internal exposure must be evaluated against the increased external exposure and related stresses caused by the use of respirators. Heat stress, reduced visibility, and reduced communication associated with the use of respirators could expose a worker to far greater risks than are associated with the internal dose avoided by use of the respirator. To the extent practical, engineering controls, such as containments and ventilation systems, should be used to reduce workplace airborne radioactive materials.

15. What are background radiation exposures?

The average person is constantly exposed to ionizing radiation from several sources. Our environment and even the human body contain naturally occurring radioactive materials (e.g., potassium-40) that contribute to the radiation dose that we receive. The largest source of natural background radiation exposure is terrestrial radon, a colorless, odorless, chemically inert gas, which causes about 55 percent of our average, nonoccupational exposure. Cosmic radiation originating in space contributes additional exposure. The use of x-rays and radioactive materials in medicine and dentistry adds to our population exposure. As shown below in Table 3, the average person receives an annual radiation dose of about 0.36 rem (3.6 mSv). By age 20, the average person will accumulate over 7 rems (70 mSv) of dose. By age 50, the total dose is up to 18 rems (180 mSv). After 70 years of exposure this dose is up to 25 rems (250 mSv).

Table 3 Average Annual Effective Dose Equivalent to Individuals in the U.S.[a]

Source	*Effective Dose Equivalent (mrems)*	
Natural		
Radon	200	
Other than Radon	100	
Total		300
Nuclear Fuel Cycle		0.05
Consumer Products[b]		9
Medical		
Diagnostic X-rays	39	
Nuclear Medicine	14	
Total		53
Total		about 360 mrems/year

[a]Adapted from Table 8.1, NCRP 93 (Ref. 11).

[b]Includes building material, television receivers, luminous watches, smoke detectors, etc. (from Table 5.1, NCRP 93, Ref. 11).

16. What are the typical radiation doses received by workers?

For 1993, the NRC received reports on about a quarter of a million people who were monitored for occupational exposure to radiation. Almost half of those monitored had no measurable doses. The other half had an average dose of about 310 mrem (3.1 mSv) for the year. Of these, 93 percent received an annual dose of less than 1 rem (10 mSv); 98.7 percent received less than 2 rems (20 mSv); and the highest reported dose was for two individuals who each received between 5 and 6 rems (50 and 60 mSv).

Table 4 lists average occupational doses for workers (persons who had measurable doses) in various occupations based on 1993 data. It is important to note that beginning in 1994, licensees have been required to sum external and internal doses and certain licensees are required to submit annual reports. Certain types of licensees such as nuclear fuel fabricators may report a significant increase in worker doses because of the exposure to long-lived airborne radionuclides and the requirement to add the resultant internal dose to the calculation of occupational doses.

Table 4 Reported Occupational Doses for 1993[a]

Occupational Subgroup	Average Measurable Dose per Worker (millirems)
Industrial Radiography	540
Commercial Nuclear Power Reactors	310
Manufacturing and Distribution of Radioactive Materials	300
Low-Level Radioactive Waste Disposal	270
Independent Spent Nuclear Fuel Storage	260
Nuclear Fuel Fabrication	130

[a]From Table 3.1 in NUREG-0713 (Ref. 9).

17. How do I know how much my occupational dose (exposure) is?

If you are likely to receive more than 10 percent of the annual dose limits, the NRC requires your employer, the NRC licensee, to monitor your dose, to maintain records of your dose, and, at least on an annual basis for the types of licensees listed in 10 CFR 20.2206, "Reports of Individual Monitoring," to inform both you and the NRC of your dose. The purpose of this monitoring and reporting is so that the NRC can be sure that licensees are complying with the occupational dose limits and the ALARA principle.

External exposures are monitored by using individual monitoring devices. These devices are required to be used if it appears likely that external exposure will exceed 10 percent of the allowed annual dose, i.e., 0.5 rem (5 mSv). The most commonly used monitoring devices are film badges, thermoluminescence dosimeters (TLDs), electronic dosimeters, and direct reading pocket dosimeters.

With respect to internal exposure, your employer is required to monitor your occupational intake of radioactive material and assess the resulting dose if it appears likely that you will receive greater than 10 percent of the annual limit on intake (ALI) from intakes in 1 year. Internal exposure can be estimated by measuring the radiation emitted from the body (for example, with a "whole body counter") or by measuring the radioactive materials contained in biological samples such as urine or feces. Dose estimates can also be made if one knows how much radioactive material was in the air and the length of time during which the air was breathed.

18. What happens if a worker exceeds the annual dose limit?

If a worker receives a dose in excess of any of the annual dose limits, the regulations prohibit any occupational exposure during the remainder of the year in which the limit is exceeded. The licensee is also required to file an overexposure report with the NRC and provide a copy to the individual who received the dose. The licensee may be subject to NRC enforcement action such as a fine (civil penalty), just as individuals are subject to a traffic fine for exceeding a speed limit. The fines and, in some serious or repetitive cases, suspension of a license are intended to encourage licensees to comply with the regulations.

Radiation protection limits do not define safe or unsafe levels of radiation exposure. Exceeding a limit does not mean that you will get cancer. For radiation protection purposes, it is assumed that risks are related to the size of the radiation dose. Therefore, when your dose is higher your risk is also considered to be higher. These limits are similar to highway speed limits. If you drive at 70 mph, your risk is higher than at 55 mph, even though you may not actually have an accident. Those who set speed limits have determined that the risks of driving in excess of the speed limit are not acceptable. In the same way, the revised 10 CFR Part 20 establishes a limit for normal occupational exposure of 5 rems (0.05 Sv) a year. Although you will not necessarily get cancer or some other radiation effect at doses above the limit, it does mean that the licensee's safety program has failed in some way. Investigation is warranted to determine the cause and correct the conditions leading to the dose in excess of the limit.

19. What is meant by a "planned special exposure"?

A "planned special exposure" (PSE) is an infrequent exposure to radiation, separate from and in addition to the radiation received under the annual occupational limits. The licensee can authorize additional dose in any one year that is equal to the annual occupational dose limit as long as the individual's total dose from PSEs does not exceed five times the annual dose limit during the individual's lifetime. For example, licensees may authorize PSEs for an adult radiation worker to receive doses up to an additional 5 rems (0.05 Sv) in a year above the 5-rem (0.05-Sv) annual TEDE occupational dose limit. Each worker is limited to no more than 25 rems (0.25 Sv) from planned special exposures in his or her lifetime. Such exposures are only allowed in exceptional situations when alternatives for avoiding the additional exposure are not available or are impractical.

Before the licensee authorizes a PSE, the licensee must ensure that the worker is informed of the purpose and circumstances of the planned operation, the estimated doses expected, and the procedures to keep the doses ALARA while considering other risks that may

be present. (See Regulatory Guide 8.35, "Planned Special Exposures.")

20. Why do some facilities establish administrative control levels that are below the NRC limits?

There are two reasons. First, the NRC regulations state that licensees must take steps to keep exposures to radiation ALARA. Specific approval from the licensee for workers to receive doses in excess of administrative limits usually results in more critical risk-benefit analyses as each additional increment of dose is approved for a worker. Secondly, an administrative control level that is set lower than the NRC limit provides a safety margin designed to help the licensee avoid doses to workers in excess of the limit.

21. Why aren't medical exposures considered as part of a worker's allowed dose?

NRC rules exempt medical exposure, but equal doses of medical and occupational radiation have equal risks. Medical exposure to radiation is justified for reasons that are quite different from the reasons for occupational exposure. A physician prescribing an x-ray, for example, makes a medical judgment that the benefit to the patient from the resulting medical information justifies the risk associated with the radiation. This judgment may or may not be accepted by the patient. Similarly, each worker must decide on the benefits and acceptability of occupational radiation risk, just as each worker must decide on the acceptability of any other occupational hazard.

Consider a worker who receives a dose of 3 rems (0.03 Sv) from a series of x-rays in connection with an injury or illness. This dose and any associated risk must be justified on medical grounds. If the worker had also received 2 rems (0.02 Sv) on the job, the combined dose of 5 rems (0.05 Sv) would in no way incapacitate the worker. Restricting the worker from additional job exposure during the remainder of the year would not have any effect on the risk from the 3 rems (0.03 Sv) already received from the medical exposure. If the individual worker accepts the risks associated with the x-rays on the basis of the medical benefits and accepts the risks associated with job-related exposure on the basis of employment benefits, it would be unreasonable to restrict the worker from employment involving exposure to radiation for the remainder of the year.

22. How should radiation risks be considered in an emergency?

Emergencies are "unplanned" events in which actions to save lives or property may warrant additional doses for which no particular limit applies. The revised 10 CFR Part 20 does not set any dose limits for emergency or lifesaving activities and states that nothing in Part 20 "shall be construed as limiting actions that may be necessary to protect health and safety."

Rare situations may occur in which a dose in excess of occupational limits would be unavoidable in order to carry out a lifesaving operation or to avoid a large dose to large populations. However, persons called upon to undertake any emergency operation should do so only on a voluntary basis and with full awareness of the risks involved.

For perspective, the Environmental Protection Agency (EPA) has published emergency dose guidelines (Ref. 2). These guidelines state that doses to all workers during emergencies should, to the extent practicable, be limited to 5 rems (0.05 Sv). The EPA further states that there are some emergency situations for which higher limits may be justified. The dose resulting from such emergency exposures should be limited to 10 rems (0.1 Sv) for protecting valuable property, and to 25 rems (0.25 Sv) for lifesaving activities and the protection of large populations. In the context of this guidance, the dose to workers that is incurred for the protection of large populations might be considered justified for situations in which the collective dose to others that is avoided as a result of the emergency operation is significantly larger than that incurred by the workers involved.

Table 5 presents the estimates of the fatal cancer risk for a group of 1,000 workers of various ages, assuming that each worker received an acute dose of 25 rems (0.25 Sv) in the course of assisting in an emergency. The estimates show that a 25-rem emergency dose might increase an individual's chances of developing fatal cancer from about 20% to about 21%.

Table 5
Risk of Premature Death from Exposure to 25-Rems (0.25-Sv) Acute Dose

Age at Exposure (years)	*Estimated Risk of Premature Death (Deaths per 1,000 Persons Exposed)*
20–30	9.1
30–40	7.2
40–50	5.3
50–60	3.5

Source: EPA-400-R-92-001 (Ref. 2).

23. How were radiation dose limits established?

The NRC radiation dose limits in 10 CFR Part 20 were established by the NRC based on the recommendations of the ICRP and NCRP as endorsed in Federal radiation protection guidance developed by the EPA

(Ref. 12). The limits were recommended by the ICRP and NCRP with the objective of ensuring that working in a radiation-related industry was as safe as working in other comparable industries. The dose limits and the principle of ALARA should ensure that risks to workers are maintained indistinguishable from risks from background radiation.

24. Several scientific reports have recommended that the NRC establish lower dose limits. Does the NRC plan to reduce the regulatory limits?

Since publication of the NRC's proposed rule in 1986, the ICRP in 1990 revised its recommendations for radiation protection based on newer studies of radiation risks (Ref. 13), and the NCRP followed with a revision to its recommendations in 1993. The ICRP recommended a limit of 10 rems (0.1 Sv) effective dose equivalent (from internal and external sources), over a 5-year period with no more than 5 rems (0.05 Sv) in 1 year (Ref. 13). The NCRP recommended a cumulative limit in rems, not to exceed the individual's age in years, with no more than 5 rems (0.05 Sv) in any year (Ref. 14).

The NRC does not believe that additional reductions in the dose limits are required at this time. Because of the practice of maintaining radiation exposures ALARA (as low as is reasonably achievable), the average radiation dose to occupationally exposed persons is well below the limits in the current Part 20 that became mandatory January 1, 1994, and the average doses to radiation workers are below the new limits recommended by the ICRP and the NCRP.

25. What are the options if a worker decides that the risks associated with occupational radiation exposure are too high?

If the risks from exposure to occupational radiation are unacceptable to a worker, he or she can request a transfer to a job that does not involve exposure to radiation. However, the risks associated with the exposure to radiation that workers, on the average, actually receive are comparable to risks in other industries and are considered acceptable by the scientific groups that have studied them. An employer is not obligated to guarantee a transfer if a worker decides not to accept an assignment that requires exposure to radiation.

Any worker has the option of seeking other employment in a nonradiation occupation. However, the studies that have compared occupational risks in the nuclear industry to those in other job areas indicate that nuclear work is relatively safe. Thus, a worker may find different kinds of risk but will not necessarily find significantly lower risks in another job.

26. Where can one get additional information on radiation risk?

The following list suggests sources of useful information on radiation risk:

- The employer—the radiation protection or health physics office where a worker is employed.
- Nuclear Regulatory Commission Regional Offices:

King of Prussia, Pennsylvania	(610) 337-5000
Atlanta, Georgia	(404) 331-4503
Lisle, Illinois	(708) 829-9500
Arlington, Texas	(817) 860-8100

- U.S. Nuclear Regulatory Commission Headquarters
  Radiation Protection & Health Effects Branch
  Office of Nuclear Regulatory Research
  Washington, DC 20555
  Telephone: (301) 415-6187
- Department of Health and Human Services
  Center for Devices and Radiological Health
  1390 Piccard Drive, MS HFZ-1
  Rockville, MD 20850
  Telephone: (301) 443-4690
- U.S. Environmental Protection Agency
  Office of Radiation and Indoor Air
  Criteria and Standards Division
  401 M Street NW.
  Washington, DC 20460
  Telephone: (202) 233-9290

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*Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555; telephone (202) 634-3273; fax (202) 634-3343. Copies may be purchased at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202) 512-2249); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161.

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U.S. Nuclear Regulatory Commission, "Radiation Dose to the Embryo/Fetus," Regulatory Guide 8.36, July 1992.[2]

[1]Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555-0001; telephone (202) 634-3273; fax (202) 634-3343. Copies may be purchased at current rates from the U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20402-9328 (telephone (202) 512-2249); or from the National Technical Information Service by writing NTIS at 5285 Port Royal Road, Springfield, VA 22161.

[2]Single copies of regulatory guides may be obtained free of charge by writing the Office of Administration, Attn: Distribution and Services Section, USNRC, Washington, DC 20555, or by fax at (301) 415-2260. Copies are available for inspection or copying for a fee from the NRC Public Document Room at 2120 L Street NW., Washington, DC; the PDR's mailing address is Mail Stop LL-6, Washington, DC 20555-0001; telephone (202) 634-3273; fax (202) 634-3343.

REGULATORY ANALYSIS

A separate regulatory analysis was not prepared for this Revision 1 to Regulatory Guide 8.29. A value/impact statement, which evaluated essentially the same subjects as are discussed in a regulatory analysis, accompanied Regulatory Guide 8.29 when it was issued in July 1981.

This Revision 1 to Regulatory Guide 8.29 is needed to conform with the Revised 10 CFR Part 20, "Standards for Protection Against Radiation," as published May 21, 1991 (56 FR 23360). The regulatory analysis prepared for 10 CFR Part 20 provides the regulatory basis for this Revision 1 of Regulatory Guide 8.29, and it examines the costs and benefits of the rule as implemented by the guide. A copy of the "Regulatory Analysis for the Revision of 10 CFR Part 20" (PNL-6712, November 1988), is available for inspection and copying for a fee in the NRC's Public Document Room at 2120 L Street NW., Washington, DC 20555-0001.

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Radiation Monitors EAL Matrix

Emergency Action Level Matrix:

The EAL Matrix (PINGP 1576) provides the emergency action level thresholds for various plant radiation monitor readings. All plant radiation effluent and area radiation monitors are used in the EAL Matrix. Review PINGP 1576 EAL Matrix for actual EAL thresholds.

Viewing Rad Monitor Alarm Setpoints:

NOTE:	**ERCS and Radiation Alarm setpoints may NOT be the same. Use the Calibration Card / Set Point Program for actual values.**

VIA Calibration Card/Set Point File:

Alarm set points are maintained in the "Calibration Card/Set Point File". All radiation monitor High Alarm setpoint may be printed by performing the following:

1. Go to "Start Menu", "Applications", "Calibration Card & Set Point File".
2. Enter your own Username and password.
3. Select "Module", "Set Point".
4. For viewing the entire radiation monitor listing, choose System "RD" and enter Search button.
5. After about 1 minute of searching, the results should be shown on the screen.
6. To print the results, choose "File", "Print". The setpoint file will be printed to your selected printer.

VIA ERCS:

ERCS may be used to display radiation monitor alarms:

1. Go to either ERCS Tabular RADMON Group or ERCS Display RADMON.
2. Click value of the rad monitor of interest. The selected rad monitor's Analog Point Attributes Definition display will be shown.
3. Click the "Alarm Limits" button in lower right corner of display. The rad monitor's alarm setpoints will be displayed.

ODCM setpoints and limits:

ODCM setpoints are managed by the Rad Protection Group. RPIP 4523, Monthly Effluent Monitor Setpoint Determination is a responsibility of the Radiation Protection Group. Forms generated by this RPIP are:

Monthly Effluent Monitor Setpoint Determination - Computer Generated Monthly Gaseous Effluent Monitor Setpoint Determination - PINGP 627 Monthly Liquid Effluent Monitor Setpoint Determination - PINGP 628

Contact Rad Protection supervision to learn of the most current specific ODCM effluent monitor setpoints.

Trending Rad Monitors on ERCS (F3-26.2):

1. Choose CUSTOM TRENDS on ERCS Main Menu.
2. Choose the desired Trend window. Up to 12 rad monitors may be plotted.
3. Select the first data point name field. A list of data points available will be presented.
4. Choose the list of desired rad monitors.
5. Choose start and end times.

Power Supplies for Effluent Rad Monitors & Ventilation Systems
Rev. 0

Developed By
Jim Payton, February, 2003

Effluent Radiation Monitors Sample Pump &
Loop Monitors (R-70/71) Power Supplies

Effluent Monitor	Panel	MCC	Bus	Bus
1R-11/12		1T2	122 or 222	16 or 26
1R-22		1T1	112 or 212	15 or 25
1R-30		1T2	122 or 222	16 or 26
1R-37		1T1	112 or 212	15 or 25
1R-50	1RPB8	1T2	122 or 222	16 or 26
1/2R-70	1RPB8	1T2	122 or 222	16 or 26
2R-11/12		1T1	212 or 112	25 or 15
2R-22		1T2	122 or 222	16 or 26
2R-30		1T2	122 or 222	16 or 26
2R-37		1T1	112 or 212	15 or 25
2R-50	1RPA8	1T1	112 or 212	15 or 25
1/2R-71	1RPA8	1T1	112 or 212	15 or 25
R-26		1K1	111	15
R-27		1K2	121	16
R-35		1RW1 Bus1	290	Non Safeguards

Ventilation System Exhaust Fan Power Supplies

Ventilation Sys	Exhaust Fan	MCC	Bus	Bus
SHIELD BUILDING VENTILATION				
	# 11	1M1	112	15
	# 12	1M2	122	16
	# 21	2M1	212	25
	# 22	2M2	222	26
AUXILIARY BUILDING SPECIAL VENTILATION				
	# 121	1MA1	112 or 212	15 or 25
	# 122	1MA2	122 or 222	16 or 26
SPENT FUEL POOL SPECIAL VENTILATION				
	# 121	1MA1	112 or 212	15 or 25
	# 122	1MA2	122 or 222	16 or 26

SHIELD BUILDING VENTILATION SYSTEM

Dampers isolate the filters during standby conditions. The ventilation system for each Shield Building also contains a vent stack, which penetrates the Shield Building dome and discharges to the atmosphere. Air can be directed to the stack from the Shield Building vent filters, the Auxiliary Building special vent filters, the Containment Purge filters or the Containment In-service Purge filters.

The Shield Building Ventilation System has two modes of operation, exhaust and recirculation. When the Shield Building recirculation fan starts from either SI or the control switch, the system is in the exhaust mode. The exhaust fan discharge damper and the recirc fan discharge damper are both open and the recirc damper is closed. All airflow is being exhausted through the stack.

When the annulus delta-P reaches -2 inches of H20, the recirc damper opens and the system is in the recirculation mode. All three dampers will stay open until the Shield Building recirc fan is stopped. During recirc mode, air is recirculated to the annulus. The system maintains negative pressure by removing in-leakage to the annulus through the exhaust path.

Design Limits

Recirculation Fans

Number	2 per unit
Type	Axial Vane
Capacity	5,000 cfm
Motor	20 hp

Exhaust Fan

Number	2 per unit
Type	Axial Vane
Capacity	200 cfm
Motor	5 hp

Filter Assembly

Number	2 per unit
Type	Particulate, absolute, charcoal with demister
Heater Capacity	16 kw
Filter Efficiency	99.9% for particulate > O. 3 microns 99.9% for elemental iodine 95% for organic iodine

AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM

The Auxiliary Building Special Ventilation System collects air leakage into the Auxiliary Building from the Containment Vessel following an accident. The system filters the leakage and directs it to the Shield Building vent stack for discharge to the atmosphere. The system is designed to create a partial vacuum in the area of highest possible contamination, producing airflow from areas of low contamination to areas of high contamination.

A single system serves both units. The system consists of two redundant trains with each train containing a fan, a filter and the associated ducting. Special ventilation exhaust fans #121 and #122 are powered from safeguards MCC 1M Bus 1 and Bus 2 respectively. Radiation detection equipment monitors the air flowing through the system.

I. ISSUE:

The best estimate for a I/NG ratio should be used when determining the Iodine source term during a Steam Generator Tube Rupture accident. The appropriate I/NG ratio is dependant upon the RCS I/NG ratio and the amount of iodine partitioning during the liquid to steam transition.

Three documents provide guidance for iodine scrubbing ratios for steam generator tube ruptures or leaks; NUREG-0800, Standard Review Plan; NUREG-0844, NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity; and NUREG/CR-2659 PNL 3794, February 1983, Iodine Transport Predicted for a Postulated Steam Line Break with Concurrent Rupture of Steam Generator Tubes.

Section 15.6.3, Radiological Consequences of Steam Generator Tube Failure (PWR), of NUREG-0800 page 15.6.3-4 provides guidance for iodine transport from the steam generator. The section states that an iodine-partitioning coefficient of 100 may be conservatively assumed.

NUREG-844 states the effective partitioning factor will depend on whether the Steam generator water level is above or below the rupture location(s). If the rupture is above the water level then no partitioning factor is assumed. It is conservative to assume if any tubes are uncovered, then the rupture is uncovered.

NUREG/CR-2659 states with the SG water level below the dryer separators, the dryers and separators would be effective in capturing most of the droplets carrying iodine. A value of 0.001 is taken in this situation. The assumption of no partitioning for the cases with the dryers and separators covered is assumed.

In an “Evaluation of Iodine/Noble Gas Ratios in SG” Mr. M. Agen to Mr. D. A. Schuelke it is concluded that an iodine to noble gas ratio of 0.1 should be used for the ratio from the RCS activity leaving the core.

From the above information it is concluded that a I/NG ratio of 0.001 should be used is the SG tubes are covered and the dryer and separator are not covered. If the SG tubes are uncovered or the dryer and separator are covered then it is conservative to use the iodine to noble gas ratio of 0.1 due to the ratio of activities leaving the core.

II. RESOLUTION:

1. Use F3-20 to determine the correct I/NG ratio during a Steam Generator Tube Rupture accident.

2. Use the attached Attachment E to 1ES-3.1 during adverse containment conditions (Containment Pressure > 5 psig or Containment Radiation >1E4 r/hr) to determine if the SG water level is in the Acceptable Region.

III. REFERENCES:

1. F3-20, Determination of Radioactive Release Concentrations
2. 1ES-3.1 (2ES-3.1)
3. NUREG-0800, Standard Review Plan
4. NUREG-0844, NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity
5. NUREG/CR-2659 PNL 3794, February 1983, Iodine Transport Predicted for a Postulated Steam Line Break with Concurrent Rupture of Steam Generator Tubes

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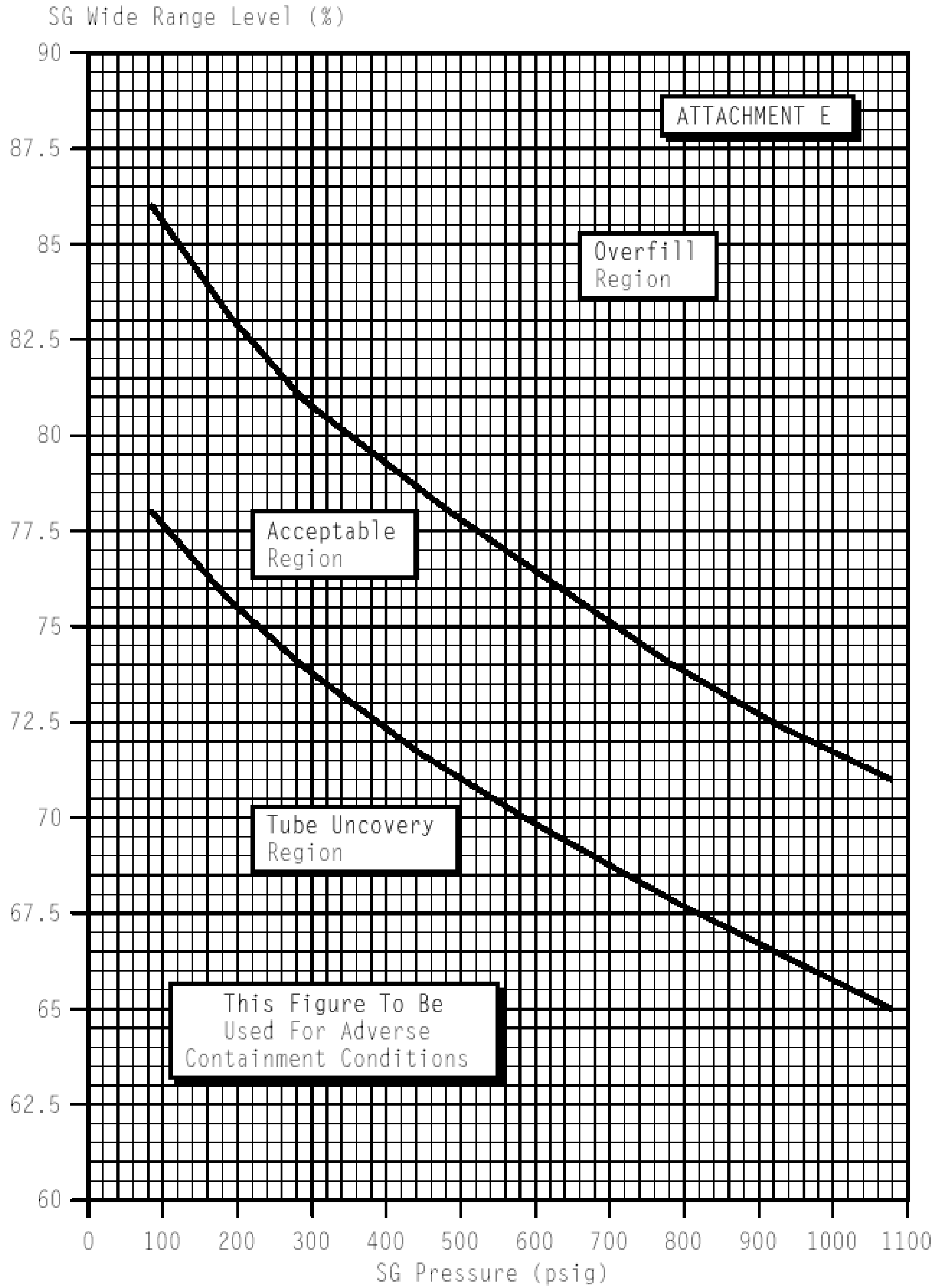

ATTACHMENT E
Wide Range SG Level
For Controlling Inventory

Taken from F3-17, "Core Damage Assessment"

5.0 DISCUSSION

The approach utilized in this methodology of core damage assessment is measurement of fission product concentrations in the primary coolant system, and containment, when applicable, utilizing the post accident sampling system.

Certain nuclides have been selected to be associated with each particular core damage state, i.e., clad damage, fuel overheat and fuel melt. These nuclides reach equilibrium quickly within the fuel cycle. Once equilibrium condition are reached, a fixed inventory of the nuclides is assumed to exist within the fuel pellet. For these nuclides which reach equilibrium, their relative ratios within the fuel pellet can also be considered to be constant. During operation, certain volatile fission products collect in the gap. The relative ratios in the gap can also be considered to be constant, however, the distribution of the nuclides in the gap is not in the same proportion as the fuel pellet inventory since the migration of each nuclide into the gap is dependent on its particular diffusion rate. The relative ratios of the nuclides analyzed during an accident may be compared to the predicted relative ratios existing in the gap and fuel pellet to determine the source of the fission product release, i.e., gap release or fuel pellet.

Clad damage is characterized by the release of these fission products, i.e., isotopes of the noble gases, iodine, and cesium which have accumulated in the gap and during the operation of the plant. When the cladding ruptures, it is assumed that the fission product gap inventory of the damaged fuel rods is instantaneously released to the primary system. For this methodology it is assumed that the noble gases will escape through the break of the primary system boundary to the containment atmosphere and the iodines will stay in solution and travel with primary system water during the accident.

Fission product release associated with overtemperature fuel conditions arises initially from the portion of the noble gas, cesium and iodine inventories that was previously accumulated in grain boundaries. In addition, small amounts of the more refractory elements, barium-lanthanum, and strontium are also released.

Fuel pellet melting leads to rapid release of many noble gases, halides, and cesiums remaining in the fuel after overheat conditions. Significant release of the strontium, barium-lanthanum chemical groups is perhaps the most distinguishing feature of melt release conditions.

Auxiliary indicators such as core exit thermocouples, reactor vessel water level, reactor coolant loop radiation monitors, containment radiation monitors, and the containment hydrogen concentration are available for estimating core damage. These indications should confirm the core damage estimates which in turn are based on the radionuclide analysis.

RADIATION MONITOR CONTROL FUNCTIONS

Channel No.	Train	Title	Control Function
R-1	B	Control Room Area Monitor	None
1R-2	B	Unit 1 Containment Area Monitor	None
2R-2	A	Unit 2 Containment Area Monitor	None
R-3	B	Radiochem Lab Area Monitor	None
R-4	B	Charging Pump Area Monitor (11,12,13)	None
R-5	B	Spent Fuel Pool Area Monitor	None
R-6	B	Sampling Room Area Monitor	None
1R-7	B	Unit 1 Incore Seal Table Area	None
2R-7	A	Unit 2 Incore Seal Table Area Monitor	None
R-8	B	Waste Gas Valve Gallery Area	None
1R-9	B	Unit 1 Letdown Line Area Monitor	None
2R-9	A	Unit 2 Letdown Line Area Monitor	None
1R-11	B	Unit 1 Containment Particulate Monitor	Isolates Containment Purge and In-Service Purge
2R-11	A	Unit 2 Containment Particulate Monitor	Isolates Containment Purge and In-Service Purge
1R-12	B	Unit 1 Containment Gas Monitor	Isolates Containment Purge and In-Service Purge
2R-12	A	Unit 2 Containment Gas Monitor	Isolates Containment Purge and In-Service Purge
1R-15	B	Unit 1 Condenser Air Ejector Gas Monitor	None
2R-15	A	Unit 2 Condenser Air Ejector Gas Monitor	None
R-16	B	Containment Fan Coil (12,14,22,24) Cooling Water Discharge Monitor	None
R-18	B	Waste Disposal System Liquid Effluent Monitor	Closes Waste Liquid Common Discharge Header Valve
1R-19	B	Unit 1 Steam Generator Blowdown Monitor	Closes SGB to Flash Tank Valves and Flash Tank to River valve

RADIATION MONITOR CONTROL FUNCTIONS

Channel No.	Train	Title	Control Function
2R-19	A	Unit 2 Steam Generator Blowdown Monitor	Closes SGB to Flash Tank Valves and Flash Tank to River valve
R-21	A	Circulating Water Discharge Monitor	None
1R-22	A	Unit 1 Shield Bldg Vent Gas Monitor	Isolates Containment Purge and In-Service Purge
2R-22	B	Unit 2 Shield Bldg Vent Gas Monitor	Isolates Containment Purge and In-Service Purge
R-23	A	Control Room Air Supply Monitor A	Closes 121 and 122 CR outside air supply dampers; closes CR exhaust steam exclusion damper, 121 CR Cleanup Fan
R-24	B	Control Room Air Supply Monitor B	Closes 121 and 122 CR outside air supply dampers, closes CR exhaust steam exclusion damper, starts 122 CR Cleanup Fan
R-25	A	Spent Fuel Pool Air Monitor	Stops 121 SFP normal supply and exhaust fans, isolates Units 1 and 2 In-Service Purge, starts 121 Spent Fuel and In-Service Purge exhaust fan
R-26	A	RHR Cubicle Air Monitor (11,21)	None
R-27	B	RHR Cubicle Air Monitor (12,22)	None
R-28	A	New Fuel Pit Criticality Area Monitor	None

RADIATION MONITOR CONTROL FUNCTIONS

Channel No.	Train	Title	Control Function
R-29	B	Shipping and Receiving Area Monitor	None
1R-30	B	Unit 1 Aux Bldg Vent Gas Monitor B	Starts 122 Aux Bldg Special Exhaust Fan
2R-30	B	Unit 2 Aux Bldg Vent Gas Monitor B	Starts 122 Aux Bldg Special Exhaust Fan, Closes Gas Decay Tank Release Isolation Valve
R-31	B	Spent Fuel Pool Area Monitor	Stops 121 SFP normal supply and exhaust fans, isolates units 1 and 2 In-Service Purge, starts 122 Spent Fuel and In-Service Purge Exhaust Fan
R-32	A	Rad Waste Bldg Control Station Area Monitor	None
R-33	A	Rad Waste Bldg Second Floor Area Monitor	None
R-34	B	Rad Waste Bldg Cement Dump Area Monitor (Abandoned)	None
R-35	B	Rad Waste Bldg Vent Gas Monitor	None
R-36	B	Charging Pump Area Monitor (21,22,23)	None
1R-37	A	Unit 1 Aux Bldg Vent Gas Monitor A	Starts 121 Aux Bldg Special Exhaust Fan
2R-37	A	Unit 2 Aux Bldg Vent Gas Monitor A	Starts 121 Aux Bldg Special Exhaust Fan, Closes Gas Decay Tank Release Isolation Valve
R-38	A	Containment Fan Coil (II,13,21,23) Cooling Water Discharge Monitor	None

RADIATION MONITOR CONTROL FUNCTIONS

Channel No.	Train	Title	Control Function
1R-39	A	Unit 1 Component Cooling System Monitor	Closes CC Surge Tank Vent Valve
2R-39	B	Unit 2 Component Cooling System Monitor	Closes CC Surge Tank Vent Valve
R-41	B	Waste Gas High Activity Loop Monitor	None
R-42	NA	Heating Boiler Dearator Rad Monitor	None (Abandoned)
R-43	NA	11-12 Filter Demin Rad Monitor	None (Abandoned)
R-44	NA	12-13 Filter Demin Rad Monitor	None (Abandoned)
R-45	NA	21-22 Filter Demin Rad Monitor	None (Abandoned)
R-46	NA	22-23 Filter Demin Rad Monitor	None (Abandoned)
R-47	NA	121 Spent Resin Transfer Tank	None (Abandoned)
1R-48	B	Unit 1 Containment High Range Area Monitor	None
2R-48	B	Unit 2 Containment High Range Area Monitor	None
1R-49	A	Unit 1 Containment High Range Area Monitor	None
2R-49	A	Unit 2 Containment High Range Area Monitor	None
1R-50	B	Unit 1 High Range Shield Bldg Vent Gas Monitor	None
2R-50	A	Unit 2 High Range Shield Bldg Vent Gas Monitor	None
1R-51	NA	11 Main Steam Loop Rad Monitor	None
2R-51	NA	21 Main Steam Loop Rad Monitor	None
1R-52	NA	12 Main Steam Loop Rad Monitor	None
2R-52	NA	22 Main Steam Loop Rad Monitor	None
RE-18305	NA	Drum Conveyor 36 in. Height Monitor	None
RE-18306	NA	Drum Conveyor 2 in. Height Monitor	None
RE-29003	NA	121 Waste Monitoring Tank Area Monitor	None

RADIATION MONITOR CONTROL FUNCTIONS

Channel No.	Train	Title	Control Function
1R-53	NA	Unit 1 SI Pump Area Monitor	None
2R-53	NA	Unit 2 SI Pump Area Monitor	None
1R-54	NA	Unit 1 CS Pump Area Monitor	None
2R-54	NA	Unit 2 CS Pump Area Monitor	None
1R-55	NA	Unit 1 Aux Bldg 695 East Area Monitor	None
2R-55	NA	Unit 2 Aux Bldg 695 West Area Monitor	None
1R-56	NA	Unit 1 Aux Bldg 695 West Area Monitor	None
2R-56	NA	Unit 2 Aux Bldg 695 East Area Monitor	None
1R-57	NA	Unit 1 Aux Bldg 715 East Area Monitor	None
2R-57	NA	Unit 2 Aux Bldg 715 West Area Monitor	None
1R-58	NA	Unit 1 Aux Bldg 715 West Area Monitor	None
2R-58	NA	Unit 2 Aux Bldg 715 East Area Monitor	None
1R-59	NA	Unit 1 Aux Bldg 715 Penet./Ltdn Area Monitor	None
2R-59	NA	Unit 2 Aux Bldg 715 Penet./Ltdn Area Monitor	None
1R-60	NA	Unit 1 Aux Bldg 735 North Area Monitor	None
2R-60	NA	Unit 2 Aux Bldg 735 North Area Monitor	None
1R-61	NA	Unit 1 A Stm Line Area Monitor	None
2R-61	NA	Unit 2 A Stm Line Area Monitor	None
1R-62	NA	Unit 1 Aux Bldg 755 East Area Monitor	None
2R-62	NA	Unit 2 Aux Bldg 755 West Area Monitor	None
1R-63	NA	Unit 1 Aux Bldg 755 West Area Monitor	None
2R-63	NA	Unit 2 Aux Bldg 755 East Area Monitor	None
1R-64	NA	Unit 1 Turb Bldg 735 North Area Monitor	None
2R-64	NA	Unit 2 Turb Bldg 735 North Area Monitor	None
1R-65	NA	Operations Support Center Area Monitor	None
1R-66	NA	D1 Diesel Gen. Room Area Monitor	None
2R-67	NA	Instruments & Control Shop Area Monitor	None
2R-68	NA	Technical Support Center Area Monitor	None
2R-69	NA	Guardhouse Area Monitor	None
2R-72	NA	D6 Bldg 707' Cable Spreading Room Area Monitor	None
2R-73	NA	D6 Bldg 718' Bus 26 Room Area Monitor	None

RADIATION MONITOR CONTROL FUNCTIONS

Channel No.	Train	Title	Control Function
2R-74	NA	D6 Bldg 735' Bus 221/222 Room Area Monitor	None

SAFETY INJECTION SIGNAL

The safety injection signal initiates 18 actions. The full explanation of safety injection actions can be found in reference B18C; "Engineered Safeguards System". Those actions important to the REC and RPSS positions are listed below:

- A reactor trip is initiated
- A Containment Isolation signal is generated (see below)
- A Containment Ventilation Isolation signal is generated (see below)
- A Control Room Ventilation Isolation signal is generated (see below)
- The Shield Building Special Ventilation System starts
- The Auxiliary Building Special Ventilation System starts and the Auxiliary Building Normal Ventilation System stops

CONTAINMENT ISOLATION SIGNAL

The containment isolation signal initiates 11 actions. The full explanation of containment isolation actions can be found in reference B18C. Those actions important to the REC and RPSS positions are listed below:

- Letdown isolates **(R-9 reading no longer reflects the current RCS activity)**
- The steam generator blowdown isolation valves close **(R-19 reading no longer reflects the current steam generator blowdown activity)**
- The containment vacuum breaker closes (unless a negative pressure exists in containment)
- The pressurizer sample line isolates
- R-11 and R-12 isolate **(R-11 and R-12 readings no longer reflect the current containment atmosphere activity)**

CONTAINMENT VENTILATION ISOLATION SIGNAL

The containment ventilation isolation signal initiates 2 actions. The full explanation of containment ventilation isolation actions can be found in reference B18C. Those actions important to the REC and RPSS positions are listed below:

- The containment purge supply and exhaust dampers close
- The in-service purge supply and exhaust dampers close

CONTROL ROOM VENTILATION ISOLATION SIGNAL

The control room ventilation isolation signal initiates 2 actions. The full explanation of control room ventilation isolation actions can be found in reference B18C. Those actions important to the REC and RPSS positions are listed below:

- The Control Room intake duct damper closes
- The 121 and 122 Control Room cleanup fans and chillers start

Calculating Field Gas Activities

Please reference the tables in Prairie Island EPIP F3-15 and Monticello EPIP A.2-410 for information on calculation of field gas activities.

Guidance for Selecting Geopolitical Subareas when Making a Protective Action Recommendation

Attachment A of PINGP 577 “Emergency Notification Report Form” contains a table for selecting Geopolitical Subareas when making a Protective Action Recommendation (PAR). The table has, for wind speeds > 5 MPH, line items for a range of “wind from” in degrees. Each range of upwind directions lists the five affected downwind sectors and affected geopolitical subareas. Twelve of the 16 wind direction ranges bisect a small portion of a geopolitical sub-area on the outermost sector and that geopolitical subarea is not listed as one of the affected areas at either 5 miles and/or 10 miles. The table in PINGP 577 is consistent with EP implementing procedure F3-8 “Recommendations for Offsite Protective Actions”.

PINGP utilizes 5 downwind sectors for evacuation. In the 1980’s the Geopolitical Sub-Areas and which areas were affected by which downwind sectors were determined. The geopolitical areas were determined in conjunction with the downwind sectors and were agreed to by the states of Minnesota and Wisconsin. The site and the states realized that certain “slivers” of geopolitical subareas were bisecting a downwind sector at the outer edge of the five downwind sectors. All entities agreed that the PARs would not automatically include the geopolitical subareas which had a small sliver if the area in one of the outermost downwind sector. The Geopolitical Sub-Areas are drawn to existing political jurisdictions and cannot conform exactly to downwind sectors. The parties involved (site and the states) review the boundaries periodically as part of the review of their emergency plans. These Geopolitical Sub-Areas boundaries have not changed since they were established in the 1980’s.

Guidance for spraying a radioactive plume with water

Condition Evaluation

CAP Number: 01522424

Description of problem:

This CAP documents a question resulting from the 5/17/2016 drill critique. During some drills one of the actions the EOF staff is tasked to do is bring in a fire truck to spray down a possible radioactive plume emitting from the shield building stack. It was questioned if spraying down the stack is an appropriate action for a drill or actual event. It is not entirely clear that this is a desirable thing to do. Would spraying the plume down onto the aux building roof result in increased control room dose? Would spraying the plume right at the exit of the vent affect the flow characteristics of the vent system? Would spraying the plume defeat the design basis for vent system as described in the USAR? It was also questioned if the RW fire dept. would even have the equipment to reach the exhaust point for the shield building vent system. If the station determines that this is a beneficial action to take, we should perform a modification to have a permanently installed spray system so the fire department can respond to other calls during an event.

Evaluation Details:

The PINGP Emergency Plan Section 5.6.4 states that one of the RW fire dept. tasks includes spraying to contain radiological release.

*“**5.6.4** Local Support Services*

A. Fire Fighting

The Red Wing Fire Department will provide assistance in the event of a fire occurring at the plant. The duties and responsibilities of the Plant Fire Brigade, insuring complete coordination with the Fire Department, are covered in the Operations Manual, Section F5, Fire Fighting.

…The Red Wing Fire Department has various firefighting apparatus and water pumping equipment available for use. All Red Wing Fire Department apparatus can perform both fire fighting tasks, including rescue, and non-fire fighting tasks, including spraying to contain radiological releases…”

Guidance for spraying a radioactive plume with water

Condition Evaluation

From the letter of agreement with the City of Red Wing,

LETTER OF AGREEMENT FOR EMERGENCY RESPONSE SERVICES

BETWEEN THE CITY RED WING, CITY OF RED WING OFFICE OF EMERGENCY MANAGEMENT, CITY OF RED WING POLICE DEPARTMENT, CITY OF RED WING FIRE DEPARTMENT, AND NORTHERN STATES POWER COMPANY, A MINNESOTA CORPORATION

11. The Red Wing Fire Department has the capability to and will provide fire, rescue and other non-fire fighting services within the Red Wing Fire Department's Minnesota and Wisconsin service areas in PINGP's emergency planning zone. The Red Wing Fire Department has various firefighting apparatus, including pumpers and an aerial platform. All Red Wing Fire Department apparatus can perform both fire fighting tasks, including rescue, and non-fire fighting tasks, including spraying to contain radiological releases and pumping water into the plant for refilling and cooling purposes. In all cases, such operations can begin once the radiological and security threats are mitigated to insure the safety of both plant personnel and fire fighters. The Red Wing Fire Department facility shall be the location for the Emergency Worker Decontamination.

From these two documents it's clear that there is intent to spray water on a radioactive release if it's safe to do so in order to help contain the release. No PI generated calculation or study or other documentation could be located that quantified the effect of spraying water on a plume.

If you use URI (dose assessment software) and a postulated release from the shield building vent with a release rate of 2.8E9 µCi/sec the release is of sufficient size to exceed the protective action guidelines (PAGs) at the site boundary. This meets the definition of a General Emergency.
Using an input for heavy rain to simulate spraying the release you find that the TEDE value at the site boundary is higher than not spraying.

Precipitation	*TEDE (mRem) SB*
None	*1.02E3*
Heavy Rain	*1.51E3*

There is a difference of almost 500 mRem TEDE. (32%)

Guidance for spraying a radioactive plume with water

Condition Evaluation

The ground deposition is also higher with heavy rain due to the water stripping radionuclides out of the plume. Because more radionuclides are removed close to the discharge site there are fewer contributing to accumulated dose downwind. In the example used the TEDE at 10 miles with no precipitation is higher than the TEDE with heavy rain.

Precipitation	*TEDE (mRem) 10m*
None	*36*
Heavy Rain	*27*

The difference is 9 mRem TEDE. (25%)

The actual difference in TEDE is not high and the accumulated dose at the discharge site is not enough to impact emergency response actions. There is a measurable benefit to the public and the method of spraying water referenced in the Emergency Plan and practiced by the ERO should continue. The probability of needing to perform this action is very low and is not required by any regulation or licensing bases. No plant modification is recommended at this time.

Actions:

OTHA 01522424-02 Communicate the CE results to ERO duty teams.

Preparer's Name: Brian J Carberry Date: 6/22/2016