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ENGINEERING ANALYSIS AND TEST COMPANY, INC.

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FINAL REPORT

**MONITORING INSTRUMENTATION
SPENT FUEL MANAGEMENT PROGRAM**

PO. 2592309

PREPARED FOR

**WASTE MANAGEMENT PROGRAM
LAWRENCE LIVERMORE LABORATORY**

MASTER

SPECIALIZED ENGINEERING SERVICES • NUCLEAR ENGINEERING
MECHANICAL ENGINEERING • EARTHQUAKE ENGINEERING
STRUCTURAL DYNAMICS • COMPONENT QUALIFICATION
SOFTWARE DEVELOPMENT

Project: 1016-3

Date: JANUARY 1979

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NOTICE

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ABSTRACT

Preliminary monitoring system methodologies are identified as an input to the risk assessment of spent fuel management. Conceptual approaches to instrumentation for surveillance of canister position and orientation, vault deformation, spent fuel dissolution, temperature, and health physics conditions are presented. In future studies, the resolution, reliability, and uncertainty associated with these monitoring system methodologies will be evaluated.

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1.0 Introduction

The Lawrence Livermore Laboratory under contract to the Nuclear Regulatory Commission (NRC) is in the process of developing a radioactive waste management information base and risk assessment methodology for spent fuel management.

The spent fuel management program as currently envisioned consists of the following scenario. The spent fuel is stored at the reactor for short periods of time before it may be transferred to an interim off-site storage pool called Away From Reactor (AFR) storage. Following a period of interim storage which may approach 20 years, the spent fuel will be shipped to the Federal Repository. Repackaging as necessary to meet the criteria of the Federal Repository will be accomplished. The repackaged spent fuel may be stored at the Federal Repository in underground retrievable storage or disposed of in a deep geologic formation.

This report presents the preliminary results of a study on the spent fuel management program.

In this study, package condition monitoring methodologies within the repository are to be evaluated. In this work, emphasis is placed on the evaluation of monitoring procedures, instrumentation, data collection, processing, interpretation and analysis. Particular attention is being placed on the determination of the levels of resolution, reliability and uncertainty associated with the monitoring methodologies as affected by the environment.

For the sake of this work, resolution is defined as a measure of the accuracy of the sensor-logic system. Reliability is defined as a measure of the continued functionality of the sensor-logic system. Uncertainty is defined as a measure of the reliance on detection of a consequence bearing event given that the sensor-logic system is functioning.

The environmental parameters under study include monitoring for package position and orientation, vault deformation, brine migration, spent fuel dissolution, temperature, and health physics.

2.0 Executive Summary

The preliminary results to date of this study indicate that package condition monitoring would indeed impact risk assessment of spent fuel management. Monitoring of the repository would provide an information base upon which the containment capability of the spent fuel burial packages could be evaluated on a real time basis. Uncertainties in the modeling assumptions can be reduced through monitoring validation of package behavior.

Monitoring methodologies for vault deformation, temperature, and health physics can be based in part on conventional technologies using procedures, instrumentation, and data acquisition systems for which considerable operating experience exists. Data interpretation and analysis will, however, represent developmental technologies requiring engineering and scientific analyses.

Monitoring methodology for package position and orientation encompasses developmental technologies. However, with regard to instrumentation, considerable experience exists with radar and sonar

echo reflection systems utilized in salt mines to locate inclusions (1 cm or bigger) during mining operations. The effect of the combined temperature, stress, and radiation gradients adjacent to the buried spent fuel canisters on the physical and chemical properties of the rock salt, which impact signal transmission characteristics, are unknown at this time. For this reason, data interpretation and analyses techniques are presently undefined.

Monitoring methodologies for spent fuel dissolution may be based in part on methodologies established for nuclear fuel reprocessing plants and other facilities handling high level waste. The high specific activities of the spent fuel during interim storage clearly indicate the desirability for utilization of nuclear detector instrumentation. Data acquisition, processing, and analysis technique development will be required.

Monitoring for brine migration may well represent the single most important environmental parameter for surveillance. Intrusion of major amounts of ground water has been identified

as a potential waste release and dispersion mechanism. The monitoring methodologies for brine migration investigated in this report have never been demonstrated insitu. For this reason they are the least well defined of the methodologies herein studied. However, resistivity measurements, radar, and sonar all appear to be potentially feasible techniques.

Before an assessment of the resolution, reliability, and uncertainty can be attained, the monitoring system methodology must be defined in terms of procedure, instrumentation, and data collection, processing, interpretation, and analysis. This report presents preliminary data on monitoring instrumentation sensors which might be employed.

3.0 Summary and Conclusion

In this study monitoring instrumentation for spent fuel element waste canisters buried in deep geologic storage salt mines were investigated and evaluated. These included monitoring for the following parameters: canister package position and orientation; bulk deformation of the salt vault repository; brine migration; spent fuel dissolution; temperature; and health physics.

The studies concluded that the determination of package position and orientation could, with some developmental engineering effort, be accomplished with either acoustic or electromagnetic (EM) reflection technologies.

The determination of vault deformation can be effected by conventional means using mechanical displacement, strain, and pressure gauges. In addition, highly accurate laser instrumentation is available and with minimal engineering development can be utilized.

Further, electromagnetic and acoustic reflection technologies would appear to have

application in the geophysical mapping of fissures, interstitial shale layers, salt repository boundaries, etc.

Developmental effort and demonstration of techniques appear to be necessary before the overall migration of brine in the repository can be monitored. However, identification of local migration of brine in the vicinity of individual canisters would appear to be technically feasible through the use of resistivity measurements. To a greater level of uncertainty, pockets of brine solution and penetration of ground water along fissures or geophysical boundaries might be detected using electromagnetic and acoustic reflection technologies.

Occurrence of spent fuel dissolution can most accurately be determined using nuclear detectors due to the high specific activity of the waste. Methodologies based on the use of Geiger-Muller tubes, scintillation crystals and thermoluminescent dosimeters (TLD) have been developed in the past for monitoring high level waste storage operations. The use of radionuclide gas sippers would appear to be a promising means of locating canisters

with pinholes or cracks.

Temperature monitoring of the salt repository can be accomplished using thermocouples buried in three dimensional arrays. In general, the determination of temperature gradients within the repository can be accomplished using conventional instrumentation systems and methodologies.

Health physics monitoring will be required and will represent an independent assessment of radionuclide isolation. However, health physics monitoring should be designed to compliment canister monitoring objectives. As in light water reactor (LWR) and fuel reprocessing operations, health physics monitoring frequently leads to the determination and identification of operational problems.

4.0 Package Position & Orientation Monitoring Instrumentation

In order to effect retrieval, the prior knowledge of the position and orientation of the buried spent fuel canisters is required. This position and orientation may well have changed since the initial emplacement (fig. 4.1) due to the environment conditions within the repository.

The heat output of the spent fuel will cause expansion in the emplacement layer and in overlying layers and will cause some resulting uplift.²⁰ This uplift may cause compaction of the backfilled overlying layers and the uplift may be somewhat offset. Ultimately the heat output of the waste decays away, and the heat in the overlying layers is conducted away to the surface or carried away by convection in an aquifer (fig. 4.2). Then the overburden will cause settlement of these layers to perhaps a lower position than they occupied at canister emplacement.

A defective canister, which is roughly 16 ft. long and 1 ft. in diameter, can release radiation in doses as high as one million rems per hour. Due to the complexities of retrieval of such a canister, whose physical integrity may have deteriorated, determination of its exact position and orientation is desirable.

Monitoring instrumentation investigated included electromagnetic radar, sonic sonar, magnetic interference, infrared scanning, and gamma and neutron scanning. Based on an extensive literature survey it would appear that systems based on electromagnetic and acoustic reflection are at the greatest level of development. Experience exists in utilizing both these systems in salt mine explorations. The other systems investigated have no experience in salt mine applications. However, theory would tend to indicate that through design and developmental efforts these other techniques may be applicable in the future.

4.1 Electromagnetic Radar

In the last 15 years extensive research has been conducted on the use of radar for exploration for discontinuities in salt mines.^{1,14,15,16} The radar frequencies employed are dependent on the size of the discontinuity and the depth of burial.

The loss of energy through attenuation is proportional to the frequency of the wave used by the radar. The lower the frequency is, the lower the attenuation is, so therefore, low frequency radar can probe deeper into the salt mine. However, when the frequency is low, the wavelength is long, so the resolution of the object being observed is less precise.

The equation for the attenuation of an electromagnetic wave in salt is $\alpha = \frac{2\pi \tan \delta}{\tau} \sqrt{(\epsilon'/\epsilon_0) (M'/M_0)}$ where: (ϵ'/ϵ_0) = the relative electric permittivity of salt (≈ 6.0), (M'/M_0) = the relative magnetic permeability (≈ 1), δ = loss tangent of transmitting material, and τ = wavelength in air.¹⁴

Most radar systems used in salt mines operate in the $10^6 - 10^9$ Hz region. If high resolution is desired and one is willing to sacrifice permeability the higher frequency radar may be utilized.

In order to locate spent fuel canisters buried within a salt mine repository, high resolution is necessary but only about 15m (50 ft.) of salt needs to be penetrated. A radar system with a frequency of 4.3×10^9 Hz can penetrate 21m (70 ft.) of salt and does have a high degree of resolution. A particular radar system which has been utilized in salt mine explorations is called Echo II and it has a signal of 4.3×10^9 Hz.¹⁴ (See fig. 4.3)

The Echo II radar is a system designed at Texas A & M University to make accurate short range measurements in salt mines. The following is a list of parameters pertaining to the Echo II radar system.

Basic frequency of operation	4.3×10^9 Hz
Range of FM sweep, Δf	70×10^6 Hz
Modulated frequency, FM	120 Hz

Wavelength in air	6.98 cm
Wavelength in salt	2.85 cm
Power output	1.5 W
Antenna beam width in salt	
transmitter horn	$\pm 3^\circ$
receiver horn	$\pm 3^\circ$

The advantages of this system for locating canisters are a short wavelength (2.85 cm) for high resolution, the horn shaped antenna which creates a narrow beam width ($\pm 3^\circ$), and the ability to operate at a very close range to the target, (2 ft).

Theoretically, a radar signal can be resolved down to about $1/12$ of the wavelength. That is, if a radar used a 12 ft. wavelength, it could determine the position of an object to within a foot. A radar with a frequency of 4.3×10^9 Hz has a wavelength in salt of 2.85 cm. Theoretically a radar of this frequency could resolve the distance to the target to the nearest .24 cm.

The Echo II radar system, however, cannot resolve down to .24 cm. The electronics of the

system are not sophisticated enough to be that accurate. The designer has stated the radar system could be accurate to within an inch, given ideal conditions where the index of refraction of the salt and its impurities were known very accurately.

However, the index of refraction, and hence the speed of light in salt, is not the same in all mines, or at two different locations in the same mine. The velocity of electro-magnetic radiation varies from 51.2×10^6 m/s to 61.0×10^6 m/s in different salt mines.^{14,15} Thus, if the speed of electromagnetic wave propagation in the mine where the canisters are stored is not determined, the measurements would have an error of on the order of 15%. Thus, if a measurement was being made 40 ft. away, the error could be $40 \times 15\% = 6$ ft.

However, the speed of electromagnetic wave propagation at a particular site can be established by direct measurement and that value can be used. In order for the location of a

spent fuel canister 40 ft. away to be found within 1 inch, the velocity of EM in salt must be measured to within 0.2%. This accuracy of measurement is possible if extreme care is taken. The speed would have to be measured in a sample of salt near each canister to account for impurity concentration variation in the mine.

The final source of error inherent in all radar systems is beam width. If the signal hits an object at an angle, the signal will be reflected from different points on the canister at different distances to the radar antenna at the same time. Thus, the readout of the radar indicates that the distance to the canister is not a finite distance, but a narrow band of distances. The width of this band is a function of the beam width (ϕ), and the distance from the radar to the object (d), and the incident angle of the beam on the canister (θ).

$$\text{Therefore: band width} = \frac{\sin \phi \times d}{\tan \theta}$$

For example, if the canister was 40 ft. away and the angle of incidence was 45° , the band width would be

$$\text{band width} = \frac{(\sin 3^\circ) \times 40 \text{ ft.}}{\tan 45^\circ} = 2 \text{ ft.}$$

To limit the width of the readout, any or all of the following can be done: First, the beam width could be made narrower. This is directly proportional to the readout band width. Second, the distance, d , from the radar to the canister could be shortened. Distance d is also directly proportional to the readout band width. Finally, the angle of incidence could approach a right angle. The band width is inversely proportional to the tangent of this angle, and by approaching 90° the error due to the incident angle could be reduced. By implementing these precautions, the error in the radar system due to bandwidth (i.e. striking the canister at an angle with the radar beam) can approach zero.

4.2 Acoustic Sonar

High frequency sonic waves of 24 kHz have been used to detect fractures, shale layers, wet areas, and objects in salt mines.¹⁴ The sonar system employed had a range of 400 meters in dry salt and 215 meters in wet salt. This system would appear to have application in the determination of canister position and orientation buried within a salt mine repository (fig. 4.4).

Sonic systems have an advantage over the electromagnetic radar system in that the sonic waves are not absorbed to the extent EM waves are when traveling through wet areas. It has been suggested that due to brine migration, the region around the canisters will be wet.

The velocity of sound waves in salt at pressures of around 0.3 kbar (3000 psi) and a temperature of 300°C (80°F) is close to 5.6×10^3 m/sec (1.8×10^4 ft/sec) for compressional waves in pure salt²⁹ and 2.7×10^3 m/sec (8.8×10^3 ft/sec) for shear waves in pure salt.³ In general, rock salt contains impurities and the velocity found²⁷ is 2.2×10^3 m/sec (7.3×10^3 ft/sec).

In order to detect a canister and receive clear non-scattered reflection signals, half the wavelength should be equal to or smaller than the canisters diameter, which is 1 foot.

This corresponds to a frequency $\nu = v/\lambda$ on the order of 4 kHz. Any higher frequency will have a better resolution. The accuracy of detection depends largely on the irregularities of velocity in rock salt. The accuracy is, therefore, estimated to be on the order of a wavelength.

At 24 kHz the wavelength is approximately 9 cm (0.35 ft) and consequently the accuracy of location is on that order. The accuracy can be improved by shortening the wavelength through using a higher frequency. As with the radar system, however, a trade off exists between resolution and penetration depth.

<u>Frequency</u>	<u>Wavelength</u>	<u>Depth of Penetration</u>	<u>Resolution & Accuracy</u>	<u>Tilt</u>
(kHz)	(cm)	(m)	(cm)	
4	66	1290	66	16°
24	9	215	9	2°10'
100	2.54	52	2.54	35'
200	1	23	1	15'

The penetration depth at 24 kHz is on the order of 215 meter.¹⁴ This is an average value. Locally and for different salt mines the real range can vary from 374 m (1122 ft) for dry salt to 100 m (325 ft) for wet salt. The absorption varies proportionally to frequency and the intensity falls off exponentially with the distance (ref. 5 page 241).

"The intensity, I , is given by:

$$I = I_0 e^{-\alpha x}$$

I_0 initial intensity

α absorption factor

x distance

With the relation $\alpha\lambda = \text{constant}$ for each type of rock, and with $v = v/\lambda$, the relationship $v_1 x_1 = v_2 x_2$ is found, where $v_1 x_1 = \text{frequency 1} \times \text{depth of penetration 1}$ and $v_2 x_2 = \text{frequency 2} \times \text{depth of penetration 2}$. Therefore, if a higher frequency and higher accuracy or resolution is desired, the range (depth of penetration) will be less.

The tilt and position of the canister can be obtained by measuring the position of the top and bottom of the canister by using reflections and by lowering receivers into boreholes situated

around the canister. The accuracy of tilt will be known by angle of

$$\tan \beta = \frac{\text{accuracy of position}}{\text{length of canister}/2}$$

$$= \frac{1 \text{ cm}}{243.84 \text{ cm}} = 0.0041$$

$\beta = 15'$ for a frequency of 200 kHz
and a canister 16 ft. long.

It is to be noted that this value of the tilt angle represents a theoretical maximum. The actual value will be somewhat greater due to inhomogeneities in the salt which result in variations of the speed of sound spacially around the canister.

4.3 Resolution, Reliability, Uncertainty

Limited experience exists in the utilization of radar and sonic echoing systems for the location of metallic objects buried within salt mines. The effects of temperature, brine, pressure, porosity, and impurity gradients on the transmission of either radar or sonar waves through the salt mine repository is not well defined. However, as the resolution of location is theoretically proportional to the wavelength, it would appear that high frequency radar and sonar echoing methodologies could provide the required accuracies for safe retrieval.

Additional study will be required to optimize the scanning frequencies, procedures, instrumentation, and data acquisition methodologies. Without a fully defined monitoring system, the system reliability and the operational uncertainty can not be realistically addressed.

BURIAL PLAN FOR DEPLETED NUCLEAR FUEL RODS

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Fig. 4.1

Fig. 4.2

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 BY JOHN FORGRAVE

BLOCK DIAGRAM OF CHARLIE II RADAR SYSTEM

Fig. 4.3

Fig. 4.4

5.0 Vault Deformation Monitoring Instrumentation

Salt vault deformation is caused by stress applied to the walls, ceiling, and floor of the surrounding rock masses. After the galleries are excavated at a depth of approximately 600 meters (1800 ft), for example, this stress level is roughly 0.27 kbar (4ksi).^{2a} Under these conditions, the salt plasticly deforms and creeps into the available cavities.

If the temperature of the surrounding rock salt is increased, the deformation is accelerated due to expansion of the salt and creep increase. The maximum temperatures to be encountered in the vault are expected to be less than 200°C (392°F).³

The expected floor-to-ceiling movement and wall to wall movement is expected to vary from 0.04 to 0.4 in/in in the first 70 days.⁸ The total movement can be as much as 120 cm (4 ft) expansion on the floor and 60 cm (2 ft) deformation on walls and sag of ceiling.

5.1 Mechanical Deformation Gages

The deformation gages to be considered for vault monitoring must satisfy several conditions. They have to be able to measure large deformations of up to 120 cm (4 ft), be noncorrosive in salt, withstand temperatures of up to 200°C (392°F), and measure the absolute deformation. This last requirement poses unique difficulties.

The entire repository will be undergoing deformation and it is very difficult under these conditions to find a fixed reference point that is not moving. In 1964 Dr. W. C. McClain^a solved this problem by anchoring reference gages deep into the salt where no deformation was expected. The galleries under study were single layered. Most of the reference gages were anchored 7.5 m (25 ft) deep and one was anchored 15 m (50 ft) deep.

5.1.1 Pipe-Type (F) Gage:

This displacement gage measures rib-side convergence or vertical convergence which means that it can measure the vertical shortening of pillars and walls, convergence of ceiling and floor, and wall to wall distances (fig. 5.1).

The gage consists of a 0.5 inch diameter pipe anchored vertically in a shallow hole in the roof by a roof-bolt anchor and a shorter piece of pipe anchored by a bolt in the floor vertically beneath it. A gap is left between the two ends of the pipe and their convergence is measured with a dial gage. As the pipes are anchored very shallowly in the salt, this type of gage measures the relative convergence rate of ceiling to floor, and not the absolute movement of floor and ceiling. The accuracy obtained with this gage is 0.0005 inches.⁸

In order to reduce the uncertainty due to horizontal movement of layers, a third dimension should be added by anchoring gages to the walls.

These pipes should be kept in horizontal and vertical position to avoid any uncertainties.

The reliability depends on the anchorage of the roof-bolts in the salt and on the corrosion rate of the metal pipes and anchors in contact with the salt.

5.1.2 Internal Strain Gages (Type A)

The horizontal strain gage consists of a number of wires anchored, one at the collar of the borehole, one at the bottom of the borehole that is not affected by deformation (25 feet in the McClain experiment)⁸, and several others at midway. The lowest anchored wire is considered as fixed and the movement of all the other wires are measured with respect to that one (fig. 5.2 shows only 2 wires). The movement of the pipe fixed to the collar with respect to the immobile wire gives the absolute movement of the salt. The other anchored wires give a relative movement. The depth of anchorage of the "fixed" wire is fixed at a depth or distance that does not undergo any deformation, and is calculated from the thermal and stress model. In the case of pillars, the anchorage is made in the middle of the pillar which is assumed to be fixed. This assumption for the waste repository is close to being correct.

The gage is composed of a number of wires anchored by roof-bolt anchors. The collar of the hole is filled with another roof-bolt anchor to

which is attached a 12 inch length of pipe and a threaded boss. Stainless steel wires come through the boss and terminate with a small brass nipple clamped to the wire. The gage is read by an extensometer screwed on to the collar anchor which serves as reference position. The wire nipples are attached to a tensioning spring by means of an adapter. The reading is done by comparing the deflection of the spring to a standard deflection by means of a dial gage. The expected reading errors are 0.002 inches. This gage was developed for Dr. McClain in 1965 by Prof. E. L. J.

Potts, Department of Mining Engineering University of Newcastle-Upon-Tyne, England.⁸

The accuracy of measurement depends also on the exact vertical and horizontal position of the wires and anchors, on how solid the anchors sit in the salt and on the depth of the lowest anchor in the salt. If the lowest anchor gets loose, the measurement is lost. The reliability depends on the corrosion rate of the wires.

5.1.3 Roof Bed Gages

It was noticed in the McClain⁸ experiment by reading vertical internal strain gages anchored at several distances in the roof that the roof was sagging as a whole unit by two feet. The anchor fixed at a height of 25 feet into the roof did not move and was taken as a reference point. Salt layers above the roof separated out from shale layers and other impure strata and were sagging as a whole unit. The sagging of the roof was read by a potentiometric displacement transducer mounted on the sagging bed and connected to a wire anchored above the sagging roof. In a control room, the movement of the roof could be observed remotely.

The accuracy of this gage was 0.05 inches and for that reason it was considered to be a very gross movement indicator.⁸

5.1.4 Floor Leveling Gages

Floor leveling gages can be used to measure the uplift of the floor due to thermal expansion of the salt. Rivets are driven into the ground. The rounded head of the rivet serves as a seat for a leveling rod. The elevations of these points are determined relative to an assumed fixed point. A fixed point is obtained by using a bench mark put at a minimum of 475 feet from the experimental area down the gallery, and this bench mark is anchored 25 to 30 feet deep.⁸

The leveling is carried out with a autaset engineer's level and a leveling rod fitted with a vernier target (e.g. Zein Model NI-2). The error read with this instrument is accurate to 0.0002 ft.

5.1.5 Borehole Leveling

Measurement of the gross vertical deformation of the pillars at various distances into the pillar, including its center line can be accomplished with this instrument.

The measuring instruments are illuminated leveling targets that are put in horizontal 5 cm (2 inch) holes in the pillars. The reading is done on a screen. Two such instruments are put one above the other and their relative movement with respect to each other is measured and the vertical deformation deduced.

This type of instrument can also be installed in walls in order to determine the borehole leveling. The accuracy obtained is .16 cm (1/16 of an inch).⁶

5.2 Stress Gages

In contrast to the preceding gages which measure deformations along the borehole axis, the diametral deformation of the boreholes provides some information about the vertical load or change in load on the canisters.

5.2.1 Diametral Deformation Gages

Diametral deformation gages measure the state of stress in a plane normal to the borehole.¹⁰ Measurements are made across at least three different diameters (fig. 5.3). The points of contact are 6 movable pistons. As the borehole deforms during overcoring, the piston activates a cantilever strain transducer on which resistance strain gages are mounted. The 4 strain gages on opposite transducers are connected to form a 4-arm bridge. In this way the electrical resistance in each strain gage can be measured. The accuracy of these gages is 1 micro inch across each diameter.

In order to measure the stress on the canister, the stress at different distances from the canister can be measured and an interpolation method used to deduce the stress on the canister. A direct measure of stress on the canister is very difficult to obtain because of the very high corrosion rate of instruments in the vicinity of the canister and the presence of a high irradiation field.

The diametral deformation gage is not for long

term installation and reading in the mine. It is used as a surveying tool that is put into the ground, the closure rate upon the gage is measured and the gage is then retrieved from the salt.

5.2.2 Pressure Gages

The detectable changes of rock pressure in the direction normal to the plane of the pressure cell is accomplished with this instrument. These gages are flat metallic cells (flat jacks) oriented horizontally. ¹⁰ The vertical stress on the gage can be read with this pressure gage.

5.3 Laser Geodolite 3G

The Spectra-Physics^R Geodolite 3G is a laser using Helium-Neon gas and is used for measuring very high precision distances and deformations (fig. 5.4). It was used by USGS for measuring the earth strain.¹⁸ Another application was the detection of structural deformation of dams.

The accuracy is very high and is unequalled in industry by any other device. For distances less than 1 Km the accuracy is $\pm 1\text{mm}$.¹⁸

Apart from its high accuracy, another advantage is that it performs non-contacting measurements and the target is not altered by its installation. The Geodolite transmits an amplitude modulated C W (Continuous Wave) laser beam to a retroreflector or to any natural surface. A small fraction of the light is reflected from the target and is received in a telescope where it is converted to an electrical signal. The phase delay between the transmitted light and the received light is measured and the distance to the target determined. In order to calculate a point to point distance (eg wall to wall) the Geodolite 3G

is used with multiple cooperative targets. The laser system is measuring relative deformations if wall to wall and floor to ceiling measurements are made. The absolute deformation can be obtained by referring the measurements to a fixed point, a bench mark, anchored deep into the rock salt, or by taking the fixed point as a cooperative target.

5.4 Resolution, Reliability, Uncertainty

The vault deformation due to thermal expansion and plastic creep of the rock salt may be determined to a high level of accuracy using conventional instrumentation systems and methodologies. The relative displacement of the vault rooms may be determined to within less than 0.00254 cm. (.001 inch),⁸ and the absolute displacement of a fixed point to within 0.100 cm. (.39 inch).⁸

The reliability of the vault deformation instrumentation systems is to a large extent dependent on their corrosion resistance.

The uncertainty associated with the data interpretation and analysis is dependent on human factors involved in the instrument calibration, data collection and processing.

RELATIVE DEFORMATION GAUGE

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Fig. 5.1

Fig. 5.2 43

THREE COMPONENT BOREHOLE DEFORMATION GAGE

Fig. 5.3

6.0 Brine and Water Migration Monitoring Instrumentation

There are at least two different types of brine and water migration expected at the repository:

(a) Brine migration in the vicinity of the canisters is expected due to the temperature gradient resulting from the heat generation of the nuclear waste. The percentage of maximum brine expected to be present is $1.4 \text{ wt}\%$ ²⁹ compared to the normal brine content in rock salt which is less than $0.18 \text{ wt}\%$.⁴

(b) Brine and water migration through cracks, fissures and faults. Cracks and fissures can be generated during mining and although they are self-healing in salt, they might re-open and formation of new ones might be generated by expansion of the salt under temperature (fig. 4.2). Natural brine pockets in the salt and water sources overlying and underlying the salt might infiltrate the salt through the cracks, fissures, and shaft and produce wet areas.

The brine content around the heat sources will be imbalanced. Although salt is very impermeable, brine and water possibly contaminated with radionuclides could conceivably migrate through geologic imperfections. For this reason brine migration within the storage vault should be monitored. A second reason for monitoring the brine content around the canister and instruments, is the high corrosion of most metals in salt brine. The more brine present, the higher the corrosion risk of the canister and instruments.

Methods for monitoring brine migration under study include:

- (1) Resistivity measurement
- (2) Electromagnetic probing
- (3) Acoustic reflection

6.1 Resistivity Measurement

Resistivity in rock salt varies with temperature and, to a greater extent, brine content. The resistivity of dry rock salt is very high, while the resistivity of brine filled rock salt is very low. An example of the effect which varying temperatures and brine content have on the resistivity of rock salt is shown in Table 6.1 below:

Table 6.1
Resistivity At Varying Temperatures &
Brine Content

Temperature (°C)	Composition			
	Dry & pure Rock Salt	Rock Salt + impurities + 0.08% moisture	pure saturated brine	Water
20	$4.6 \times 10^{16} \Omega \text{ cm}$	$10^6 \Omega \text{ cm}$	$4 \Omega \text{ cm}$	$10^{-1} \Omega \text{ cm}$
100	$1.4 \times 10^{13} \Omega \text{ cm}$		$1.5 \Omega \text{ cm}$	$3 \times 10^{-3} \Omega \text{ cm}$

The resistivity method uses an artificial source of current which is introduced into the ground by current electrodes (fig. 6.1). The procedure is then to measure, at different points, the potential and current. Thus it is possible to determine an apparent resistivity of the subsurface (of the salt, in this case). If the electrode spacing is

small, the apparent resistivity corresponds to the surface resistivity of a homogeneous surface (fig. 6.4). For large electrode spacing the apparent resistivity is the volume resistivity of a uniform ground.

6.1.1 Equipment for Resistivity

The different components necessary for resistivity measurements are (fig. 6.2):

- power source
- meters for measuring current & voltage or resistance
- electrodes, cables & reels.

a) Power source:

The power source may consist of either a direct current or low frequency alternating current supply. D.C. batteries of 45 to 90 volts might be used as power source and connected in series to give several hundred volts. For large scale work a motor-generator of several hundred watts is used instead. Because of self-polarization the d.c. polarity should be inversed regularly.

A portable source of a.c. power is a low-frequency sine-wave transistor oscillator with transformer output of a few watts. Larger sources can be obtained from a motor-driven alternator. A direct current source permits direct resistivity measurements but has the disadvantage that it also measures spontaneous potentials. Therefore, an alternating current source can measure lower resistivities than a direct current source.

b) Meters

The current is measured with a d.c. milliammeter (range of 5-500 mA) and the potential is measured with d.c. volt meter (range of 10 mV to 20 V). Corresponding a.c. meters are used when measuring a.c. power.

c) Electrodes & Wire

When utilizing a.c. power, all of the electrodes may be steel, aluminum, brass, or stainless steel. The last is the most suitable for use in salt. When utilizing d.c. power, the potential electrodes should be porous pots, because of spontaneous potentials.

Experiments would have to be performed to determine the effect of brine pore size, temperature, surface effects, and the presence of the canister on resistivity values.

6.1.2 Electrode Spread

There are numerous electrode spreads used in resistivity. The most commonly used is the Wenner spread⁵ (fig. 6.3). In the Wenner spread the electrodes are uniformly spaced in a line. The apparent resistivity in this case is:

$$\rho_a = \frac{2\pi\alpha\Delta v}{I}$$

where α = spacing between electrodes, Δv = potential read between P_1P_2 , and I = current input.

For exploration at depths, the electrodes are expanded about a fixed center. For surface exploration, the electrodes are displaced as a whole but their spacing does not change. In order to get a more extensive measure of resistivity with depth the electrodes can be buried at varying depths, (fig. 6.5) which is called resistivity logging.

6.1.3 Resistivity logging

Three formulas, derived from Archie's Law, are used in well logging to relate the resistivity of rock, the resistivity of fluid in rock pores, the porosity, and the amount of fluid filling the pore spaces.^{5,17} The empirical formula of Archie's Law is:

$$\rho_e = a \phi^{-m} S^{-n} \rho_w$$

with:

ρ_e : electrical resistivity of salt = $\frac{RA}{L}$

ϕ : porosity

S : fraction of pores containing fluid

ρ_w : resistivity of the fluid

a, m : constant $0.5 \leq a \leq 2.5$ & $1.3 \leq m \leq 2.5$

n : concentration factor

R : resistance

A : cross section

L : length

The conduction in rock salt is not electronic, it is electrolytic, which means that the ions in brine such as Na^+ , Ca^{++} , Mg^{++} , Cl^- , . . . etc. are going to carry the current. The resistivity depends consequently on their mobility, size of inclusion, degree of concentration, and dissociation.

Formula 1: Formation resistivity factor F.

$$F = \rho_o / \rho_w \quad \rho_o: \text{ bulk water-wet resistivity of a rock}$$

Formula 2: The formation factor is dependant on the porosity and the permeability.

$$F = 1/\phi^m \quad m: \text{ concentration factor}$$

Formula 3: If the pores are not completely filled with fluid:

$$S_w = (\rho_o / \rho_t)^{1/n}$$

ρ_t : true resistivity derived from ρ_a , the measured resistivity, by accounting for electronic equipment, inhomogeneous ground etc.....

n = saturated exponent,
 $n \approx 2$

The basic methods of resistivity logging are similar to those used in surface-resistivity prospecting. Direct current or a low-frequency alternating current is applied between current electrodes and the potential is measured between two or more potential electrodes. The record is a plot of potential variation versus depth.

Most of the well-logging arrangements are such that they are only suitable to detect large

areas of wet rock salt. To measure the resistivity of small local areas around a canister, the microlog arrangement is probably the best (fig. 6.6).

The buttonsize electrodes are embedded in an insulating pad which is pressed against the borehole wall by means of an expansion device. Since the electrodes are so closely spaced (1½ and 2 in. apart) horizontal areas can be sharply defined, but the depth investigation is less than 4 inches.⁵

To measure the resistivity of larger areas, 'normal' or 'lateral' electrode arrangement is used (figures 6.7a & 6.7b).

The 'normal' arrangement (fig. 6.7a) has one current and one potential electrode closely spaced in the borehole (16 inches for short, 64 inches for long spacing).^{5,17} The other two electrodes are on top of the borehole far enough away to be considered at infinity. The penetration into the formation is about twice the electrode spacing.

In the 'lateral' arrangement (fig. 6.7b) the down hole potential electrodes are separated

by 32 inches with their center 18 ft. 8 in. from the near current electrode.⁵ The curves obtained by lateral arrangement are asymmetric. The depth of investigation is taken to be equal to the electrode spacing.

6.2 Electromagnetic Radar

Electromagnetic radar probing in rock salt has in recent years proven to be useful for detecting micro-fractures, cracks, brine cavities, (larger than 1 cm) wet areas, impurity layers, salt roof, bottom and flanks, objects, borehole and galleries in salt mines.^{15,16}

In order to determine brine migration, it will be of importance to know the location and the size of the brine pockets. Very little is known to date of what the behavior of the brine contained in those pockets is under a temperature gradient. If the brine cavities are filled with brine the effect on the EM waves amplitude will be very different from an empty cavity. EM waves are heavily attenuated by water. Other areas of concern are 'wet' areas in rock salt, created by water infiltrating through shafts, boreholes, and any other opening. Regular EM radar probings could be used to localize these areas. The

Charlie II radar developed by Unterberger^{14,15}
would be suitable to be used in this case.

With a frequency of 440 MHz it has a penetration
depth of 300 m and a resolution of 28 cm. (fig. 6.8).

6.3 Acoustic Sonar

As discussed previously in this report in
connection with canister position and orientation,
sonar techniques can be used to locate brine
cavities, cracks longer than 1 cm., and wet
areas in rock salt. Such techniques employ
portable random scanning instrumentation based
on echo sounding technologies.

6.4 Resolution, Reliability, Uncertainty

The resolution obtainable in monitoring brine migration is at present unknown. Resistivity measurements have been made by seismologists and mineral prospectors. However, no data exists in the open literature on the measurement of resistivity in salt mines with temperature gradients, radiation fields, brine in varying pore sizes, and adjacent carbon steel structures. However, such measurements have been made by geologists, oil prospectors, etc. to detect water in formations other than salt.

The resolution of EM and acoustic reflection systems in detecting brine presence in varying pore sizes within rock salt is yet to be documented. However, for high frequencies at short penetration depths, both systems show promise.

The resolution, reliability and uncertainty associated with brine migration monitoring can not be assessed until the instrumentation, procedures and data acquisition systems are more fully defined.

Fig. 6.1

PORTABLE EQUIPMENT FOR MEASURING RESISTIVITY

Fig. 6.2

C_1 AND C_2 ARE CURRENT ELECTRODES
 P_1 AND P_2 ARE POTENTIAL ELECTRODES
 a SPACING BETWEEN ELECTRODES

Fig. 6.3

Fig. 6.4

Fig. 6.5

MICROLOG

Fig. 6.6

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Fig. 6.7a

Fig. 6.7b

BLOCK DIAGRAM OF CHARLIE II RADAR SYSTEM

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Fig. 6.8

7.0 Spent Fuel Dissolution Monitoring Instrumentation

The buried spent fuel canister will be at an elevated temperature, the magnitude of which will depend on its age subsequent to its removal from the light water reactor (LWR) and the power history of its exposure. During storage in the repository, due to the presence of a corrosive media, an electrolytic solution and the thermal gradient, some degree of corrosive attack on the carbon steel canister is expected to occur. Should this result in a loss in canister containment capability, corrosive attack and dissolution of the radioactive spent LWR fuel elements might be expected to occur.

The release mechanisms of the radioactive constituents within the fuel elements include direct dissolution, volatilization, and airborne particulate dispersion. Selective absorption of the chemically more soluble constituents of the spent fuel such as cesium, selenium, tellurium, iodine, etc. might be expected to occur. Radionuclides existing in a gaseous state within a spent fuel storage canister might

include ^3H , ^{85}Kr , ^{106}Ru , ^{129}I , ^{134}Cs , ^{135}Cs , ^{137}Cs , and others of lesser concentrations. Micron sized particulates of the radioactive fuel and cladding might be available for dispersion.

The monitoring systems evaluated for detecting spent fuel dissolution include gas sippers, local detectors and beta-gamma scanners. Monitoring methodologies evaluated include the detection of gross levels of activity and the detection of specific radionuclides at distances away from the spent fuel canister.

7.1 Gas Sippers

Gas sippers are used in the LWR utility industry to locate radioactive gas leaking fuel bundles during refueling operations. Such systems consist of chambers into which the fuel bundles are placed. The chambers are then evacuated and gas samples analyzed for the presence of radionuclides. Such a system could conceivably be modified to locate gas leaking spent fuel canisters in the repository. In addition to the detection of gross levels of radiation, energy discriminators (i.e. single and multichannel analyzers) may be used to identify specific radionuclides in the gas samples.

7.1.1 Tritium Detection

Tritium, ^3H exists in a gaseous state as water vapor (HTO) and as hydrogen gas (H_2). Tritiated water, HTO, is measured by liquid scintillation counting. The three most common forms of measuring tritium in air are use of an ionization chamber, gas proportional counter, or removing HTO vapor from the air and measuring it with a scintillation technique.

Flow-through measurement analysis of total tritium in air on a continuous basis using ionization chambers and proportional counters have been conducted for the last 25 years. Several commercially-available instruments now exist. Ion chambers with well developed compensation systems are capable of measuring down to 1 pCi/cm^3 in the presence of gamma fluxes in the few-mrad/hour range.³² However, ion chambers cannot distinguish between other gaseous emitters (eg Ar-41, Kr-85) and are sensitive to cigarette smoke, ions and aerosols.

Proportional counters can partially discriminate against other radionuclides and are

less sensitive to aerosols. Their resolution is comparable to ion chambers. To detect levels below about 1 pCi/cm³ it is necessary to remove tritiated water from the air or use combustion techniques.

7.1.2 Detection of ^{85}Kr

Krypton is a noble gas which is present in the atmosphere at 1.14 ppm. Krypton -85 decays by β - emissions whose maximum beta energy is 0.672 Mev. The half life of ^{85}Kr is 10.76 years. The two best methods of detecting ^{85}Kr in the field are use of flow through proportional counters or scintillation crystals in conjunction with a multichannel analyzer.

The EPA Eastern Environmental Lab recommends separating Krypton from gas samples with charcoal and molecular sieve cold traps, calcium sulfate, ascarite (NaOH preparation) and a titanium furnace (900°C).³³ They then dissolve the sample in a liquid scintillator and count it with a photomultiplier system. In this process, Kr-83m is used as a tracer in the Krypton separation process. Krypton recoveries are reported in the 80 to 90% range with about 1m^3 of air processed in four hours.

7.1.3 Detection of ^{106}Ru

Ru in the form of RuO_4 would be expected to leak from a failed storage canister. Because of the low boiling point of RuO_4 , 108°C (226°F), it is assumed that most of the ^{106}Ru in a waste canister could be released as a gas if a leak path existed through the canister and salt. Ruthenium -106 undergoes a beta decay emitting a 0.039 MeV beta. Its daughter ^{106}Rh is associated with a very high energy beta particle (3.53 MeV) which would suggest the use of a single channel analyzer (3.4 to 3.6 MeV) to determine the presence and activity of ^{106}Ru against other betas.

7.1.4 Detection of Iodine-129

Iodine-129 is a beta emitter ($E_{\max} = 150 \text{ KeV}$) which is followed by a 40 KeV gamma and has a half life of 1.7×10^6 years. For low level quantities of I-129, two measurement techniques are employed: liquid scintillation counting and neutron activation analysis. The former is useful when the activity to be measured is known to exceed a few pCi of I-129, while the latter is recommended for the ultimate in resolution.

In either case, the I-129 must be chemically isolated from the sample matrix in which it is found. For this purpose, a specially prepared filter impregnated with barium hydroxide - iodide - iodate is recommended. After sample collection, the I-129 is concentrated chemically by dissolution and exchange between the radioiodine and the stable iodine.

7.1.5 Detection of Cesium

Radioactive cesium as Cs-134, Cs-135 or Cs-137 is isolated according to ASTM¹⁴ procedures by extracting it at a pH of 7.0 in a water solution as cesium tetra phenyl borate in amyl acetate with ethylenediamine tetra acetic acid, disodium salt (EDTA) present to prevent the extraction of undesired fission product activity.

A known fraction of the organic phase is then taken and gamma-counted in a standard vial. The cesium tetraphenyl borate solution is gamma counted and a gamma ray spectrum prepared using a multichannel pulse height analyzer. This spectrum is resolved into its components, by graphical or algebraic methods, and the count rate due to each isotope is obtained.

7.2 Local Detectors

For the detection of migrating sources of radionuclides in the repository, fixed radiation detectors are being evaluated. The radiation sensors most appropriate for this application include geiger muller tubes, proportional counters, and scintillation crystals.

Due to the high concentrations and roughly thirty year half lives of Cs-137 and Sr-90, these two radionuclides might be selected for failed canister identification. Ba-137m, the daughter product of Cs-137 decays with the emission of a 0.662 MeV gamma ray. Sr-90 decays to Y-90 with the emission of a 0.546 MeV beta particle. Subsequently, the Y-90 daughter emits a 2.27 MeV beta with a 64 hour half life. The gamma spectra for the spent fuel rods is primarily populated in the 0.6 to 0.8 MeV range.

7.2.1 Ionization Chambers

1 Ionization is the process of removing an electron from a neutron atom by imparting to the electron an energy greater than its binding energy. Ionization chambers are enclosed volumes of gas in which the ionization caused by irradiation can be collected and measured. Electrostatic fields in the sensitive volume force the free electrons and positively charged atom to opposite electrodes and the total charge or current is then measured by electrometers.

Ionization chambers are most frequently used as either pocket personnel dosimeters or survey meters. The "cutie pie" survey meter is a free air ionization chamber which can determine the dose due to alpha particles of greater than 3.5 MeV, betas greater than 40 KeV and gammas greater than 7 KeV, with sensitivities nominally ± 10 percent of the full scale reading.³⁷ An operational problem associated with the use of ionization chambers is that they require frequent calibration due to the electronic drift of the electrometer.

7.2.2 Proportional Counters

As opposed to ionization chambers which measure the total charge or current produced, proportional counters record the number of pulses of current generated by the incident radiation. The amplitude of the pulse created is proportional to the energy of the incident radiation particle. The physical design of the proportional counter is similar to that of the ionization chamber, however, the applied voltage at the field electrodes is higher.

Proportional counters can be used to discriminate between different types of radiation particles. For a given radiation type they can be used to establish an energy distribution of the primary radiation. Proportional counters are typically 30 percent efficient in detecting the alpha particles from ^{239}Pu (5.15 MeV) and are 40 percent efficient in detecting the beta particles from $^{90}\text{Sr} - ^{90}\text{Y}$ (2.27 MeV) in a 4π geometry.³² Proportional counters can detect particles of lower ionization energy than can be measured with ionization chambers due to the higher applied voltage.

7.2.3 Geiger-Muller Tubes

The G-M counter has a higher operating voltage than proportional counters but is otherwise similar in design. Its operation differs from that of a proportional counter in that the amplitude of the pulse of ion pairs produced is independent of the incident energy of the particle. The effect of increased operating voltage increases the number of ion pairs created by each incident radiation particle and results in a large enough pulse to omit the requirement of electronic amplification. A disadvantage of the G-M tube is that in general it cannot be used to determine dose rates since every particle entering the tube produces a pulse of equal height irrespective of the particle type (α, β, γ or n) or its energy.

Due to the higher applied voltage, G-M tubes are capable of detecting particles of lower ionization energy than can be measured by ionization chambers or proportional counters. The utility of a G-M tube is due to its high resolution, versatility for use with different types of radiation, and reasonable cost.

7.2.4 Scintillation Counters

Scintillation materials are phosphors which produce fluorescent radiation, or light, when ionizing particles pass through them. The advantages of scintillation counters over gas filled counters are their higher efficiencies due to their greater mass density, and their ability, when used in conjunction with a photomultiplier tube and pulse height analyzer, to determine the energy spectra of the incident radiation.

Depending upon the materials of construction and the geometrical design, scintillation detectors can be used to detect either alphas, betas, gammas or neutrons.

7.2.5 Semiconductor Detectors

Solid state semiconductors have very high energy resolution capabilities but rather poor efficiencies. Their low efficiency is due to their very small size (typically a few millimeters on a side). Their high energy resolution is due to the very low ionization energies required to create an ion pair (typically 3 ev

as compared with 30 ev for gas filled detectors and 300 ev for scintillation crystals).³²

There are five basic types of solid state detectors: homogeneous (CdS), diffused junction (Fe or Si), surface barrier junction, lithium drifted, and totally depleted. Of these, the surface barrier is predominately used for alpha and beta spectrometry, and the diffused junction and lithium drifted for gamma spectrometry.

7.2.6 Emulsion Detectors

Photographic film is in wide use for monitoring personnel for radiation doses due to x-ray, gamma and beta radiation. The response of the film varies with the x and gamma ray energies, however, by the suitable selection of absorbers the film dosimeter can be made essentially energy independent. Thermal neutron dosage may be accomplished through the comparisons of film exposures obtained under cadmium and brass filters.

7.2.7 Chemical Dosimeters

The chemical reaction which is in the most wide scale use for dosimetry is the oxidation of ferrous ions to ferric ions in 0.8 normal solutions of sulfuric acid.³² Alternately a solution of ceric sulfate in 0.8N sulfuric acid when subjected to radiation will be reduced to cerous ions. In either case, the product ion concentration is determined spectrophotometrically using infrared absorption techniques.

7.3.0 Radioactive Particulate Surveillance Methods

Radioactive particulate sampling refers to the collection of air particulates by passing air through a filter. The addition of an on-line radiation detector, for examining particulate buildup on the filter, gives an air particulate sampler a radiation surveillance capability.

7.3.1 Low Volume Air Particulate Samplers

The air sampler consists of a vacuum pump powered by an electric motor which draws air through a filter at the rate of about 5 to 10 CFM. A shielded radiation detector, mounted near the filter, would provide on-line surveillance of the particulate buildup for the presence of radioactivity. This surveillance method would permit gross counting of gamma, beta and alpha radiation depending on the detection scheme employed.

Air filters could be changed manually on a periodic basis or automatically based upon a pressure drop (DP) across the filter. A simplified

design of an automatic filter radioactive particulate detection device is suggested in DWG N. 1016-3-4. As shown in the figure, the filter supply spool could automatically feed clean filter media to the filter holder on either a preset time interval basis or on command from a central control computer. Used filter media would be stored on the take-up spool where it could be counted for gross radioactivity due to gamma decay. Shielding the take-up spool and radiation detector from the background radiation would serve to reduce counting time as well as improve counting statistics.

7.4 Resolution, Reliability, and Uncertainty

Due to the extremely good counting statistics available with nuclear particle detectors in radiation fields, very good resolution is obtainable in spent fuel dissolution monitoring systems. The detection of specific radionuclides in gaseous air samples and the detection of gross levels of beta-gamma fission product activity in a region adjacent to a local detector may be accomplished.

The determination of the monitoring system reliability and uncertainty is dependent on the establishment of the balance of the data acquisition, processing and analysis system.

ION CHAMBER

Fig. 7.1

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IODINE/PARTICULATE SAMPLER ASSEMBLY

Fig. 7.2

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MOVING FILTER PARTICULATE SAMPLER

Fig. 7.3

GEIGER-MULLER TUBE

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Fig. 7.4

THERMOLUMINESCENT DOSIMETER

Fig. 7.5

AUTOMATIC RADIOACTIVE PARTICULATE DETECTOR

Fig. 7.6

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DRAW. NO. 1016-3-4

BY JOHN FORGRAVE

8.0 Temperature Monitoring Instrumentation

Temperature monitoring in the salt mine repository is important because of the impact of temperature on canister corrosion rates, brine migration rates, creep rates of the plastically flowing salt, and spent fuel dissolution chemistry.

The maximum temperature to be encountered for three year old nuclear waste is calculated to be 127°C (260°F). The minimum temperature, the temperature of the salt before any waste is put in the mine, is 23°C (74°F).

There are three existing electrical temperature sensing devices that are suitable for use in salt at temperatures ranging from 32°C to 127°C . They are thermocouples, thermistors, and resistance temperature detectors (RTD).

8.1 Thermocouples

A thermocouple consists of two dissimilar metallic conductors in contact at one end to form the "hot" or measuring junction, the other two ends constitute the "cold" or reference junction. These two dissimilar conductors produce an electrical current whose magnitude depends on the temperature difference between the measuring junction and the reference junction.

8.1.1 Noble Metal Thermocouples

The noble metal thermocouples made out of platinum and iridium or rhodium as companion alloy, have excellent mechanical and chemical properties. Their operational temperature range goes from 0°C to 1500°C (32°F to 2732°F).¹⁹

Type S: Platinum vs. Platinum 90%-rhodium 10% thermocouple has a working temperature range of 0° to 1550°C with a corresponding EMF of 0 to 16 millivolts for an ice bath reference junction.¹⁹

Type R: Platinum vs. Platinum 87%-rhodium 13% is primarily used in industrial applications. For a working temperature range of 0° to 1600°C , the corresponding EMF is 0 to 18.6 millivolts with an ice bath reference junction.¹⁹

The type S and R thermocouples have a high resistance to oxidation and corrosion. However, they can be contaminated by hydrogen,

carbon and other metal vapors. Noble metal thermocouples are primarily utilized at higher temperatures (i.e. 400 to 1500°C) due to their high temperature chemical stability. Also, the cost of noble metal thermocouples is relatively high. These last two facts indicate that noble metal thermocouples are not ideally suited for use in salt mine repositories.

8.1.2 Base Metal Thermocouples¹³

The base metal thermocouples have a lower operating temperature range, and are less expensive than the noble metal thermocouples. Thus, widespread utilization of the base metal thermocouples has taken place industrially.

IRON-CONSTANTAN (TYPE J)

This base metal thermocouple is not feasible for use in salt due to high oxidization. It is, however, a very low cost thermocouple.

COPPER-CONSTANTAN (TYPE T)

Copper-constantan is the most homogenous and uniform base metal. Its temperature range is from 25°C to 350°C (-19°F to 660°F). This thermocouple oxidizes rapidly and, therefore, is not suited for use in salt.

CHROMEL-ALUMEL (TYPE K)

Chromel is made of 90% nickel and 10% chromium. Alumel consists of 93% nickel with aluminum, silicon, and manganese. Its

resistance to oxidation is better than any other base metal thermocouple. Its temperature range extends from -210°C to 1325°C (-320°F to 2400°F). Type K is a good thermocouple to be used in salt mines due to its low oxidation rate and low cost.

CHROMEL-CONSTANTAN (TYPE E)

This thermocouple has characteristics similar to the Type K thermocouple, but higher EMF. Type E is also a good thermocouple to be used in the salt mine temperature monitoring.

8.1.3 Protective Materials

In order to protect the thermocouples from mechanical and chemical reactions, the thermocouples can be covered with glass, ceramics, or plastics which can shield them from shocks and corrosion.

8.2 Thermistors

A thermistor is an electrical resistor whose resistance is a known linear function of temperature (fig. 8.1). The temperature range in which they are used varies from -100°C to 300°C (-148°F to 572°F) and the resistance may change as much as 10 million ohms.¹⁹ Because of its high sensitivity, the thermistor can be regarded as a very effective temperature measuring instrument. It also has the advantage of requiring no calibration or expensive matching wire as is needed in the case of thermocouples.

The thermistor, in order to indicate any temperature change, must be integrated as one element of a bridge circuit with an indicating galvanometer (fig. 8.2).

8.3 Resistance Thermometer

Resistance thermometers (fig. 8.3) are the most expensive and most accurate temperature reading devices. They consist essentially of platinum wire and they indicate a change in electrical resistance as a function of temperature. For platinum, the change in electrical resistance with temperature is a linear function. No special wiring is needed from the resistor to the ohm meter and no reference junction is required. The RTD operates at temperatures between -270°C and 660°C (-454°F to 1220°F).¹⁹

8.4 Resolution, Reliability, Uncertainty

Temperature monitoring of the spent fuel repository salt mine can be accomplished using conventional instrumentation systems. The sensitivity obtainable with either thermocouples, thermistors or resistance temperature detectors allows for less than 1°C (1.8°F) differential measurements.¹⁹

The reliability associated with the temperature monitoring system will largely depend on the corrosion resistance of the sensors. The uncertainty associated with detecting temperature anomalies is dependent on the balance of the data acquisition, processing, and interpretation system.

TYPICAL RESISTANCE-TEMPERATURE OF
THERMISTOR MATERIAL COMPARED TO
THAT OF PLATINUM.

Fig. 8.1

TYPICAL THERMISTOR TEMPERATURE
INDICATION CIRCUIT

Fig. 8.2

RESISTANCE THERMOMETER

ENGINEERING ANALYSIS AND TEST CO INC.

DRAW NO 2016-3-10

BY JOHN FORGRAVE

Fig. 8.3

9.0 Health Physics Radiation Monitoring

The present licensing and monitoring policy for fuel reprocessors and other nuclear facilities is to keep effluent concentrations as low as practicable, and well below the maximum permissible concentrations published in 10 CFR 20.

The monitoring requirements of 10 CFR 20 are deemed mandatory if a radiation worker is likely to receive a dose in any calendar quarter in excess of

- . 25 Percent of the applicable value listed in Table 9-1.
- . 5 Percent of the applicable value listed in Table 9-1 for individuals under 18 years of age.

The mandatory monitoring procedures include personnel monitoring equipment for each individual to establish external radiation dose, and periodic (monthly, quarterly, annually, etc.) urinalysis and/or evaluation of air concentrations to which the individual is exposed to establish internal radiation dose. Allowable radionuclide concentrations in air and water are established in 10 CFR 20, Appendix B, for controlled and uncontrolled areas

and are reproduced in Table 9-2 for the major fission products in PWR-U spent fuel waste. In addition to specifying the maximum permissible concentrations in air and water of the major radioisotopes of each element, 10 CFR 20 states that if either the identity or the concentration of any radionuclide in the mixture is not known, the limiting values shall be those listed in Table 9-2 for mixtures. Generally, the concentrations of unknown mixtures of radionuclides in air are limited to 10^{-4} to 10^{-5} of the concentrations allowed for individual isotopes in air. This would indicate the desirability of spectrally analyzing air samples at the spent fuel repository when gross activities in air approach 6×10^{-13} $\mu\text{Ci/ml}$.

For the vault storage area monitoring, the specification of primary interest is that for a controlled area in air as indicated in Table 9-2. Due to the requirement for the detection of such small amounts of radioactivity, periodic laboratory analysis of grab samples, wipes and filters left in place for predetermined periods will be necessary. Where the gross activity in air exceeds 6×10^{-13}

μCi/ml, chemical separation followed by spectroscopic analysis isotope identification will be necessary to meet the requirements of 10 CFR 20.

The major fission products expected in PWR-U spent fuel wastes, together with their maximum permissible concentrations (MPC), are listed in Table 9-2. The seven major isotopes, and decay products, listed in Table 9-2 account for 99.5% of the total nuclear activity in 10-year-old PWR-U waste. The isotopes, ^{90}Sr and ^{137}Cs , and their decay products, ^{90}Y and $^{137\text{m}}\text{Ba}$, account for 93% of the total nuclear activity in 10-year-old PWR-U waste and 98% of the total nuclear activity in 100-year-old PWR-U waste. This would indicate the need for specialized monitoring techniques for determining the concentrations of ^{90}Sr and ^{137}Cs at various locations throughout a vault storage area. As stated in 10 CFR 20, Appendix B, "if the identity and concentration of each radionuclide in the mixture are known, the limiting values should be derived as follows:"

$$\frac{C_A}{\text{MPC}_A} + \frac{C_B}{\text{MPC}_B} + \frac{C_C}{\text{MPC}_C} + \dots = 1,$$

where C_A , C_B , C_C, are the concentrations of radionuclides A, B, C, and MPC_A , MPC_B , MPC_C , are the applicable maximum permissible concentrations from 10 CFR 20, Appendix B.

The limitations imposed on radionuclide mixtures in the above equation serve to further reduce the MPC's of the individual isotopes listed in Table 9-2. A first approximation indicates that for a completely dispersed, homogeneous, mixture of PWR-U waste the MPC's for the two major constituents, ^{90}Sr and ^{137}Cs , would be 2×10^{-10} $\mu\text{Ci/ml}$ for ^{90}Sr and 3×10^{-9} $\mu\text{Ci/ml}$ for ^{137}Cs .

The present policy on the licensing of nuclear facilities is to require calculated values of the concentration of radioactive emissions for preoperational licensing. Due to the individual idiosyncrasies of each facility because of siting, meteorological and operational differences, a given facility normally will be limited by individualized technical specifications written into the license to ensure that effluent concentrations will be below 10 CFR 20 values.

Among other licensing requirements is the study of engineering features for prevention and mitigation of the consequences of various postulated accidents; and a study of the site for possible long-term contamination which would hinder ultimate decommissioning of the facility and the return of the area to other uses. It is noted that the maximum permissible concentrations (MPC) of the radiation concentration guide (RCG) fall into essentially two categories; those for beta-gamma emitters (10^{-5} to 10^{-9} $\mu\text{Ci/ml}$), and those for alpha emitters (10^{-10} to 10^{-12} $\mu\text{Ci/ml}$). The reason for this distinction is that the precursors of alpha particles are predominantly the heavy elements (atomic mass greater than lead) which when ingested or inhaled, ultimately become concentrated in the bone marrow.

TABLE 9-1

RADIATION PROTECTION STANDARDS FOR
EXTERNAL AND INTERNAL EXPOSURE¹

<u>TYPE OF EXPOSURE</u>	<u>CONDITION</u>	<u>DOSE OR DOSE COMMITMENT (REM)</u>
Whole body; head and trunk; active blood forming organs; lens of eyes; or gonads	Calendar quarter	1.25 ²
Skin of whole body	Calendar quarter	7.50
Hands and fore- arms	Calendar quarter	18.75

Notes:

1. Based upon Code of Federal Regulations, Title 10
Part 20, "Standards For Protection Against Radiation,"
January 1, 1974.
2. This may be exceeded provided:
 - a) the whole body dose is less than 3 rems
per calendar quarter, and
 - b) the accumulated occupational dose to the
whole body is less than 5 (N-18) where N
is the age in years at last birthday, and
 - c) a proper determination of the whole body
dose has been made.

TABLE 9-2

FISSION PRODUCT CONCENTRATIONS ABOVE
NATURAL BACKGROUND IN AIR AND WATER¹

ISOTOPE ²	CONTROLLED AREAS		UNCONTROLLED AREAS	
	Air ($\mu\text{Ci/ml}$)	Water ($\mu\text{Ci/ml}$)	Air ($\mu\text{Ci/ml}$)	Water ($\mu\text{Ci/ml}$)
Mixtures	6 E-13	4 E-7	2 E-14	3 E-8
⁹⁰ Sr	1 E-9	1 E-5	3 E-11	3 E-7
(⁹⁰ Y)	1 E-7	6 E-4	4 E-9	2 E-5
¹⁰⁶ Ru	8 E-8	4 E-4	3 E-9	1 E-5
(¹⁰⁶ Rh)	--	--	--	--
¹³⁴ Cs	4 E-8	3 E-4	1 E-9	9 E-5
¹³⁷ Cs	6 E-8	4 E-4	2 E-9	2 E-5
(^{137m} Ba)	--	--	--	--
¹⁴⁷ Pm	6 E-8	6 E-3	2 E-9	2 E-4
¹⁵¹ Sm	6 E-8	1 E-2	2 E-9	4 E-4
¹⁵⁴ Eu	4 E-9	6 E-4	1 E-10	2 E-5

Notes:

1. Concentrations are based on values from 10 CFR 20, Appendix B, for soluble radionuclides.
2. Decay products are shown in parentheses.
3. These concentrations apply if either the identity or concentration of any radionuclide in a mixture is not known.
4. Reads $6 \times 10^{-13} \mu\text{Ci/ml}$.

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APPENDIX A - STUDY BASIS

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A.0 Study Basis

1.0 Introduction

The monitoring of the spent fuel storage canisters envisioned for use in this study is of primary importance to the success of the program. This monitoring includes the accurate determination of canister position at the storage site for periods of up to 25 years after storage to facilitate the retrieval of these canisters if the need arises. Determination of canister integrity is necessary so that repairs or replacement may be effected. Standards of acceptance set down in 10 CFR 20 with regards to the maximum permissible concentrations (MPC) of radionuclides in the air, water, and soil must be met.²² For these purposes, accurate data is needed concerning the amount and type of actinides, fission products, and gaseous effluents expected in the canisters during normal operation. Once these amounts are known the choice of monitoring and detection equipment may be made based on the

sensitivity and accuracy of the equipment.
Methods governing the placement and use of
this equipment may also impact the instru-
mentation decision.

2.0 Waste Characteristics

The waste products computed for this study were assumed to be generated only by LWR's operating in the United States. No other reactor types (ie LMFBF, HTGR) were considered. The fuel used was assumed to be enriched uranium fuel for a typical 1000 MWe PWR. Waste from BWR's was assumed to be very close in form and total curie content to that from a PWR. For example, the Office of Waste Isolation computed a waste source strength of 4.06×10^5 Ci/MTHM for PWR spent fuel and 3.38×10^5 Ci/MTHM for BWR spent fuel.³¹ The differences come mainly from the actinide contents of the respective fuels. A PWR has more Uranium, Plutonium, Americium and Cesium at discharge than a BWR due to the variation of the initial isotopic content of the fuel at reactor loading with respect to isotopic distribution of the activation products of uranium. Since the PWR fuel contains more radioactivity at discharge this fuel was chosen for consideration in order to produce a conservative analysis.

The fuel was assumed to have undergone a 30,000 MWD burn up at a power of 30.00 MW/MT. The scenario envisioned, was that this fuel was then removed from the reactor and placed, as is, in a 1 ft. diameter by 16 ft. long low carbon steel canister with a volume of approximately 10 ft.³ The waste from 5 metric tons of heavy metal (MTHM) was assumed to just fill this canister. The canisters would then be shipped to a surface storage repository for 10 years of above ground storage, after which they would be buried in the underground vaults. Retrieval periods of up to 25 years after underground burial (35 years since discharge from the reactor) are considered.

Table 2.1 shows the assumed characteristics of the PWR used in this study, and Table 2.2 gives the radiation characteristics of 1 MTHM irradiated in this reactor for the assumed period. For the reference U-235 fueled PWR fuels (PWR-U), the ORSAC³⁰ and ORIGEN computer codes were used to calculate the radioactivity per MTHM as a

function of time in the form of light element isotopes induced in the cladding, fission product isotopes in the spent fuel, and actinide isotopes. These results are given in later sections of this report.

Table 2.1

ASSUMED CHARACTERISTICS OF PWR
FUELED WITH ENRICHED URANIUM^{2,3}

Electric Power, MW(e) (net)	1,000
Thermal Power, MW	3,077
Avg. Specific Power, MW/Metric Ton ⁽¹⁾	37.5
Avg. Burnup, MWD/Metric Ton	32,873
Refueling Interval, days ⁽²⁾	365.25

Steady-State Charge, Kg.

U-233	0
U-235	875.2
Total U	27,350
Fissile Pu ⁽³⁾	0
Total Pu ⁽⁴⁾	0
U + Pu	27,350

Steady-State Discharge, Kg.

U-233	0
U-235	243.4
Total U	26,137
Fissile Pu ⁽³⁾	180.1
Total Pu ⁽⁴⁾	254.9
U + Pu	26,572

Table 2.1
(continued)

NOTES:

- (1) Based upon full power and fuel charged.
- (2) At 80% load factor.
- (3) Pu-239 + Pu-241
- (4) Pu-238 + Pu-239 + Pu-240 + Pu-241 + Pu-242

Table 2.2

RADIATION CHARACTERISTICS OF ONE METRIC TON OF
ENRICHED URANIUM FUEL FROM A PWR (PWR-U) ⁽¹⁾
BASIS = 1 MTHM CHANGED TO REACTOR

Time Since Discharge(days)	Weight (Kg)	Power (W)	Hazard (M ³ at RCG ⁽²⁾)	
			Air	Water
0	2.71 x 10 ²	7.75 x 10 ²	6.67 x 10 ¹³	1.29 x 10 ⁹
90	2.71 x 10 ²	3.14 x 10 ²	4.05 x 10 ¹³	6.46 x 10 ⁸
160	2.71 x 10 ²	2.03 x 10 ²	3.03 x 10 ¹³	4.35 x 10 ⁸
365	2.71 x 10 ²	1.04 x 10 ²	2.04 x 10 ¹³	2.36 x 10 ⁸
1,095	2.71 x 10 ²	6.93 x 10 ¹	1.48 x 10 ¹³	1.65 x 10 ⁸
3,650	2.71 x 10 ²	2.72 x 10 ¹	6.02 x 10 ¹²	7.52 x 10 ⁷

FISSION PRODUCTS

0	3.49 x 10 ¹	1.52 x 10 ⁶	3.67 x 10 ¹⁶	6.40 x 10 ¹²
90	3.49 x 10 ¹	2.71 x 10 ⁴	1.23 x 10 ¹⁶	5.37 x 10 ¹¹
160	3.49 x 10 ¹	1.92 x 10 ⁴	1.01 x 10 ¹⁶	4.49 x 10 ¹¹
365	3.49 x 10 ¹	1.05 x 10 ⁴	7.05 x 10 ¹⁵	3.58 x 10 ¹¹
1,095	3.49 x 10 ¹	3.50 x 10 ³	3.67 x 10 ¹⁵	2.74 x 10 ¹¹
3,650	3.49 x 10 ¹	1.04 x 10 ³	2.29 x 10 ¹⁵	2.11 x 10 ¹¹

ACTINIDES

0	9.65 x 10 ²	7.46 x 10 ⁴	1.05 x 10 ¹⁷	2.16 x 10 ¹¹
90	9.65 x 10 ²	1.06 x 10 ³	1.03 x 10 ¹⁷	2.77 x 10 ⁹
160	9.65 x 10 ²	8.40 x 10 ²	1.01 x 10 ¹⁷	2.48 x 10 ⁹

Table 2.2
(continued)

365	9.65×10^2	4.78×10^2	9.87×10^{16}	2.00×10^9
1,095	9.65×10^2	2.31×10^2	9.47×10^{16}	1.66×10^9
3,650	9.65×10^2	2.24×10^2	8.65×10^{16}	1.63×10^9

(1) See Table 2.1 for initial composition of fuel and motor operating characteristics.

(2) Radiation concentration guides correspond to Appendix B, Table II, 10CFR 20.

3.0 Cladding

Table 3.1 indicates the radioactivity of light element isotopes expected to be present in PWR-U spent fuel cladding per MTHM. Table 3.2 gives the inventory per canister. The major contributors will be CO-60, Fe-55, Ni-59, and Ni-63.

The cladding was assumed to be Zircalory-4 with Inconel spacers. Up to 0.1% of the actinides and fission products, and several percent of the fission product tritium, may be associated with the cladding wastes.

Table 3.1

CALCULATED RADIOACTIVITY OF LIGHT-ELEMENT ISOTOPES
PRESENT IN SPENT PWR-U FUEL CLADDING

POWER = 30.00MW, BURN UP = 33,000MWD, FLUX = 2.92×10^{13} N/cm²- sec

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 MTHM CHARGED TO REACTOR

Isotope	Half-life ⁽¹⁾	Change	Discharge ⁽²⁾ 1 Y		10 Y	35 Y
Ca 45	163 D	0.0	3.17 - 2	6.83 - 3	6.87 - 9	7.83 - 26
SC 46	83.8 D	0.0	4.26 + 0	2.09 - 1	3.35 - 10	0.0
Cr 51	27.71 D	0.0	2.85 + 4	3.29 + 0	0.0	0.0
Mn 54	312.5 D	0.0	2.49 + 2	1.08 + 2	5.87 - 2	1.21 - 10
Fe 55	2.7 Y	0.0	2.00 + 3	1.53 + 3	1.39 + 2	2.51 - 1
Fe 59	44.6 D	0.0	2.28 + 2	8.24 - 1	8.39 - 23	0.0
Co 58	71.3 D	0.0	9.32 + 3	2.68 + 2	3.53 - 12	0.0
Co 60	5.272 Y	0.0	6.44 + 3	5.64 + 3	1.72 + 3	6.47 + 1
Ni 59	8 + 4 Y	0.0	3.85 + 0	3.85 + 0	3.85 + 0	3.85 + 0
Ni 63	1 + 2 Y	0.0	5.65 + 2	5.61 + 2	5.24 + 2	4.43 + 2
Sr 89	50.5 D	0.0	3.81 + 1	2.94 - 1	2.73 - 20	0.0
Y 91	58.6 D	0.0	1.01 + 2	1.37 + 0	2.02 - 17	0.0
Zr 93	9.5 + 5 Y	0.0	1.03 - 1	1.03 - 1	1.03 - 1	1.03 - 1
Zr 95	65.5 D	0.0	2.89 + 4	5.90 + 2	3.51 - 13	0.0
Nb 93M	12 Y	0.0	8.32 - 3	1.39 - 2	5.31 - 2	1.10 - 3
Nb 95	35.1 D	0.0	2.80 + 4	1.23 + 3	2.46 - 13	0.0
Mo 93	3 + 3 Y	0.0	2.00 - 2	2.00 - 2	2.00 - 2	1.98 - 2
Tc 99	2.13 + 5 Y	0.0	1.40 - 2	1.40 - 2	1.40 - 2	1.40 - 2

Table 3.1
(continued)

Sn 117M	14 D	0.0	1.21 + 4	1.72 - 4	0.0	0.0
Sn 119M	245 D	0.0	2.41 + 1	8.74 + 0	9.62 - 4	4.74 - 15
Sn 121M	50 Y	0.0	3.37 - 1	3.34 - 1	3.08 - 1	2.07 - 1
Sn 123	129 D	0.0	4.47 - 1	5.90 - 2	7.14 - 10	0.0
Sb 124	60.2 D	0.0	4.73 + 0	6.98 - 2	2.24 - 18	0.0
Sb 125	2.73 D	0.0	4.47 + 1	3.46 + 1	3.43 + 0	6.19 - 3
Te 125M	58 D	0.0	1.62 + 1	1.43 + 1	1.42 + 0	0.0
Total			1.18 + 5	1.00 + 4	2.40 + 3	5.12 + 2

(1) Year = Y

Day = D

Hour = H

Second = S

(2) Read as 3.17×10^{-2}

Table 3.2

CALCULATED RADIOACTIVITY OF LIGHT-ELEMENT ISOTOPES
PRESENT IN SPENT PWR-U FUEL CLADDING (PER CANISTER)

POWER = 30.00 MW, BURN-UP = 33,000 MWD, FLUX = 2.92×10^{13} N/cm² -sec

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 CANISTER (5MTHM)

Isotope	Discharge ⁽¹⁾	10 Y	35 Y
Ca 45	1.58 - 1	3.44 - 8	3.92 - 25
Sc 46	2.13 + 1	1.68 - 9	0.0
Cr 51	1.48 + 5	0.0	0.0
Mn 54	1.25 + 3	2.94 - 1	6.05 - 10
Fe 55	1.00 + 4	6.95 + 2	1.26 + 0
Fe 59	1.04 + 3	4.20 - 22	0.0
Co 58	4.66 + 4	1.77 - 11	0.0
Co 60	7.34 + 6	8.60 + 3	3.24 + 2
Ni 59	1.93 + 1	1.93 + 1	1.93 + 1
Ni 63	2.83 + 3	2.62 + 3	2.22 + 3
Sr 89	1.91 + 2	1.37 - 19	0.0
Y 91	5.05 + 2	1.01 - 16	0.0
Zr 93	5.15 - 1	5.15 - 1	5.15 - 1
Zr 95	1.45 + 5	1.76 - 12	0.0
Nb 92M	4.16 - 2	2.66 - 1	5.50 - 3
Nb 95	1.40 + 5	3.73 - 12	0.0
Mo 93	1.00 - 1	1.00 - 1	9.90 - 2
Tc 99	7.00 - 2	7.00 - 2	7.00 - 2
Sn 117M	6.05 + 4	0.0	0.0

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Table 3.2

(continued)

Sn 119M	1.21 + 2	4.81 - 3	2.37 - 14
Sn 121M	1.69 + 0	1.54 + 0	1.04 + 0
Sn 123	2.24 + 0	3.57 - 9	0.0
Sb 124	2.37 + 1	1.12 - 17	0.0
Sb 125	2.24 + 2	1.72 + 1	3.10 - 2
Te 125M	8.10 + 1	7.10 + 0	0.0
Total	5.90 + 5	1.20 + 4	2.56 + 3

(1) Read as 1.58×10^{-1}

4.0 Actinide Isotopes

The actinide isotope inventory is given in Table 4.1 for 1 MTHM. Table 4.2 gives the results for one typical storage canister.

As seen in Table 4.1, one MTHM consists of a charge of 2.03 curies of actinide isotopes, increasing to 1.96×10^7 Ci at discharge. Because of the complex decay schemes among the actinide it can be seen that some of the isotopes increase in curie content with time. An example of these decay schemes is the production of Pu-238 via the following chain:

Another example is the burn-up of Pu-239 to produce Cm 244:

From Table 4.1 it is seen that the major isotopes, as far as curies contributed, are Pu-238, Pu-240, Pu-241, and Am-241. Plutonium, because of its long half-life, will be present in substantial quantities for extremely long times.

Table 4.1

**CALCULATED RADIOACTIVITY OF ACTINIDE ISOTOPES
PRESENT IN SPENT PWR-U FUEL**

POWER = 30.00 MW, BURN-UP = 33,000 MWD, FLUX = 2.92×10^{13} N/cm²-sec

NUCLIDE RADIOACTIVITY, CURIES
BASIS = 1 MTHM CHARGED TO REACTOR

Isotope	Half-Life ⁽¹⁾	Charge	Discharge ⁽²⁾	1 Y	10 Y	35 Y
Pb 212	10.64H	0.0	1.50 - 3	3.40 - 3	1.60 - 2	5.0 - 2
Bi 212	60.6 M	0.0	1.50 - 3	3.40 - 3	1.60 - 2	5.0 - 2
Po 212	45 S	0.0	9.59 - 4	2.17 - 3	1.02 - 2	5.0 - 2
Po 216	15 S	0.0	1.50 - 3	3.40 - 3	1.60 - 2	5.0 - 2
Ra 220	55.6 S	0.0	1.50 - 3	3.40 - 3	1.60 - 2	5.0 - 2
Ra 224	3.64 D	0.0	1.50 - 3	3.40 - 3	1.60 - 2	5.0 - 2
Th 228	6.13 H	0.0	1.49 - 3	3.38 - 3	1.60 - 2	5.0 - 2
Th 231	25.52 H	0.0	7.93 - 1	1.71 - 2	1.71 - 2	1.71 - 2
Th 234	24.1 D	0.0	3.14 - 1	3.14 - 1	3.14 - 1	3.14 - 1
Pa 233	2.70 D	0.0	3.24 - 1	3.40 - 1	3.43 - 1	3.45 - 1
Pa 234M	6.67 H	0.0	3.21 - 1	3.14 - 1	3.14 - 1	3.14 - 1
U 232	72 Y	0.0	6.07 - 3	9.01 - 3	1.74 - 2	1.71 - 2
U 234	2.44 + 5 Y	1.64 + 0	7.51 - 1	7.59 - 1	8.30 - 1	1.83 - 0
U 235	7.04 + 8 Y	7.07 - 2	1.71 - 2	1.71 - 2	1.71 - 2	1.71 - 2
U 236	2.34 + 7 Y	0.0	2.88 - 1	2.88 - 1	2.88 - 1	2.88 - 1
U 237	6.75 D	0.0	8.65 + 5	2.41 + 0	1.57 + 0	4.77 - 1
U 238	4.47 + 9 Y	3.22 - 1	3.14 - 1	3.14 - 1	3.14 - 1	3.14 - 1

Table 4.1
(continued)

Np 237	2.14 + 6 Y	0.0	3.33 - 1	3.40 - 1	3.43 - 1	3.5 - 1
Np 239	2.35 D	0.0	1.85 + 7	1.82 + 1	1.82 + 1	1.82 + 1
Pu 236	2.85 Y	0.0	3.50 - 1	2.75 - 1	3.08 - 2	7.05 - 5
Pu 238	87.8 + 4 Y	0.0	2.72 + 3	2.86 + 3	2.70 + 3	2.30 + 3
Pu 239	2.43 + 4 Y	0.0	3.18 + 2	3.23 + 2	3.23 + 2	3.23 + 2
Pu 240	6.54 + 3 Y	0.0	4.77 + 2	4.77 + 2	4.79 + 2	4.79 + 2
Pu 241	15 Y	0.0	1.05 + 5	1.00 + 5	6.53 + 4	2.08 + 4
Pu 242	2.87 + 5 Y	0.0	1.38 + 0	1.38 + 0	1.38 + 0	1.38 + 0
Am 241	433 Y	0.0	8.59 + 1	2.50 + 2	1.41 + 3	1.54 + 3
Am 242 M	14.0mS	0.0	9.16 + 0	9.12 + 0	8.75 + 0	7.77 + 0
Am 242	16.02 H	0.0	6.34 + 4	9.12 + 0	8.75 + 0	7.77 + 0
Am 243	7.37 + 3 Y	0.0	1.81 + 1	1.82 + 1	1.82 + 1	1.82 + 1
Cm 242	163 D	0.0	3.34 + 4	7.12 + 3	7.18 + 0	8.25 - 20
Cm 243	28 Y	0.0	3.71 + 0	3.63 + 0	2.99 + 0	1.56 + 0
Cm 244	17.9 Y	0.0	2.44 + 3	2.35 + 3	1.67 + 3	6.29 + 2
Cm 245	8.5 + 3 Y	0.0	3.41 - 1	3.41 - 1	3.41 - 1	3.40 - 1
Cm 246	4.76 + 3 Y	0.0	6.84 - 2	6.84 - 2	6.83 - 2	6.81 - 2
Bk 249	311 D	0.0	3.58 - 3	1.60 - 3	1.13 - 6	1.52 - 15
Total		2.03 + 0	1.96 + 7	1.14 + 5	7.20 + 4	2.61 + 4

(1) Year = Y

Table 4.1
(continued)

Day = D

Hour = H

Second = S

Millisecond = mS

(2) Read as 1.50×10^{-3}

Table 4.2

CALCULATED RADIOACTIVITY OF ACTINIDE ISOTOPES
PRESENT IN SPENT PWR-U FUEL (PER CANISTER)

POWER = 30.00, BURN-UP = 33,000 MWD, FLUX = $2.92 \times 10^{13} \text{N/cm}^2\text{-sec}$

NUCLIDE RADIOACTIVITY CURIES
BASIS = 1 CANISTER (5 MTHM)

<u>ISOTOPE</u>	<u>DISCHARGE</u>	<u>10 Y</u>	<u>35 Y</u>
Pb 212	7.50 - 3 ⁽¹⁾	8.00 - 2	2.50 - 1
Bi 212	7.50 - 3	8.00 - 2	2.50 - 1
Po 212	4.80 - 3	5.10 - 2	2.50 - 1
Po 216	7.50 - 3	5.10 - 2	2.50 - 1
Po 216	7.50 - 3	8.00 - 2	2.50 - 1
Rn 220	7.50 - 3	8.00 - 2	2.50 - 1
Ra 224	7.50 - 3	8.00 - 2	2.50 - 1
Th 228	7.45 - 3	8.00 - 2	2.50 - 1
Th 231	3.97 + 0	8.55 - 2	8.55 - 2
Th 234	1.57 + 0	1.57 + 0	1.57 + 0
Pa 233	1.62 + 0	1.72 + 0	1.73 + 0
Pa 234 M	1.61 + 0	1.57 + 0	1.57 + 0
U 232	3.04 - 2	8.70 - 2	8.55 - 2
U 234	3.76 + 0	4.15 + 0	9.15 + 0
U 235	8.55 - 2	8.55 - 2	8.55 - 2
U 236	1.44 + 0	1.44 + 0	1.44 + 0
U 237	4.33 + 6	7.85 + 0	2.39 + 0
U 238	1.57 + 0	1.57 + 0	1.57 + 0
Np 237	1.67 + 0	1.72 + 0	1.75 + 0

Table 4.2
(continued)

Np 239	9.25 + 7	9.10 + 1	9.10 + 1
Pu 236	1.75 + 0	1.54 - 1	3.53 - 4
Pu 238	1.36 + 0	1.35 + 4	1.15 + 4
Pu 239	1.59 + 3	1.62 + 3	1.62 + 3
Pu 240	2.39 + 3	2.40 + 3	2.40 + 3
Pu 241	5.25 + 5	3.27 + 5	1.04 + 5
Pu 242	6.90 + 0	6.90 + 0	6.90 + 0
Am 241	4.30 + 2	7.05 + 3	7.70 + 3
Am 242 M	4.58 + 1	4.38 + 1	3.89 + 1
Am 242	3.17 + 5	4.38 + 1	3.89 + 1
Am 243	9.05 + 5	9.10 + 1	9.10 + 1
Cm 242	1.67 + 5	3.59 + 1	4.13 - 19
Cm 243	1.86 + 1	1.50 + 1	7.80 + 0
Cm 244	1.22 + 4	8.35 + 3	3.15 + 3
Cm 245	1.41 + 0	1.71 + 0	1.70 + 0
Cm 246	3.42 - 1	3.42 - 1	3.42 - 1
Bk 249	1.79 - 2	9.04 - 9	7.60 - 15
Total	9.80 + 7	3.60 + 5	1.31 + 5

(1) Read as 7.50×10^{-3}

5.0 Fission Products

Table 5.1 shows the calculated activity of fission product isotopes present in spent PWR-U fuel, per MTHM while Table 5.2 shows the same thing but for 1 canister.

Because of their gaseous nature, several of the fission product isotopes are of concern. These gases will be released from the canisters if integrity is lost, and are therefore important for detection and monitoring reasons. The gaseous effluents present among the fission product isotopes will be H-3 (Tritium), Kr-85, Ru-106, I-129, I-131, Xe-131M, Xe-133, Xe-135, Cs-134, Cs-135, and Cs-137.

5.1 Noble Gases

The noble gases will consist primarily of isotopes of Kr and Xe. Typically 5 to 6 Kg./MTHM are produced, but only Kr-85 is of concern after about 150 days. Therefore, the detection of Kr-85 will be a primary means for determination of canister leakage.

5.2 Tritium

Most tritium arises in LWR wastes principally from neutron reactions with light elements such as lithium or boron in the primary coolant. However, about 0.02 to 0.03 Ci of H-3/MWD (thermal) (700 to 1200 Ci/MTHM) will be produced in fission. A few percent may be associated with the Cladding but most will be present with the core material.

5.3 Iodine

Iodine is a semi-volatile fission product requiring attention because of its complex biological and chemical reactions. About 0.01 gram of iodine isotopes are formed per MWD (thermal) with I-131 of greatest concern during the operation of the reactor. However, because of its long half-life (1.6×10^7 years), I-129 becomes more important when dealing with long-term management of wastes. I-129 composes 75% by weight of the fission product iodine isotope.

5.4 Cesium

Cesium will be present in the form of cesium oxide, and as indicated in Tables 5.1 and 5.2 the cesium isotopes are among the "hottest" of the fission product isotopes. Therefore, their containment or detection is of high importance in canister design and monitoring.

Table 5.1

**CALCULATED RADIOACTIVITY OF FISSION-PRODUCT ISOTOPES
PRESENT IN SPENT PWR-U FUEL**

Power = 30.00 MU, Burn-up = 33,000 MWD, Flux = 2.92×10^{13} N/cm²- sec

Nuclide Radioactivity, Curies
Basis = 1 MTHM Charged to Reactor

<u>ISOTOPE</u>	<u>HALF-LIFE</u> ⁽¹⁾	<u>CHARGE</u>	<u>DISCHARGE</u>	<u>1Y</u>	<u>10Y</u>	<u>35Y</u>
H 3	12.33 Y	0.0	7.09 + 2 ⁽²⁾	6.70 + 2	4.03 + 2	9.92 + 1
Se 79	6.5 + 4 Y	0.0	3.98 - 1	3.98 - 1	3.98 - 1	3.98 - 1
Kr 85	10.73 Y	0.0	1.13 + 4	1.06 + 4	5.96 + 3	1.18 + 3
Rb 86	18.65 D	0.0	4.93 + 2	6.57 - 4	0.0	0.0
Sr 89	50.5 D	0.0	7.18 + 5	5.33 + 3	5.15 - 16	0.0
Sr 90	29 Y	0.0	7.76 + 4	7.57 + 4	6.07 + 4	3.36 + 4
Y 90	64 H	0.0	8.07 + 4	7.58 + 4	6.07 + 4	3.36 + 4
Y 91	5.86 D	0.0	9.38 + 5	1.28 + 4	1.89 - 13	0.0
Zr 93	9.5 + 5 Y	0.0	1.89 + 0	1.89 + 0	1.89 + 0	1.89 + 0
Nb 93M	12 Y	0.0	1.45 - 1	2.31 - 1	8.40 - 1	1.92 - 2
Zr 95	6.55 D	0.0	1.37 + 6	2.80 + 4	1.67 - 11	0.0
Nb 95M	3.61 D	0.0	2.80 + 4	5.94 + 2	3.54 - 13	0.0
Nb 95	3.51 D	0.0	1.38 + 6	5.95 + 4	3.61 - 11	0.0
Tc 99	2.13 + 5 Y	0.0	1.43 + 1	1.43 + 1	1.43 + 1	1.43 + 1
Ru 103	3.96 D	0.0	1.22 + 6	2.05 + 3	2.09 - 22	0.0
Rh 103M	56 M	0.0	1.22 + 6	2.05 + 3	2.09 - 22	0.0
Ru 106	396 D	0.0	5.45 + 5	2.73 + 5	5.50 + 2	2.03 - 5
Rh 106	29.9 S	0.0	7.40 + 5	2.73 + 5	5.50 + 2	2.03 - 5
Pd 107	6.5 + 6 Y	0.0	1.10 - 1	1.10 - 1	1.10 - 1	1.10 - 1

Table 5.1

(Continued)

Ag 110 M	252 D	0.0	3.68 + 3	1.35 + 3	1.66 - 1	1.98 - 12
Ag 110	252 D	0.0	1.59 + 5	1.76 + 2	2.16 - 2	2.68 - 13
Ag 111	7.47 D	0.0	3.96 + 4	8.89 - 11	0.0	0.0
Cd 113 M	14.6 Y	0.0	1.05 + 1	9.99 + 0	6.40 + 0	1.99 + 0
In 114 M	44mS	0.0	1.55 + 0	9.84 - 3	1.58 - 22	0.0
In 114	71.9 S	0.0	2.21 + 0	9.49 - 3	1.53 - 22	0.0
Cd 115 M	44.6 D	0.0	5.51 + 2	1.53 + 0	1.48 - 23	0.0
Sn 117 M	14 D	0.0	5.96 + 1	8.45 - 7	0	0.0
Sn 119 M	234 D	0.0	1.64 + 1	5.96 + 0	6.56 - 4	3.23 - 15
Sn 123	129 D	0.0	8.88 + 3	1.17 + 3	1.42 - 5	1.32 - 26
Te 123 M	119.7 D	0.0	6.10 - 1	7.02 - 2	2.44 - 10	0.0
Sb 124	60.2 D	0.0	4.06 + 2	5.99 + 0	1.92 - 16	0.0
Sn 125	9.65 D	0.0	1.35 + 4	2.77 - 8	0.0	0.0
Sb 125	2.73 Y	0.0	1.35 + 4	2.77 - 8	0.0	0.0
Te 125 M	58 D	0.0	3.11 + 3	2.83 + 3	2.81 + 2	4.59 - 1
Sn 126	1 + 5 Y	0.0	5.46 - 1	5.46 - 1	5.46 - 1	5.46 - 1
Sb 126 M	1.90 M	0.0	2.25 + 1	5.46 - 1	5.46 - 1	5.46 - 1
Sb 126	12.4 D	0.0	5.70 + 1	5.41 - 1	5.41 - 1	5.41 - 1
Te 127M	109 D	0.0	1.54 + 4	1.57 + 3	1.30 - 6	0.0
Te 127	9.4 H	0.0	7.20 + 4	1.55 + 3	1.29 - 6	0.0
Te 129 M	33.4 D	0.0	5.73 + 4	3.38 + 1	0.0	0.0
Te 129	70 M	0.0	3.37 + 5	2.17 + 1	0.0	0.0

Table 5.1
(continued)

I 129	1.59 + 7 Y	0.0	3.71 - 2	3.74 - 2	3.74 - 2	3.74 - 2
I 131	8.04 D	0.0	8.61 + 5	1.99 - 8	0.0	0.0
Xe 131 M	11.99 D	0.0	6.39 + 3	1.07 - 5	0.0	0.0
Xe 133	5.29 D	0.0	1.61 + 6	2.80 - 15	0.0	0.0
Cs 134	2.06 Y	0.0	2.46 + 5	1.76 + 5	8.38 + 3	1.89 + 0
Cs 135	2.3 + 6 Y	0.0	2.86 - 1	2.86 - 1	2.86 - 1	
Cs 136	13.0 D	0.0	6.08 + 4	2.16 - 4	0.0	0.0
Cs 137	30.1 Y	0.0	1.08 + 5	1.05 + 5	8.56 + 4	4.82 + 4
Ba 137 M	29.98 Y	0.0	1.05 + 5	9.85 + 4	8.00 + 4	4.50 + 4
Ba 140	12.79 D	0.0	1.45 + 6	3.79 - 3	0.0	0.0
La 140	40.23 H	0.0	0.50 + 6	4.36 - 3	0.0	0.0
Ce 141	32.53 D	0.0	1.39 + 6	5.67 + 2	0.0	0.0
Pr 143	13.58 D	0.0	1.20 + 6	1.29 - 2	0.0	0.0
Ce 144	284.4 D	0.0	1.11 + 6	4.56 + 5	1.51 + 2	3.27 - 8
Pr 144	17.28 M	0.0	1.12 + 6	4.56 + 5	1.51 + 2	3.27 - 8
Nd 147	10.99 D	0.0	5.88 + 5	7.39 - 5	0.0	0.0
Pm 147	2.623Y	0.0	1.02 + 5	8.39 + 4	7.75 + 3	9.83 + 0
Pm 148 M	41.3 D	0.0	3.88 + 4	9.40 + 1	2.57 - 22	0.0
Pm 148	5.37 D	0.0	1.99 + 5	7.55 + 0	2.07 - 23	0.0
Am 151	93 Y	0.0	1.25 + 3	1.24 + 3	1.16 + 3	9.63 + 2
Eu 152	13 Y	0.0	1.25 + 1	1.18 + 1	7.04 + 0	1.93 + 0
Fs 153	241.5 D	0.0	3.55 + 1	1.25 + 1	1.02 - 3	4.06 - 17
Eu 154	8.6 Y	0.0	6.99 + 3	6.69 + 3	4.53 + 3	4.17 + 2

Table 5.1
(Continued)

Eu 155	4.8 Y	0.0	7.48 + 3	5.11 + 3	1.63 + 2	4.78 + 1
Eu 156	15.2 D	0.0	2.26 + 5	1.07 - 2	0.0	0.0
Tb 160	72.3 D	0.0	1.28 + 3	3.84 + 1	7.21 - 13	0.0
Total		0.0	2.10 + 7	2.22 + 6	3.18 + 5	1.63 + 5

- (1) Year = Y
Day = D
Hour = H
Minute = M
Second = S

- (2) Read as 7.09×10^2

Table 5.2

**CALCULATED RADIOACTIVITY OF FISSION-PRODUCT ISOTOPES
PRESENT IN SPENT PWR-U FUEL (PER CANISTER)**

Power = 30.00 MW, Burn-up = 33,000 MWD, Flux = 2.92×10^{13}
N/cm²- sec.

Nuclide Radioactivity Curies
Basis = 1 Canister (5 MTHM)

<u>ISOTOPE</u>	<u>DISCHARGE</u>	<u>10 Y</u>	<u>35 Y</u>
H 3	$3.55 + 3^{(1)}$	$2.02 + 3$	$4.96 + 2$
Se 79	$1.99 + 0$	$1.99 + 0$	$1.99 + 0$
Kr 85	$5.65 + 4$	$2.98 + 4$	$5.90 + 3$
Rb 86	$2.47 + 3$		0.0
Sr 89	$3.59 + 6$	$2.58 + 15$	0.0
Sr 90	$3.88 + 5$	$3.04 + 5$	$1.68 + 5$
Y 90	$4.04 + 5$	$3.04 + 5$	$1.68 + 5$
Y 91	$4.69 + 6$	$9.45 - 13$	0.0
Zr 93	$9.45 + 0$	$9.45 + 0$	$9.45 + 0$
Nb 93 M	$7.25 - 1$	$4.20 + 0$	$9.60 - 2$
Zr 95	$6.85 + 6$	$8.35 - 11$	0.0
Nb 95 M	$1.40 + 5$	$1.77 - 12$	0.0
Nb 95	$6.90 + 6$	$1.81 - 10$	0.0
Tc 99	$7.15 + 1$	$7.15 + 1$	$7.15 + 1$
Ru 103	$6.10 + 6$	$1.05 - 21$	0.00
Rh 103 M	$6.10 + 6$	$1.05 - 21$	0.00
Ru 106	$2.73 + 6$	$2.75 + 3$	$1.02 - 4$
Rh 106	$3.70 + 6$	$2.75 + 3$	$1.02 - 4$

Table 5.2
(Cont'd)

Pd 107	5.50 - 1	3.50 - 1	5.50 - 1
Ag 110 M	1.84 + 4	8.30 - 1	9.90 - 12
Ag 110	7.95 + 5	1.08 - 1	1.34 - 12
Ag 111	1.98 + 5	0.0	0.0
Cd 113 M	5.25 + 1	3.20 + 1	9.95 + 0
In 114 M	7.75 + 0	7.90 - 22	0.0
In 114	1.11 + 1	7.65 - 22	0.0
Cd 115 M	7.76 + 3	7.40 - 23	0.0
Sn 117 M	2.98 + 2	0.0	0.0
Sn 119 M	8.70 + 1	3.28 = 3	1.62 - 14
Sn 123	4.44 + 4	7.10 - 5	6.60 - 26
Te 123 M	3.05 + 0	1.22 - 9	0.0
Sb 124	2.03 + 3	9.60 - 16	0.0
Sh 125	6.75 + 4	0.0	0.0
Sb 125	4.35 + 4	3.39 + 3	6.00 + 0
Te 125 M	1.56 + 4	1.41 + 3	2.30 + 0
Sn 126	2.73 + 0	2.73 + 0	2.73 + 0
Sb 126 M	1.13 + 2	2.73 + 0	2.73 + 0
Sb 126	2.85 + 2	2.71 + 0	2.71 + 0
Te 127 M	7.70 + 4	6.50 - 6	0.0
Te 127	3.60 + 5	6.45 - 6	0.0
Te 129 M	2.87 + 5	0.0	0.0

Table 5.2

(Cont'd)

Te 129	1.69 + 6	0.0	0.0
I 129	1.86 - 1	1.87 - 1	1.87 - 1
I 131	4.31 + 6	0.0	0.0
Xe 131 M	3.20 + 4	0.0	0.0
Xe 133	8.05 + 6	0.0	0.0
Cs 134	1.23 + 6	4.19 + 4	9.45 + 0
Cs 133	1.43 + 0	1.43 + 0	1.43 + 0
Cs 136	3.04 + 5	0.0	0.0
Cs 137	5.40 + 5	4.28 + 5	2.41 + 5
Ba 137 M	5.05 + 5	4.00 + 5	2.25 + 5
Ba 140	7.35 + 6	0.0	0.0
La 140	7.50 + 6	0.0	0.0
Ce 141	6.95 + 6	0.0	0.0
Pr 143	6.00 + 6	0.0	0.0
Ce 144	5.55 + 6	7.55 + 2	1.64 - 7
Pr 144	5.60 + 6	7.55 + 2	1.64 - 7
Nd 147	2.94 + 6	0.0	0.0
Pm 147	5.10 + 5	3.88 + 4	4.92 + 1
Pm 148 M	1.94 + 5	1.29 - 21	0.0
Pm 148	9.95 + 5	1.04 - 22	0.0
Sm 151	6.25 + 3	5.80 + 3	4.82 + 3

Table 5.2

(Cont'd)

Ev 152	6.25 + 1	3.52 + 1	9.65 + 0
Gd 153	1.77 + 2	5.10 - 3	2.03 - 16
Ev 154	3.50 + 4	2.27 + 4	2.09 + 3
Ev 155	3.74 + 4	8.15 + 2	2.39 + 2
Ev 156	1.13 + 6	0.0	0.0
Tb 160	6.40 + 3	3.61 - 12	0.0
Total	1.05 + 8	1.59 + 6	8.15 + 5

(1) Read as 3.55×10^3

6.0 Summary

Previous sections have dealt with the curie content of spent fuel from a typical PWR operating within the United States. These sections have dealt with specific isotopes within the spent fuel, i.e., fission products and actinides. The total curie content, per MTHM, of the spent fuel when removed from the reactor will be the sum of the individual values found in Tables 3.1, 4.1 and 5.1. From these tables we see that the total curie content expected at discharge for the cladding plus spent fuel assembly will be $(1.18 \times 10^5 + 1.96 \times 10^7 + 9.80 \times 10^7) = 1.18 \times 10^7$ curies. The majority of this comes from the fission products, but the actinides, because of their long half-lives, will be of the major concern for long term management of waste. It will be the gaseous fission products, however, which will be most readily detected by gas sampling devices in case of canister leakage, and emphasis should be placed on monitoring equipment capable of detecting these isotopes.

Table 6.1 gives a summary of the expected curie content stores within a typical waste

canister. As shown in this table each canister will have approximately 2×10^8 Ci at discharge from the reactor. This will decay to approximately 2×10^5 Ci by the end of the surface storage period, and to 9.5×10^5 curies by the end of the retrieval period considered for this report.

Table 6.1

CALCULATED RADIOACTIVITY OF ALL ISOTOPES
PRESENT WITHIN SPENT FUEL STORAGE CANISTER

Basis = 1 Canister, 1 ft. Diameter, 16 ft. Length filled with

5 MTHM charged to reactor

Radioactivity, Curies

	<u>Discharge</u>	<u>10 Y⁽¹⁾</u>	<u>35 Y⁽²⁾</u>
Light element isotopes in cladding	5.90 + 5 ⁽³⁾	1.20 + 4	2.56 + 3
Actinide Isotopes in Fuel	9.80 + 7	3.60 + 5	1.31 + 5
Fission Product Isotopes in Fuel	1.05 + 8	1.59 + 6	8.15 + 5
Total	2.04 + 8	1.96 + 6	9.49 + 5

(1) End of surface burial period, beginning of underground isolation period

(2) End of underground retrieval period

(3) Read as 5.90×10^5