PNC TN4600 88-006

アクティブ中性子測定装置調査
海外出張報告書

昭和63年12月

動力炉・核燃料開発事業団

1. 期間及び出張先

(1) 期　間　1988年9月19日（月）～28日（水）

(2) 出張先　パジャリト社，アイダホ国立研究所，サンディア国立研究所

2. 出張者　本　社　環境資源部　環境計画課　天野　一朗

東海事業所　環工部　減容技術開発室　宮田　和俊

3. 目　的

米国パジャリト社（ＰＳＣ社）から購入するアクティブ中性子測定装置（納期：昭和64年11月30日）の設計製作に際し，日本での許認可に必要な条件（特に耐震，遮へい）が十分に反映されていることを確認するとともに，同型式装置運転状況（アイダホ国立研究所）の確認ならびに中性子発生装置等に関する技術情報を得る。

4. 非破壊測定（ＮＤＡ）技術開発の現状

放射性廃棄物中の核物質の測定に適用されるＮＤＡには，パッシブ法とアクティブ法がある。

① パッシブ法

核物質のほとんどの同位体はα崩壊やβ崩壊に伴なって，それぞれ固有のγ線を放出する。このγ線を測定して核物質量を測定するのがパッシブγ線法である。又Ｕ，Puの同位体のあるものは，ある割合で自発核分裂をおこし中性子を放出する。又α崩壊をするものでは，放出された数MeVのエネルギーを有するα粒子が試料中の軽元素と反応して中性子を放出する。この（α，ｎ）反応をおこす軽元素としては，酸化物燃料では^{17}O，^{18}Oが，フッ化物では^{19}Fが知られている。

これら自然崩壊にともなって放出される中性子を計数して，核物質の定量を行なう方法がパッシブ中性子法である。

パッシブγ線法では，検出対象がγ線であるため，試料容器や試料自体による吸収があり，定量分析を行うには吸収補正を行なう必要がある。ことにエネルギーの低いγ線の場合には，試料容器及び廃棄物自体による吸収が大きくなり測定が困難になる。吸収

補正を行なう方法としては，外部γ線源を使用して，その吸収を測定する方法と試料中の同一の核種から放出されるエネルギーの異なるγ線強度を測定して核データと比較することにより吸収を評価する方法とがある。

パッシブ中性子法は，検出対象が中性子であるため，試料容器，その他に対する透過性が一般にγ線より高く，鉄，鉛等を含む可能性がある廃棄物などの測定にも利用される。

パッシブ中性子法には大別して２つの方法がある。１つは試料から放出される中性子を単純に計数する全中性子計数法であり，他は自発核分裂で放出される複数個の中性子を適当なゲート巾で同時計数して，（α，ｎ）反応で単発的に発生する中性子と弁別し，自発核分裂物質の測定を行う同時計数法である。（α，ｎ）反応で発生する中性子量は試料の化学形態により異なり，従って全中性子計数法はその影響を受けるが，同時計数法はその影響を受けない。しかし，いずれの方法でも，中性子の減速や吸収，増倍の効果が試料や検出体系によって異なるため，外部中性子源を使用する等により，その補正が必要になる。代表的な装置としては，Neutron Well Coincidence Counterがある。

② アクティブ法

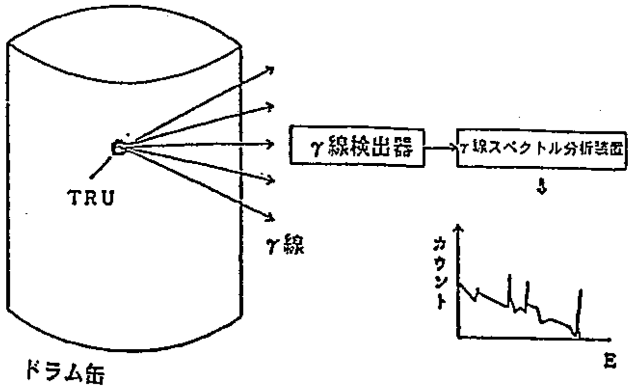
試料を中性子，γ線などで照射して，核物質に核反応を誘起し，その結果発生する放射線を計数することで核物質の分析を行なう方法がアクティブ法である。

核反応としては，（ｎ，ｆ），（ｎ，γ），（γ，ｆ）反応等が利用されるが，ＲＩ中性子源や，小形加速器による中性子発生装置が利用しやすいこと，反応断面積が大きく高い分析感度が期待できることなどから（ｎ，ｆ）反応がもっとも一般的である。アクティブ法は照射線源強度を増大することにより分析感度を高めることが出来るので，一般的にパッシブ法よりは，感度が良い。また，高速中性子を用いれば透過性が高いので，大きな試料や高密度の試料の測定が可能である。

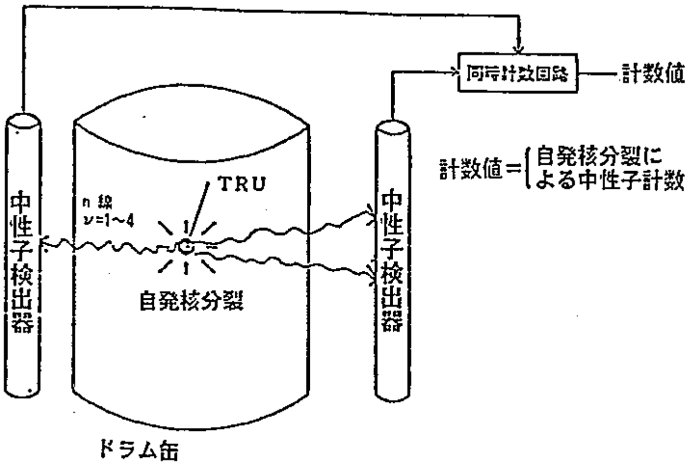
もっとも一般的な中性子核分裂法の場合の計数対象は即発中性子，遅発中性子，遅発γ線などである。即発中性子計数の場合には，照射用中性子との弁別が必要となる。この方法として，線源から発生する中性子と試料から核分裂で発生する中性子のエネルギー差を利用するエネルギー弁別法や減速材中の中性子消滅時差を利用するＤＤＴ法（Dif-ferenical Die-away Technique），核分裂中性子放出の同時中性子を利用した同

時計数法などがある。

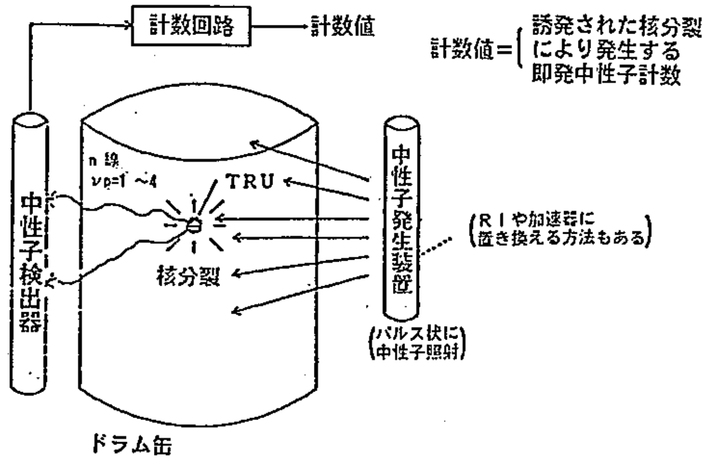
ＮＤＡ装置原理図を図４－１に，放射性廃棄物中のＴＲＵ核種測定技術開発の現状を表４－１に示す。



パッシブγ線法の原理図



パッシブ中性子法（コインシデンス法）の原理図



アクティブ中性子法の原理図（DDT法）

図４－１　ＮＤＡ装置原理図

5. アクティブ中性子測定装置に関するＰＳＣ社との打合内容

5－1　全　般

(1) 本打合せに先立ち，ＰＮＣで購入予定のアクティブ中性子測定装置（以下，Ａ／n装置と称する）と同型式の実機を９月21日，アイダホ州アイダホフォールにあるアイダホ国立研究所（Idaho National Engineering Laboratory ： ＩＮＥＬ）で見学した。

(2) Ａ／n装置製作，据付，運転についての日本国内での許認可に関連するＰＮＣからＰＳＣ社への質問事項に対する回答を再確認または入手すべくＰＳＣ社と協議した（９月22日～24日）。

5－2　協議事項及びＰＳＣ社の回答

(1) 装置サイズは，112Ｌ×106 Ｗ×87Ｈ（インチ）で変更ないか。

多少小型化される。現設計では，図５－１に示されるように 120Ｌ× 54.37Ｗ× 71.87 Ｈ（インチ）であるが確定寸法ではない。ＰＷＴＦ P-303室の天井には換気ダクトが通っているため，廃棄物ドラム缶の搬出入に天井クレーンを追加設置することができない。従って，ドラム缶搬出入はドラムポーターによることとする。

(2) 中性子発生装置

① 初期トリチウム装荷量　：　８～10Ci

② 中性子発生密度　：　10^6 n/pulse ，５～10μsec/pulse ，50pps(pulse/sec)

③ 運転モード　：　運転は上記のパルス１モードのみであり，連続した中性子の発生は不可能である。

④ 寿命を縮める原因　：　ゼータトロンへの高圧電源のブレイクが多発するとゼータトロン内部部品がへたり，ゼータトロンの寿命が短縮される。検出体系のドアを確実に閉じる等，注意を要す。

⑤ 中性子パルス間で中性子発生装置から発生する中性子はほとんどないため，ノイズ発生等の問題はない。

(3) ＰＷＴＦ P-303室の室内環境

① 換気（エアコン）は必ず行い，室内相対湿度を50%以下に維持することが望ましい。

オークリッジ国立研究所では，高湿度のため，プロポーショナルカウンターのコネクションが腐食してトラブルを起こしたことがある。

② P-303室のバックグランド0.4cpsについては，測定上問題はない。（バックグランド測定時の計測器の直径，長さ及び遮へい材厚さについてはＰＳＣ社確認済）

③ Ａ／ｎ装置のメンテナンスに要するスペース及び高さについては，Ａ／ｎ装置各モジュールとも全て手作業で分解可能（ピースの最大重量は約40kg）ではあるが，装置外形図及びレイアウト図を基に確認する必要がある。

(4) ユーティリティ

使用するユーティリティは電気のみであり，以下の３系統に分けることとする。

① ドラム回転用モータ ： 100Ｖ 50Hz 40Ａ

② 電気計装品（プリアンプ，高圧電源，プロポーショナルカウンタ等）

： 100Ｖ 50Hz 40Ａ

③ 制御装置(P-304室）用： 100Ｖ 50Hz 50Ａ

廃棄物測定時のノイズ及びサージングを防止するため，上記電気系統のうち①と②，③とは必ず分けること，サージプロテクタを設置すること，ならびに，ケーブルトレイについても動力用と制御用を分離することが必要である。

(5) 二次廃棄物とその処理法

Ａ／ｎ装置運転中に発生する二次廃棄物はゼータトロン（トリチウムは８～10Ci残）のみである。これは，トリチウムの残量は充分でも，高電圧ON-OFFの繰返し運転により，部品が劣化しゼータトロンの寿命がくるためである。

処理法については以下の２通りがある。

（第１案） ゼータトロンをＰＳＣ社へ返送する。

（第２案） 日本国内で適切な処理を行う。

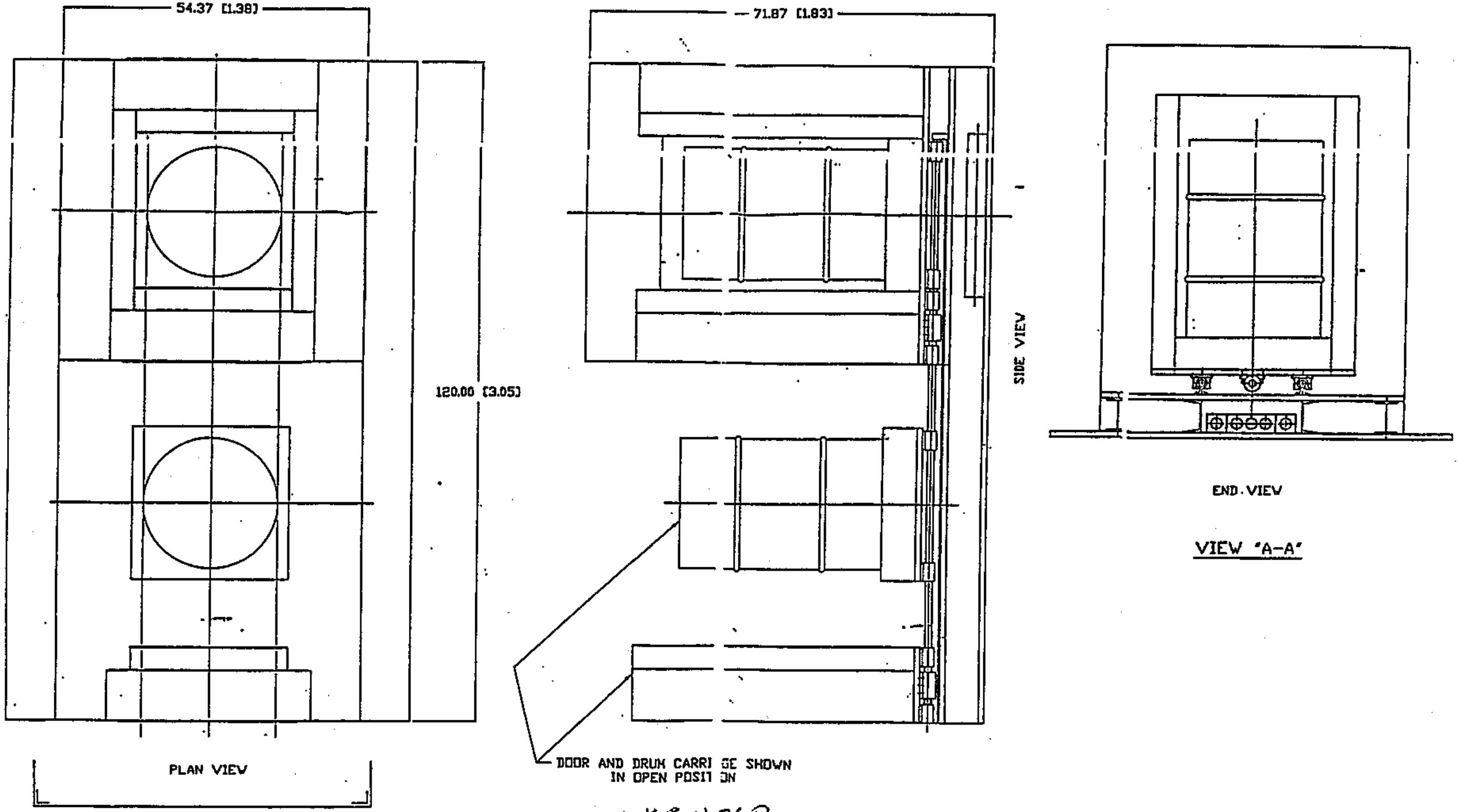

54.37 [1.38]
120.00 [3.05]
PLAN VIEW
"A"
"A"
71.87 [1.83]
SIDE VIEW
DOOR AND DRUM CARRI GE SHOWN
IN OPEN POSIT ON
END VIEW
VIEW "A-A"

図5-1 A/M装置外形図

(6) 熱，酸及び放射線に対する装置材質

A／n装置運転中は，ゼータトロン表面でも最高49℃にしか達しないため，火災発生の可能性は全く無く，耐熱上の配慮は不要である。

また，いかなる酸も使用していないので，耐酸性の配慮も不要である。

耐放射線性についても，米国内での実績上問題ないと考えられる。

(7) 安全装置

A／n装置に取付ける安全装置は以下の3種類とする。

① 装置のドアが「開」の場合ゼータトロンが作動しない，または作動中に装置のドアが「開」の場合ゼータトロン用高圧電源がブレークするインタロック機構。

② 装置運転中を示す警告灯

③ P-303室ドアが「開」の場合，装置が作動しないインタロック機構。

(8) 遮へい計算

① 中性子による線量率

グラファイト及びポリエチレンの遮へい材付の場合，装置表面で2.3mrem/hr，装置から4m離れた点(P-303室のドアの位置）で0.14mrem/hr である。

② γ線による線量率

約1MeVのガンマ線が発生し，これにより装置表面で0.09mrem/hr，装置から4m離れた点で0.0056mrem/hr と中性子に比較し，極めて小さい。

③ ドラム内のPu核分裂による線量率

約0.0003mrem/hr であり無視できる。

遮へい計算の詳細は，入手資料－4 Calculation And Measurement of Pulsed 14MeV Neutron Generator Neutron And Gamma Ray Dose Ratesを参照のこと。

(9) 耐震計算

入手資料－5 Seismic Analysis JGC／PNC NDA System を参照のこと。

(10) 検査項目と判定基準

部品，材料とも，military specificationかU.S DOE standard specificationのいずれかを使用する。

ＰＮＣの立会検査項目及び判定基準は，日本側で決定の上，別途連絡することとした。

(11) 工業所有権

Ａ／ｎ装置に関する特許はＤＯＥが所有している。ＰＳＣ社はＤＯＥから設計，製作，販売のライセンスを取得している。従って，本Ａ／ｎ装置の製作等に関し，改めてライセンスを取得することは無い。

また，Ａ／ｎ装置を米国から日本へ輸出する際の，輸出許可申請手続（対ＤＯＣ：商務省）は適宜ＰＳＣ社及びＪＧＣ－ＵＳにて行なう。

(12) その他

① Ａ／ｎ装置で使用するパーソナルコンピュータについては，ＩＢＭ，ＮＥＣ等ほとんどの機種を適合することができる。日本側で希望があれば連絡すること。

言語はＦＯＲＴＲＡＮである。

② Ａ／ｎ装置のドラムローダの高さについては，ＰＮＣで使用するハンドリング装置に合わせるため，別途連絡すること。

③ Ａ／ｎ装置の電気仕様に関し，契約仕様書では 100V，50Hzと示されているが，米国内で調達できる部品は 120Ｖ60Hz仕様である。したがってＰＳＣ社では，120Ｖ60Hz仕様の部品により装置を製作するが，取合を 100Ｖ50Hzでできるよう，トランス(100Ｖから 120Ｖに増大させる）及び周波数変換器をＡ／ｎ装置に組込む方法を考慮している。

日本国内での安全審査，予備品の調達及び修理等で問題がないか否か調査検討しＰＳＣ社へ回答する。

④ 納期の短縮について，ＰＳＣ社としては短縮を保証できないが努力する旨の回答を得た。

5－3　日本側／米国側，双方における検討事項

今回の打合せで確認しきれなかった事項については，その項目及び提出期限を以下に示す。

1）ＪＧＣ／ＰＮＣ　⟶　ＰＳＣ

① DIAMETER, LENGTH, OF PROPORTIONAL COUNTER, AND THICKNESS OF MODERATOR (OCT. 10)

② THE HEIGHT OF DRUM LOADER (OCT. 10)

③ COMMENT FOR THE LAY OUT (OCT. 10)

④ HUMIDITY OF ROOM P303, HOW MUCH % ? (OCT. 10)

⑤ 100V, 50Hz OR 120V, 60Hz OUR OPINION (OCT. 17)

⑥ INSPECTION ITEMS LIST
(WHICH ＪＧＣ／ＰＮＣ WILL INSPECT AT FACTORY) (OCT. 10)

⑦ STANDARD SOURCE FOR CALIBRATION (AFTER PSC's ANS)

⑧ COMMENTS FOR SEISMIC ANALYSYS (OCT. 10)

ＰＮＣ　I. Amano

ＪＧＣ　Tatsuo Sawaki

ＰＳＣ　John Caldwell

2）ＰＳＣ　⟶　ＪＧＣ／ＰＮＣ

① CONDITIONS
ATTENTIONS　} FOR NDA SYSTEM,
MANNUALS*　* EXAMPLE (OCT. 20)
(*OCT. 20)

② ACTUAL MEASUREMENT DATA FOR SURFACE DOSE RATE (OCT. 20)

③ WHAT IS NECESSARY FOR CALIBRATION ?

④ INSPECTION CRITERIA, PROCEDURE, (＋２WKS) or (OCT. 20)
(Which every is lafer)

5－4　入手資料リスト

1）Operation and Life of the Zetatron :

A Small Neutron Generator For Borehole Logging

L. A. Shope, R. S. Berg, M. L. O'Neal and B. E. Barnaby

(IEEE Transactions on Nuclear Science Vol NS-28. No2, April 1981)

2）ＴＲＵ Waste Attribute Measurements Using A Combined Passive And Active Neutron Assay System

John. T. Caldwell

3）Making Transuranic Assay Measurements Using Modern Controllers

(Form the 9th ＥＳＡＲＤＡ Symposium 12-14 May 1987)

T. H. Kuckertz, J. T. Caldwell, P. A. Medvick

W. E. Kunz, and R. D. Hastings

4）Calculation And Measurement of Pulsed 14MeV Neutron Generator Neutron And Gamma Ray Dose Rates

(ＰＳＣ Informal Report ＰＳＣ-88011 Sept23 1988)

J. T. Caldwell

5）Seismic Analysis ＪＧＡ／ＰＮＣ ＮＤＡ System

(letter form T. T. Fife to J. T. Caldwell Sept 6 1988)

T. T. Fife

6）Misc

－ＰＳＣ acceptance criteria and inspection items

－He-3 proportional counter Specifications

－Optima 860 Powered Camac Crate (Specifications)

－Federal Specifications Aluminum MILL Products

6. 施設訪問

6－1　アイダホ国立研究所（ＩＮＥＬ）

(1) 概　要

ＰＮＣで購入予定のＡ／ｎ装置の円滑な導入を目的として，Ａ／ｎ装置ならびにその他の放射性廃棄物処理処分施設を見学した。

(2) 内　容

見学した施設及び設備は以下の通りである。

① Waste Experimental Reduction Facility (ＷＥＲＦ)

本施設はPower Burst Facility（ＰＢＦ）area内にある低レベル放射性廃棄物（可燃性固体，不燃性固体，液体）の減容固化処理施設である。

対象廃棄物および処理方式は以下の様に大別される。

- 金属廃棄物を高温バーナにより切断した後，誘導炉により溶融固化する。
- 難燃性廃棄物をコンパクタ（200ton）により圧縮減容する。
- 可燃性廃棄物（紙，布，木片，酢酸ビニル製の使用済保安用品等のうちα汚染のないもの）を専焼炉により焼却した後，焼却灰をセメント固化する。
- 液体廃棄物をセメント（Grout)固化する。

今回の見学においては，可燃物専焼炉に重点が置かれた。専焼炉の仕様は以下の通り。

処　理　量	：	1,300ドラム／year（実績）
燃 焼 温 度	：	700℃
オフガス処理系	：	バグフィルタ(テフロン製)，プレフィルタ，ＨＥＰＡフィルタ

ＷＥＲＦはＩＮＥＬ内の低レベル放射性廃棄物を定常処理する施設であり，開発的要素の説明は特に受けなかった。また，切断処理対象である汚染されたタンク等の金属廃棄物が施設の裏に野積みされていた。

② Stored Waste Experimental Pilot Plant(ＳＷＥＰＰ)

本施設は，米国内のディフェンス廃棄物（ＴＲＵ廃棄物）を貯蔵する施設であり

Radioactive Waste Mamagement Complex（ＲＷＭＣ）のエリア内にある。同エリアにはShallow Landの低レベル廃棄物試験処理施設も設置されている。

ＳＷＥＰＰはドラム缶内の内容物検視装置，今回の見学の主目的であるＡ／ｎ装置等の検査エリアおよび貯蔵エリアから構成されている。

Ａ／ｎ装置に関しては，標準試験体（Pu含有量が既知のもの）を使用しての運転を見学した。測定結果は実際の含有量とよく一致した。

貯蔵エリアは日本の屋内のテニスコートにおいてよく見受けられる様なシートを加圧して貯蔵庫とするものであり，一昨年，雪の重みで貯蔵庫がつぶれたため鉄骨の骨組を追加したとのことであった。

その貯蔵庫内にドラム缶がスペース板をはさんで６段程度に縦積みされていた。手前のドラム缶のラベルを見るとPu24ｇ含有と記載されており，ＳＷＥＰＰの基本的な考え方が日本では参考にならないことが理解できた。

ＩＮＥＬの廃棄物処理処分の概要に関してはＩＮＥＬパンフレット(P15～19）に記載されており，その部分の仮訳を以下に示す。

<u>アイダホ国立研究所における廃棄物管理計画</u>（ＩＮＥＬパンフレットP15-19の仮訳）

アイダホ国立研究所（ＩＮＥＬ）で管理している廃棄物は以下の２つに分類される。

① ＴＲＵ廃棄物：プルトニウム-239，α核種ならびに20年以上の半減期を有する超ウラン元素で汚染された廃棄物であり，軍事ならびにＩＮＥＬ及びその他国立研究等における研究開発に伴い発生する。

② 低レベルβγ廃棄物：短半減期のβγ核種で汚染され，その放射線量率が低く，比較的短時間でバックグラウンドレベルに減衰する廃棄物である。主にＩＮＥＬで発生したものである。

ＴＲＵ廃棄物は主に，布，紙，プラスチック，金属，ゴム，スラッジ及びコンクリート等である。これらは，ＲＷＭＣ(Radioactive Waste Management Complex)の貯蔵施設で貯蔵されている。

1970年以来，ＴＲＵ廃棄物はＴＳＡ(Transuranic Storage Area)で貯蔵されている。廃棄物は，200ℓドラム缶，鋼製コンテナあるいはガラス繊維強化木製コンテナに収

納されている。コンテナは土壌により区切られたセルに積重ねた後，合板プラスチック及び約１ｍの土壌により覆われている。

ＴＳＡには2つの貯蔵場（45ｍＷ×230 ｍＬ）があり，１つは1975年に満杯になったため土壌で閉鎖した。

ＩＮＥＬで貯蔵中のＴＲＵ廃棄物は，1989年にニューメキシコ州のＷＩＰＰ(Waste Isolation Pilot Plant)に永久処分する計画である。貯蔵場から取り出した廃棄物についてはＩＮＥＬから搬出する前に，ＷＩＰＰ受入条件を満足することを確認しなければならない。

ＩＮＥＬでは，ＴＲＵ廃棄物を処理し，ＷＩＰＰで処分するためにＳＷＥＰＰ(The Stored Waste Examination Pilot Plant) 及びＰＲＥＰＰ(The Process Experimental Pilot Plant)を建設した。

ＳＷＥＰＰは，ＷＩＰＰへ搬送されるＴＲＵ廃棄物を対象に重量測定，リアルタイム式Ｘ線撮影法による内容物の確認，アクティブ中性子法によるＴＲＵ核種濃度測定，容器健全性の確認ならびに廃棄物表面線量率を行う。ＷＩＰＰ受入条件を満足する廃棄物は，ＷＩＰＰへ払出するまでＳＷＥＰＰで一時保管する。

ＷＩＰＰ受入条件を満足しない廃棄物はＰＲＥＰＰへ移送し，切断及び焼却した後セメント固化する。最終プロダクトであるセメント固化体(200ℓドラム）はＳＷＥＰＰでＷＩＰＰ受入条件を満足することを確認後一時保管する。

ＰＲＥＰＰの焼却設備は，ロータリキルン焼却炉であり，キルンの温度は， 800～1000℃で，可燃性ガスはアフターバーナで1100～1260℃で焼却する。処理能力は50ドラム／日である。オフガス系は，スクラバ，ミスト除去装置，再加熱装置及びＨＥＰＡフィルタにより構成されている。

6－2　サンディア国立研究所（ＳＮＬ）

(1) 目　的

ＴＲＵ廃棄物の処理処分に関して，ＳＮＬとＰＮＣの意見交換を行う。

(2) 打合相手

Mr. J. E. STIEGLER (Manager)

Dr. D. R. ANDERSON

Ms. M. M. WARRANT

Mr. L. E. LOMESBERG

Mr. G. C. ALLEN

Dr. T. O. HUNTER (Manager for NEVADA HLW)

(3) 打合議題

① ＳＮＬのＴＲＵ廃棄物処分に関連する研究活動の説明（Mr. STIEGLER)

② 上記のうち特にＷＩＰＰに関する説明（Dr. ANDERSON)

③ ＰＮＣのＰＷＴＦに関する説明（宮　田）

④ ＮＥＶＡＤＡ，ＨＬＷ処分に関する説明（Dr. HUNTER)

(4) 内　容

① ＳＮＬのＴＲＵ処分に関連する研究活動

ａ）職員構成

ＳＮＬ全体では 8,200名の職員があり，そのうちアルバカーキには 3,000名のリサーサャ（大卒以上）と 2,000名のテクニシャンがいる。リサーチャの内訳は，2,400名のエンジニア（主としてelectrical, mechanical）と 600名のサイエンティストがいる。アルバカーキ以外ではリバモア，トモファ（ネバダ州）及びニューメキシコ州内の他地域に職員がいる。

ｂ）出　身

今回打合せに参加したＳＮＬの研究者は，ほとんどがＡＴＴ（アメリカの電信電話会社）の子会社の人間である。ＡＴＴがＤＯＥ及びＳＮＬと協力協定を締結し，ＡＴＴ System Inc.から出向している。

ｃ）ＳＮＬの研究項目

研究は下記の３項目に大別される。

　　　ⓐ　核兵器　　(Nuclear Weapon)

　　　ⓑ　通常兵器　(Other Defence Weapon)

　　　ⓒ　エネルギー (Energy)

　　Nuclear Waste Managenemtは項目ⓒに含まれる。

d）Nuclear Waste Managenemtに関する研究内容

　　　ⓐ　ＷＩＰＰ (for defence waste)

　　　ⓑ　Subseabed ＨＬＷ Disposal

　　　ⓒ　Nevada Waste Storage (commercial spent fuel)

　　　ⓓ　Transportation

②　ＷＩＰＰ

ＷＩＰＰはニューメキシコ州カールスバットの地下約2500ft（ある文献では 660ｍとある）に軍の核兵器製造に伴って発生するＴＲＵ廃棄物を処分する施設である。

本施設に処分される廃棄物は表面線量率200mR/h を境界としてCH（直接取扱）ＴＲＵ廃棄物とRH（遠隔取扱）ＴＲＵ廃棄物に分けられる。RH－ＴＲＵ廃棄物はＩＴＶにより監視されながら取扱われる。

安全評価の結果，プラグの必要性はないと判断されたが設計にはプラグが含まれている。

シーリング及びバックフィルには，ベントナイト30%，岩塩70%の混合体を使用する。

安全評価シナリオは廃棄体ドラムの腐食及び岩塩層中のガスの移動によりRI核種の移動が起こるというものであり，ベントナイト等のバリアの劣化も考慮れさている。

Ｑ１）ＷＩＰＰの廃棄体搬入時期の遅れとその理由は何か。

Ａ１）1988年10月の搬入開始時期は早くても1989年の中頃にずれ込むであろう。表向きの理由はrock mechamismの問題であるが，実際にはワシントンの政治的な問題であり大統領のサインをもらうには時間がかかるであろうとのことである。

Ｑ２）搬入廃棄体の認可条件は何か。

Ａ２）搬入廃棄体はＥＰＡの40CFR191を満足するものでなければならない。水分に

ついては重量比で２％以下，体積比で１％以下でなければならない。その他，搬送，処分時に可燃性ガス（例えばH_2）を一定量以上発生しないこと，有機物の蒸気発生が小さいこと，酸素を発生しないことならびにバクテリアによる腐食の影響がないこと等が定められている。

Ｑ３）搬入エレベータは１段(660ｍ直通）か。

Ａ３）１段である。

Ｑ４）リトリーバブルタイムの根拠は何か。

Ａ４）50年間とは１世代の人間が管理できる期間内であると考えられるためである。50年の妥当性については現在も議論されている。

Ｑ５）廃棄体の認可条件としての核種及びその濃度は何か。

Ａ５）主要核種はPu^{239}，Pu^{241}（ビルドアップのAm^{241})，Cf^{252}，Cm^{242} 等であり 200ℓドラム缶中の許容量は合計 200ｇまでである。

③　ＰＮＣにおけるＴＲＵ廃棄物管理

ＰＮＣにおけるＴＲＵ廃棄物管理について，ＰＷＴＦを中心に約20分の説明を行った。

Ｑ１）ＰＷＴＦの建設費はいくらか。

Ａ１）約 120億円である。（サンディア側より予想より安いとのコメントがあった）

Ｑ２）ＰＮＣにおけるＴＲＵ廃棄物の貯蔵量と今後の処理計画はどのようになっているか。また，ＰＷＴＦの償却予定期間は何年か。

Ａ２）現在， 200ℓドラム缶換算で約16,000本のプルトニウム廃棄物が貯蔵されている。今後新規に発生する廃棄物の処理と並行して約16,000本の廃棄物の処理を今後約10年間で行う。この後約10年間は新規発生廃棄物を処理する予定である。したがって，償却期間は20年程度と考えている。

Ｑ３）廃棄物ドラム１本当りの処理費はいくらか。

Ａ３）ＰＷＴＦの年間処理予定ドラム本数は約 2,000本であり，人件費を含む年間経費は約７億円である。

したがってドラム１本当りの処理費は約35万円となる。

但し７億円の中には，施設償却費及び金利等は含まれない。

Ｑ４）含塩素廃棄物を難燃性廃棄物として分別した理由は何か。

Ａ４）ＰＶＣ及びネオプレンゴムなどの含塩素廃棄物は焼却処理した場合，塩素ガスが発生し，塩素ガスによる焼却炉構成材の腐食が生じるため，紙，布，木片類の可燃性廃棄物と分別管理している。

Ｑ５）難燃性廃棄物の処理設備として，酸消化設備とサイクロン焼却設備の２種類を設置した理由は何か。また，どちらの処理法が有効か。

Ａ５）難燃性廃棄物処理法に関する調査研究の結果，有効な処理法として酸消化法とサイクロン焼却法を選定した。コールド試験の結果では優劣が判断できなかったのでＰＷＴＦに２種類の設備を設置し，実証試験を行っている。優劣については，実証試験データの評価解析を行うことにより，今後判断する予定である。

Ｑ６）マイクロ波溶融固化体の鉱物相は何か。

Ａ６）フォルステライトあるいはアノルサイト等である。

最後にMr. StieglerよりＴＲＵ廃棄物の総合処理施設として，非常に興味深い施設である。特に酸消化，サイクロン焼却，マイクロ波溶融及び金属溶融がユニークで興味深い，とのコメントがあった。

④　ＮＥＶＡＤＡ，ＨＬＷ処分

使用済燃料に関する試算は下表の通りである。

	１ＭＴＵ	1 REPOSITORY (70,000MTU)
発　電　量	260,000,000 kwh	80 GWh
電力価値	26,000,000 $	1.85 TRILLION $
処分用基金	260,000 $	18.5 BILLION $

Ｑ１）安全評価における水の影響はどのようなものか。

Ａ１）処分予定地の雨量は0.5mm/yearである。処分予定岩層には帯水層が全く無く，予定岩層より下部の水位の上昇は10,000年で10ｍと評価されており安全の面で心配はない。

この地域の凝灰岩に対する耐震評価でも安全性に関する心配はないとのことであった。

Q２）操業時の安全性はどうか。

A２）立坑のエレベータのトラブルが生じた場合，もしくは緊急避難の必要が生じた場合は以下に示すような横坑（廃棄体の搬入路としても使用される）を通って外部に脱出することが可能である。

この他操業時の従事者の安全に関しては，予備立坑および空気換気坑の設計に関して余裕（stand-by）を持たせた設計となっている。

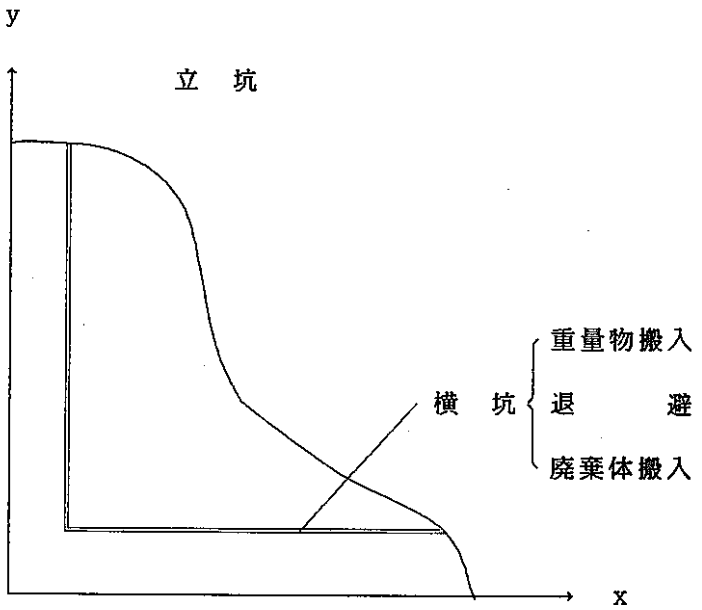


表４－１　放射性廃棄物中のＴＲＵ核種測定技術開発の現状

測定法	測定対象放射線	測定原理	測定対象核種	方式名（機関）	対象物	検出器	測定限界	対象物評価	特徴と問題点	コスト比率	設置スペース
パッシブガンマ法	γ線 （X線）	ＴＲＵ核種がα崩壊する際に放出されるγ線のエネルギースペクトルを測定し，エネルギーピーク位置とエネルギー強度により核種量を求める。	^{239}Pu, ^{241}Pu, ^{241}Am, ^{235}U, ^{237}U等	ＭＥＧＡＳ (Multi-energy Gamma Assay System) (LANL)	低密度雑固体 ｶｰﾄﾝ（６０ℓ）	NaI (Tl)	0.1mg～1mg Pu/10 分 ±30～50%	低レベル 低密度雑固体 (可燃性，難燃性)	〔特　徴〕 ○測定系が簡易，ポータブル可能 ○ＴＲＵ核種の固定が可能 〔問題点〕 ○高β，γ，B.G 廃棄物に不適 ○密度の高い物は不適 ◎大容量の対象物では吸収が大きく不適 ○測定時間が長い ◎ＭＯＸ中のＵの分析は難しい	1	ｶｰﾄﾝ，ﾄﾞﾗﾑ缶ｻｲｽﾞ $\sim 2^W \times 1^L \times \sim 2^H$ (m) ﾎﾟｰﾀﾌﾞﾙｻｲｽﾞ $0.3^W \times 0.5^L \times 0.3^H$ (m) (計測系は別)
				ＳＧＳ (Segmented Gamma Scanner) (CANBERRA)	低密度 カートン， ドラム缶	Ge	ｶｰﾄﾝ ^{235}U　3g/10分 ^{239}Pu 0.2g/10分 ﾄﾞﾗﾑ 缶　Pu～ 1g/10分				
				パンケーキ カウンター(LANL)	低密度カートン	NaI(Tl)	^{239}Pu 6μg/10秒 ^{241}Am8.5mg/10秒±50%	低レベルＴＲＵ廃棄物 ウラン系廃棄物 （ハルに対してはＦ，Ｐからの相関法）			
				原　研	低密度　20ℓ 200ℓ	NaI(Tl)	^{239}Pu　1mg/10分±25% ^{239}Pu　0.02～10g				
				動　燃（ＳＧＳ）	低密度　20ℓ 200ℓ	Ge(Li)	^{239}Pu 15mg/ 8分±25% ^{239}Pu 30mg/17分±20%				
パッシブ中性子法	全中性子法 中性子	廃棄物内のα核種と周囲軽元素との（α，ｎ）中性子と自発核分裂中性子を検出し，α放出核種を定量する。	（全ての放射能量を推定）	ウェルコインシデンスｶｳﾝﾀｰ(LANL)	低～高密度ｶｰﾄﾝ ﾄﾞﾗﾑ	^{3}He又はBF$_3$ ～80本	^{240}Pu　1mg/10分 (既知試料)	低～高密度雑固体 (低，高レベル) a．セメント固化体 b．マイクロ波固化体 c．ガラス固化体 d．アスファルト固化体 e．プラスチック固化体 f．金属鋳塊 ａ～ｆ に関しては，内容物の構成比を考慮した中性子の減速，増倍によるマトリックス補正が必要。 (開発中)	〔特　徴〕 ○マトリックスの影響が少ない ○高β，γ，B.G 廃棄物でも測定可能 〔問題点〕 ○装置が大型，ポータブル不可 ○全中性子法では（α，ｎ）反応の収率が変化するので実用困難 ◎富化度，組成比の情報がなければ濃度算出は困難 ○核種分別は困難 ◎減速材（水分等），吸収材（鉄等）を含む場合は評価が難しい	2	ｶｰﾄﾝｻｲｽﾞ $1^W \times 1^L \times 1^H$ (m) ﾄﾞﾗﾑ 缶 ｻｲｽﾞ $1.5^W \times 1.5^L \times 1.5^H$ (m) ｸﾚｰﾄｻｲｽﾞ $1.5^W \times 3^L \times 1.5^H$ (m) (計測系は別)
				４π中性子ｺｲﾝｼﾃﾞﾝｽｶｳﾝﾀｰ(LANL)	低～高密度 ドラム缶	^{3}He ～80本	^{240}Pu　2mg/10分 ±10～30%　(既知試料)				
				Sensitive Neutron Counter(PNL)	33Φ×50cm (43ℓ)	BF$_3$	約１mgPu/1000分				
	コインシデンス法 自発核分裂中性子	ＴＲＵ核種及びＵ等の核種が自発核分裂し，発生する２個以上の中性子を同時計数測定するもので，自発核分裂性核種の合計量を定量する。	^{238}Pu, ^{240}Pu, ^{242}Pu等 (^{240}Pu(eff)) 自発核分裂性核種の合計量	ミニランダム ドライブ(LANL)	低～高密度 １ℓ程度	ﾌﾟﾗｽﾁｯｸ ｼﾝﾁﾚｰﾀ	^{240}Pu　5mg/10分 ±5%				
				ランダム ドライバ(LANL)	カートン ドラム缶	ﾌﾟﾗｽﾁｯｸ ｼﾝﾁﾚｰﾀ	^{240}Pu 15mg ^{240}Pu 50mg				
				動　燃	低～高密度 カートン	^{3}He 20本	^{240}Pu(eff) 3mg/ 8分±20%	ＴＲＵ廃棄物			
アクティブ中性子法	即発中性子	外部から照射した速中性子によって誘発される核分裂によって生ずる即発中性子の数と減衰特性を利用して，核分裂性核種を計量する。	^{235}U, ^{239}Pu, ^{241}Pu等 核分裂性核種の合計量	ランダム ドライバ(LANL)	低～高密度 １ℓ程度 ｶｰﾄﾝ（20ℓ） ドラム缶	ﾌﾟﾗｽﾁｯｸ ｼﾝﾁﾚｰﾀ	^{239}Pu 10mg/10分±5% ^{239}Pu 30mg/10分 ^{239}Pu150mg/10分	低～高密度雑固体 (低，高レベル) 同　上 (a ～f)	〔特　徴〕 ○マトリックスの影響が少ない ○高β，γ，B.G でも適用可能 ○測定時間が短い 〔問題点〕 ○装置が大型，ポータブル不可 ◎DT中性子発生装置等の加速器を用いるので保守が大変である。 ◎加速器の寿命が比較的短い ◎富化度組成比の情報がなければ濃度算出は困難 ○核分裂性核種以外の測定はできず，核種の定量は不可。 ◎中性子線防護対策が必要 ◎減速材，吸収材の影響がある。 注）◎今後対応可能	4	ﾄﾞﾗﾑ 缶 ｻｲｽﾞ $2^W \times 2^L \times 2^H$ (m) (計測系は別)
				Ｃｆシャフラー (LANL)	7.6ℓ缶 4ℓ ドラム缶	^{3}He ～108本	^{235}U　1g/10分±10% ^{235}U　0.4mg ^{235}U　4.2mg				
	遅発中性子	核分裂を起こした後に発生する遅発中性子を検出し，その数により核分裂性核種を定量する。		ＤＤＴ法 (Differetial Diaway Technique) (LANL)	ドラム缶	^{3}He	^{239}Pu　1mg/ 2分	ＴＲＵ廃棄物 ウラン廃棄物 ハル			
				ＤＤＴ法 LINAC (LANL, EG&G)	ドラム缶	^{3}He	Pu　1mg				

参　考　資　料

参考資料－1

ＴＲＵ廃棄物区分管理技術開発について

(1) 技術開発の背景

動燃事業団に於ける廃棄物中のPu量非破壊測定のための技術開発は昭和56年より実施され，現在Pu廃棄物に対して20～30mgのPuレベルまで検出しうる技術を確立している。

しかし，放射能レベル等に応じた合理的な管理（区分管理）を目指すためには，廃棄物中のPu濃度を数ｎCi／ｇのレベルまで核種ごとに短時間で精度よく検出することが必要となってきている。

米国においては，既にアクティブ中性子測定装置が開発され，βγ核種からの表面線量率１Ｒ／ｈ以下のＴＲＵ廃棄物を対象に測定試験が進展中である。

一方我が国においても，海外再処理返還固化体の受入検査のための測定技術開発（科学技術庁委託５ヶ計画，予算10億）が原研において，昭和59年より開始されている。

また国の原子力委員会放射性廃棄物対策専門部会報告60年10月において，ＴＲＵ廃棄物はその中に含まれる核種，その放射能レベル等に応じて合理的かつ実態に即した管理方法を採用することが強く求められている。また原子力施設等安全研究年次計画にもＴＲＵ廃棄物区分基準の研究が提案されている。

動燃においてはパッシブγ線法及びパッシブ中性子線法に関する技術開発が進行中であり，更にアクティブ中性子線法についても調査研究を行い，その有効性を確認している。

現状のPu取扱施設からのＴＲＵ廃棄物は，βγ核種濃度が低くドラム缶表面においても 200ｍＲ／ｈ以下のものがほとんどである。また減容固化体中のPu量の測定技術については報告がない。

以上の背景と動燃がＴＲＵ廃棄物の主要発生場所であるという現状を勘案すると，その区分管理のための測定法として，パッシブ中性子法及びアクティブ中性子法についての技術開発を一層強

力に進めることが必要である。

(2) アクティブ中性子法技術開発の必要性

ＴＲＵ核種濃度が極端に低いＴＲＵ廃棄物から放出されるγ線（放出核種 Pu^{239}＞ Pu^{238}＞ Pu^{241}）及び中性子線（放出核種 ^{236}Pu ＞ ^{238}Pu＞ ^{242}Pu＞ ^{240}Pu）数は，微少で既存のパッシブγ線法とパッシブ中性子線計測法との組合せでは，測定時間及び検出レベル（20～30mg，測定時間 4,000秒）に限界がある。またパッシブγ線法は，金属等を含む高密度廃棄物では，マトリックスでのγ線吸収が大きく，検出感度が極端に落ちる欠点がある。

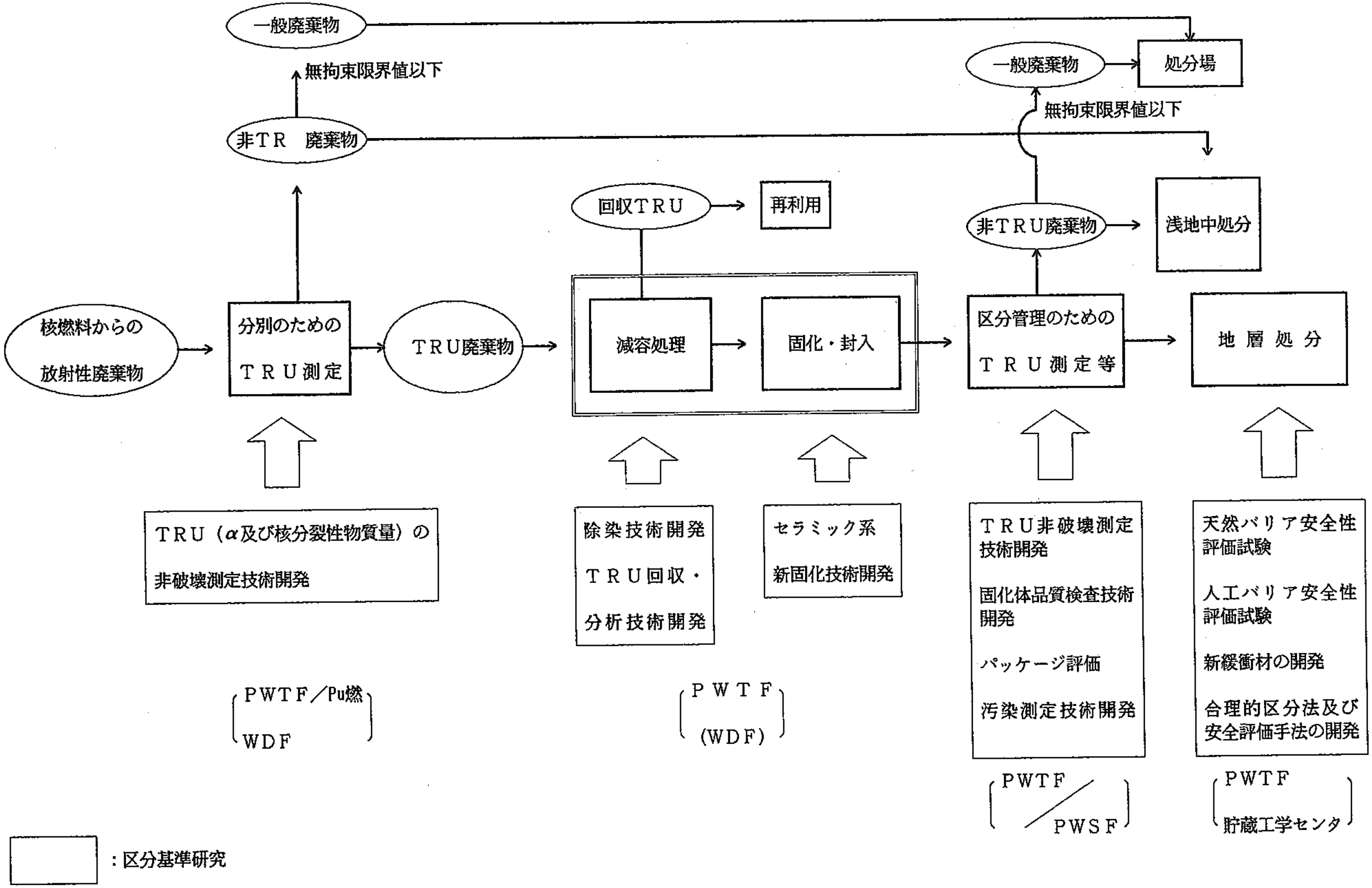
このため密度による影響が少なく，かつPu^{239}，Pu^{241}等核分裂性核種を高感度で測定しうるアクティブ中性子法とPu^{238}，Pu^{240}等の非核分裂性核種が測定できるパッシブ中性子法とを組み合わせた計測法が有効であると考えられる。一方，パッシブ中性子法については，これまでの研究開発で基礎特性を把握し実証化を進める段階であるので，今後は主にアクティブ中性子法について研究開発を進めるべきである。

(3) アクティブ中性子測定装置導入の経緯

動燃におけるＮＤＡ技術開発はプルトニウム燃料取扱施設及び再処理施設等において廃棄物中に移行した放射性核種及びその量を把握するための非破壊測定法の開発が進展中であり，将来の区分管理，廃棄物輸送，処分に係る安全評価の観点からも今後一層充実した研究開発を進める必要のあるテーマの１つである。これまでの開発経緯は以下の通りである。

Ｓ56年	動燃におけるＮＤＡ技術開発を開始
	Pu燃；パッシブγ線法（Ｓ.Ｇ.Ｓ装置導入）

57年	日米廃棄物管理ＷＧにおいて植松理事がアクティブ法を米国より導入することを示唆
58年	ＬＡＮＬ，アクティブ中性子測定装置システム調査実施（於ＵＳＡ） 大洗；パッシブγ線法，βγ核種の固定 （NaI型カートンスキャナー設置） 東海；パッシブ中性子法 （カートンサイズ試作装置製作）
59年	ＰＷＴＦに設置すべくＮＤＡ装置システムの概念設計実施
60年10月	動燃におけるＮＤＡ技術開発計画方針策定 アクティブ中性子測定装置をＰＷＴＦに導入，研究開発を進めることとした。
61年３月	第４回日米廃棄物管理ＷＧにおいてＮＤＡ装置システムについてＤＯＥ／ＬＡＮＬ側の協力確認
５月	上記導入に関して技術仕様，納期，今後の技術協力体制等についてＤＯＥ／ＬＡＮＬと動燃との間に基本的合意がなされた。
６月	61年５月の時点で 1.7億円でＬＡＮＬが製作する約束であったが，６月の時点で 3.4億円に値上げの要請があった。
７月	アクティブ中性子装置については，米国ＤＯＥの民間への技術移転の一環として，ＬＡＮＬよりＰＳＣ社に移った。費用 3.4億円（当方予算 1.7 億円）
７月	値下げの交渉開始
63年３月	動燃は，日輝㈱に対して約 2.2億円で発注した。納期64年11月。 （ＰＳＣ社は日本の代理店として日輝を希望）



ＴＲＵ廃棄物処理処分研究開発概念

参考資料－2

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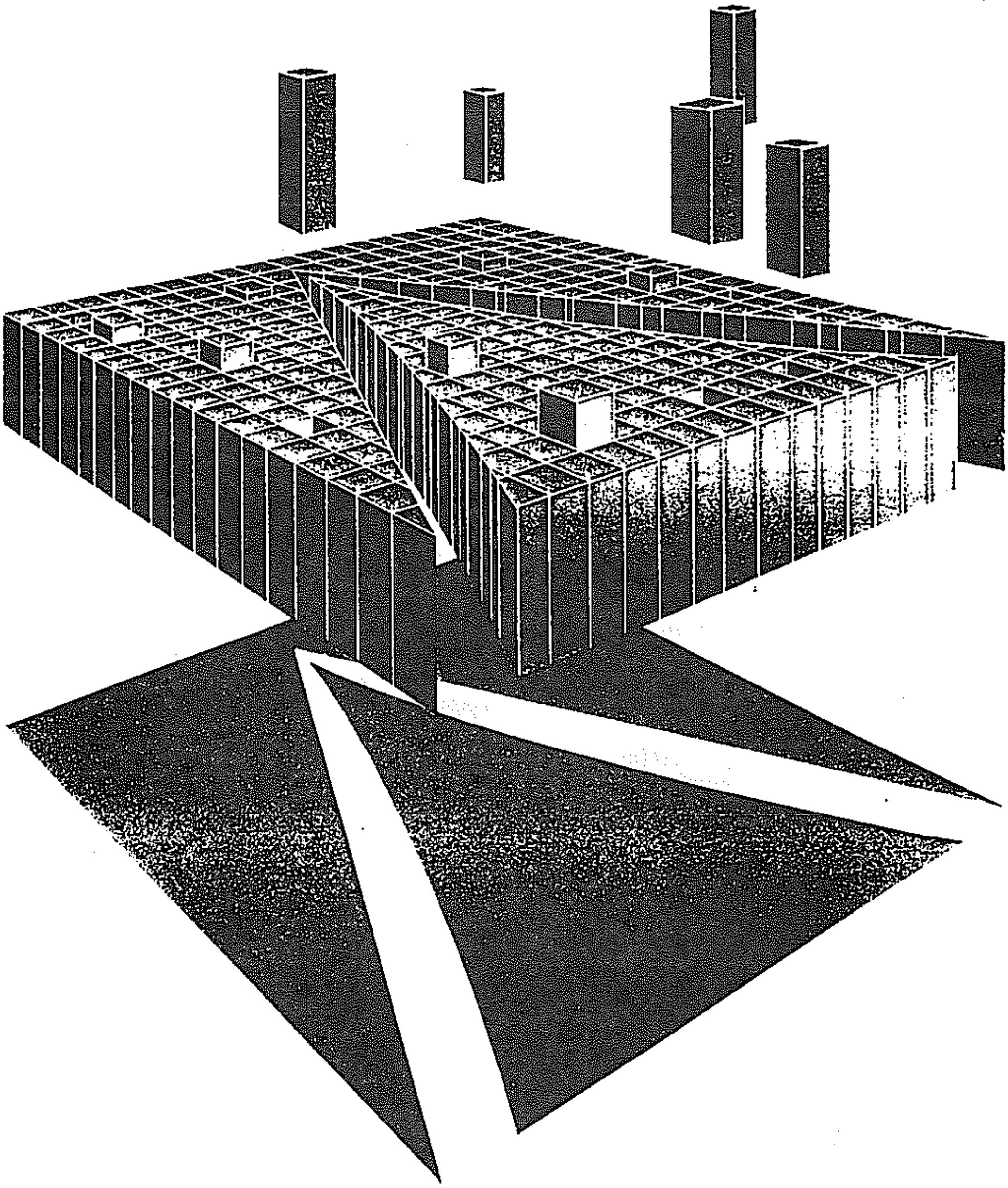

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MAKING TRANSURANIC ASSAY MEASUREMENTS USING MODERN CONTROLLERS*

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Abstract

This paper describes methodology and computer-controlled instrumentation developed at the Los Alamos National Laboratory that accurately performs nondestructive assays of large containers bearing transuranic wastes and nonradioactive matrix materials. These assay systems can measure fissile isotopes with 1-mg sensitivity and spontaneous neutron-emitting isotopes at a 10-mg sensitivity. The assays are performed by neutron interrogation, detection, and counting in a custom assay chamber. An International Business Machines Personal Computer (IBM-PC) is used to control the CAMAC-based instrumentation system that acquires the assay data.

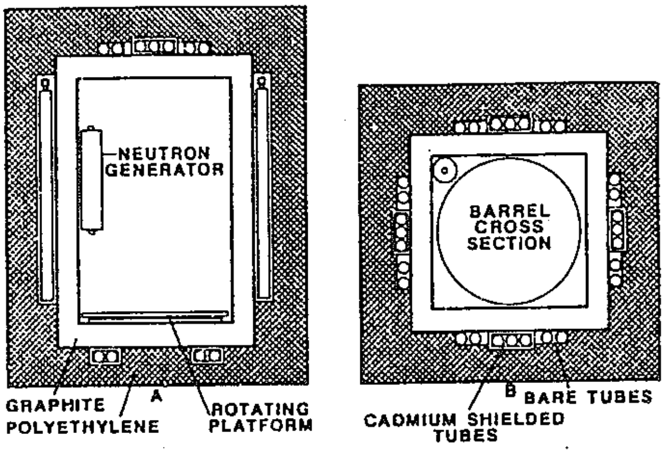

Fig. 1

1. Introduction

This paper describes methodology and computer-controlled instrumentation developed at the Los Alamos National Laboratory that accurately performs nondestructive assays[1-4] of containers bearing transuranic wastes and nonradioactive matrix materials. Transuranic waste is defined in United States Department of Energy (DOE) Order 5820.2 and consists primarily of long-lived (half lives greater than 20 y) isotopes of uranium, plutonium, and other elements with atomic numbers greater than 92. With suitable calibrations, the assay systems described herein can measure fissile isotopes with 1-mg sensitivity and spontaneous neutron-emitting isotopes at a 10-mg sensitivity. The assays are performed by neutron detection and counting and are accurate in the presence of substantial backgrounds generated by alpha-emitting sources and substantial quantities of neutron-absorbing-and-moderating matrices.

These systems are primarily used for the assay of containers bearing plutonium-contaminated waste. United States DOE disposal regulations provide for waste having an activity level less than 100 nCi/g to be buried as low-level waste. Containers with an activity level greater than 100 nCi/g are to be treated as transuranic waste and held in retrievable storage.

2. General Schema

Assay systems have been built for 208-l barrels and 3-m^3 crates.[1-4] A 208-l assay system is shown in Figure 1. The system consists of a graphite inner liner that is 10-cm thick and an outer liner of polyethylene that is 25-cm thick, both of which are held together by aluminum structural members. These materials were chosen to provide the optimum time and energy response to neutrons detected by ^{3}He proportional counters that are present in all sides and in the top and bottom of each assay chamber.

Two different measurements are made in the assay systems. A pulsed neutron generator is used to interrogate a sample during an active assay. The time history of the resulting hard-fission spectrum is integrated along with the time history of the interrogating neutron flux to produce a measure of the mass of any fissile isotopes in the sample. The interrogation also produces a measure of absorbers in the matrix. During a passive assay, neutron counting and neutron-coincidence counting are used to measure the mass of spontaneous neutron-emitting isotopes. Various count statistics are used to eliminate the effects of alpha emitters, moderators, and absorbers in the matrix.

3. Logic Design and Instrumentation

A block diagram of the neutron counting instrumentation and the counting signals produced therein is shown in Figure 2. Four types of ^{3}He proportional counters are shown. The set of cadmium-shielded detectors measures the fission spectrum neutrons and the set of unshielded detectors boosts the efficiency of the total detector system during passive assays. A low-pressure unshielded ^{3}He detector measures the interrogating neutron flux during active assays. Also used during active assays is a collimated ^{3}He proportional counter, whose function is to determine flux inside the assay container. During active assays the pulse generator drives the neutron generator to produce 14-MeV neutron pulses of 10^6 neutrons each. The shielded detectors, flux monitor, and barrel-flux monitor measure

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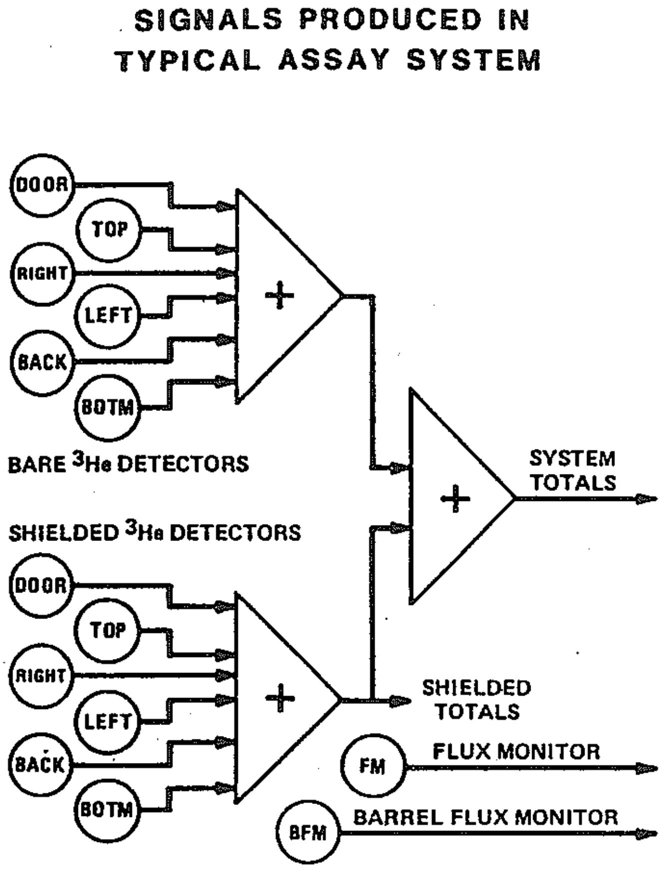

Fig. 2

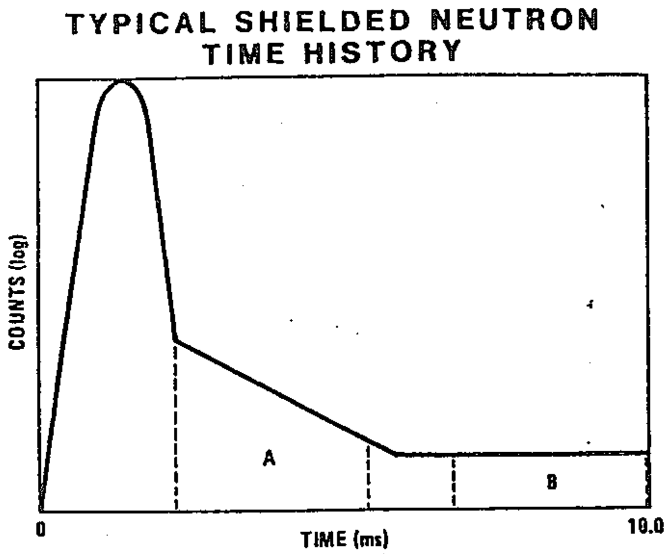

Fig. 3

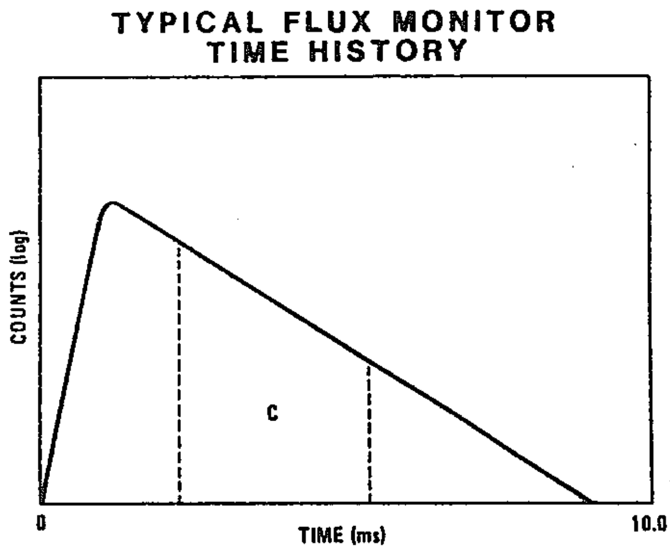

Fig. 4

quantities of interest for determining the fissile mass. During passive assays, neutrons in the cavity are counted by both the shielded and bare detectors. Data acquired by these detectors are used to determine masses of spontaneous neutron-emitting isotopes.

Figure 3 shows a typical, averaged, time history of epithermal neutrons detected by the shielded detector system. At early times, those neutrons detected can be attributed to the pulsed neutron generator. This is denoted by the sharp rise and fall of the curve shown in Figure 3. As the neutrons produced by the neutron generator become thermalized, the nuclei of fissile isotopes capture these neutrons and fission. Any such fission will produce neutrons, which are designated by the area A. The area A gets larger as the amount of fissile material in the cavity increases. The neutron population in the cavity will eventually die away to a background level designated by area B in the curve. This background level results from cosmic rays, spontaneous fission neutron emitters, and (α,n) sources in the drum, and, in general, is indicative of source size. Figure 4 shows a typical, averaged, time history of the interrogating flux in the assay chamber as measured by the flux monitor. Area C in Figure 4 represents the interrogating flux during the time when induced fissions are occurring in any fissile material in the assay chamber. In general, the amount of fissile material in the cavity is proportional to (A - B)/C.

The presence of neutron moderators and absorbers in the assay chamber complicates any measurement.[5] A second flux monitor, called the barrel-flux monitor, is shielded and collimated such that most of what it observes originates inside the container being assayed. The time histories observed by this detector have the same shape as those shown in Figure 4; however, as the amount of absorber in the assay chamber increases, the quantities of neutrons observed by this second flux monitor decreases. Thus, the ratio of the barrel-flux-monitor area C_B to the flux-monitor area C_F gives a measure of absorber effects for which the calibration with known sources can compensate.

Moderator effects are a function of the absorber index and the system-and-shielded-totals rates taken during passive assays. Both the absorber and moderator indices along with the calibration are unique to a particular assay chamber.

Caldwell et al.[1] detail how this information was determined for a drum assay system.

Initial versions of the assay system used a LeCroy 3500 as the system control computer. More recent versions use an International Business Machines Personal Computer (IBM-PC) as the control computer. In addition to a substantial reduction in cost from $15000 to $5000 per control computer, numerous other advantages accrue from the use of the IBM-PC. Some of these are: standard CAMAC modules can be used, data storage medium is compatible with many off-line computer systems, and system control functions are performed noticeably faster. A block diagram of the computer system with CAMAC interface is shown in Figure 5.

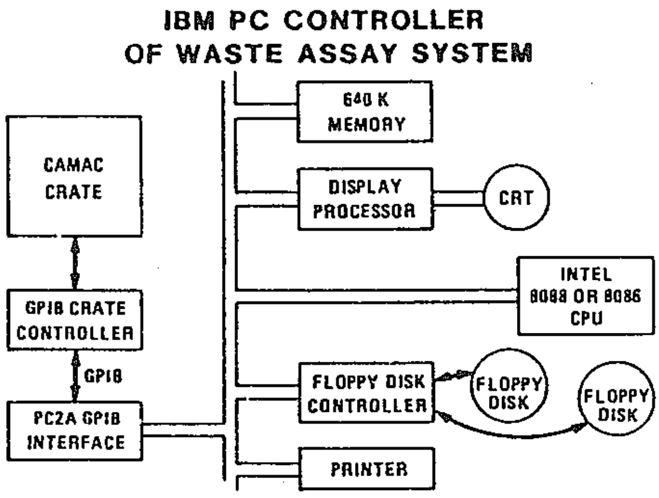

Fig. 5

A block diagram of the data acquisition and control system for active assays is shown in Figure 6. A single CAMAC crate is controlled via a National Instruments PC2A to GPIB interface and a LeCroy 8901A GPIB interface to Type U crate controller. LeCroy 2323 programmable gate generators are used to generate the integration gates for the shielded totals, the flux monitor and the barrel-flux monitor. The integrals associated with areas A, B, and C in Figures 2 and 3 are accumulated in Kinetic Systems 3610 hex scalers under control of these gates.

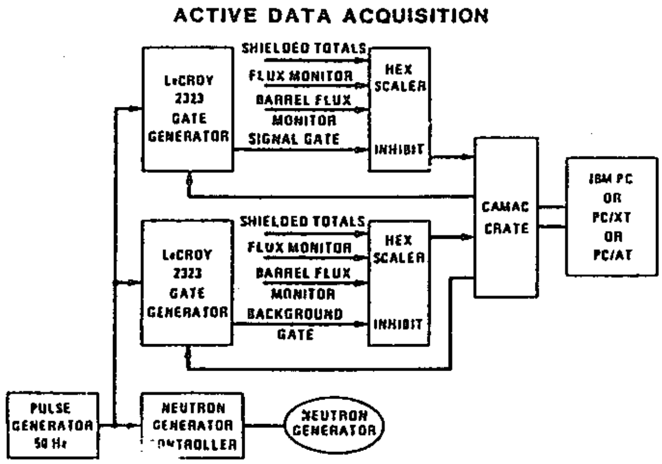

Fig. 6

The LeCroy 2323 gate generators are driven by the same pulse generator, which fires the neutron generator. Thus, average integrals are accumulated in part each time the pulse generator is fired. Typically 2000 pulses are required to assay a barrel and 5000 pulses are required to assay a crate. The integrals are read directly into the IBM-PC for further processing.

Spontaneous fission neutron-emitting isotopes produce neutrons at a rate proportional to mass of any such isotope. Because each fission produces on the average more than one neutron, the neutrons produced by fission are clustered. The neutrons produced by an (α,n) source appear randomly and not in groups. To separate the effects of (α,n) sources from spontaneous neutron-emitting sources, net coincidence rates must be measured accurately. A block diagram of signal-processing electronics for acquiring data during passive assays is shown in Figure 7. The shielded totals and system totals are used to drive coincidence gate generators. The coincidence gates so generated are typically delayed 2 to 5 μs and have lengths of 35 μs and 250 μs for shielded totals and system totals, respectively. If the time that the coincidence gates are active is typically less than 10% of the entire count time, the first neutron in a group will open the gate and the remainder of neutrons in the group will occur during the open gate and be counted in gated scalers (Standard Engineering QS-450). Coincidence rates proportional to spontaneous neutron-emitting masses are determined in this fashion. Additionally, system-gate counts and accurate clocks are used to determine accurate gate lengths. LeCroy 2551 12-channel/scalers perform both this counting and diagnostic counting for monitoring each detector module. We process the systems totals coincidence data (detector efficiency = 12.5%, dieaway time = 100 μs) and the shielded totals data (detector efficiency = 2.8%, dieaway time = 20 μs) separately. The latter is used for high sensitivity assays of low source drums and the former for more accurate assays of high (α,n) source drums.

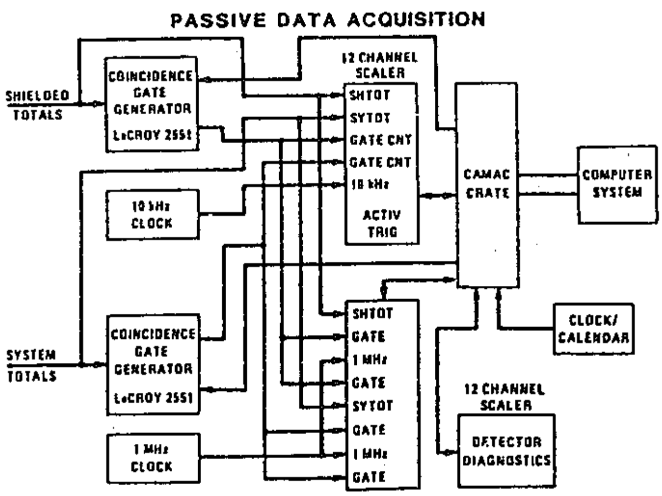

Fig. 7

4. Software

A complex software (180K bytes) code called NEUT has been produced to perform a dialogue with the operator, validate input data, archive collected data, perform active assays, perform passive assays, and generate a production report with the assay quantities in engineering units. Subsections of NEUT are the ANEUT and PNEUT subroutines, which perform the control, data acquisition, and raw data archival functions of the active assays and passive assays respectively. ANEUT and PNEUT were separate programs, which were used to perform the data acquisition for active assays and passive assays, respectively, in the LeCroy 3500 versions of the assay system. Their frequent usage consumed considerable disk-access time on the LeCroy 3500. The NEUT code and the data acquisition and control electronics were written and designed such that they can be used in a production environment with minimally skilled operators.

The NEUT code will run on any hardware configuration of an IBM-PC computer that contains at least 256K bytes of memory and a National Instruments PC2A GPIB interface. NEUT is menu driven with extensive on-line operator instructions and input data checking. NEUT will run from either a diskette system or a hard disk. All data collected either from operator input or from the data-acquisition electronics is saved in an indexed file called DBAS. The logical unit of data storage is the run, which consists of three parts, each of which pertain to the assay of a container: (1) identifying information that is entered by the operator; (2) raw data acquired during an active and passive assay; and (3) processed information such as the activity level. NEUT has the capability of using the raw data found in DBAS and recalculating the processed data quantities. This feature is useful for testing improved data-processing algorithms for computing assay quantities without having to reassay containers.

When NEUT is run, the operator is presented with twelve options that can be selected to perform assays. A list follows:

- Initial configuration
- Display data saved in summary data base
- Assay in standalone mode
- Assay in remote computer mode
- Retrieve data in remote computer mode
- Initialize summary data base
- Acquire passive background
- Recalculate assay from raw data
- Send summary data to remote computer
- ANEUT standalone mode
- PNEUT standalone mode
- Exit to MS-DOS

Several of the options allow the operator to set the environment for assays. The "Initial configuration" is used to change the counting time of passive assays (normally 200 s) and to specify the type (diagnostic, production, or none) of report produced on the printer for each assay. For accurate results, any passive data acquired are corrected to account for background levels of neutron radiation. The "Acquire passive background" option performs a passive assay on an empty container. The background so acquired is used for correcting passive assays of containers of unknown contents. One such background acquisition is performed for every 100 assays.

A single diskette (5-1/4 inch, double-density, double-sided) can contain a bootable operating system (MS-DOS), the code NEUT, and a file DBAS large enough to contain 100 assays. NEUT can be viewed as running from a single isolated diskette, which can contain as many as 100 assays or its logical equivalent (a separate directory) on a hard disk. The "Initialize summary data base" option allows the operator to set up an empty DBAS file. If such a file already exists, NEUT informs the operator of this fact and indicates which runs in the data base have not been used; the operator can then use the existing data base or continue the initialization after being warned of the consequences of overwriting an existing data base. If initialization proceeds, the operator must provide an eight-character volume serial number, which uniquely identifies the data base being created. NEUT then creates the data base with 100 dummy runs, which are replaced with real assays as containers are assayed. The size of DBAS is approximately 40000 bytes. NEUT will not allow more than 100 assays in a data base and will not permit the operator to accidently overwrite a previously completed assay.

Two of the options are used to actually perform the assay, the "Assay in standalone mode" and the "Assay in remote computer mode". Once the "Assay in standalone mode" is selected, an operator dialogue is entered, wherein the following information is collected and verified: (1) run number (1 to 100); (2) primary identification (15 ASCII characters); (3) secondary identification (12 ASCII characters); (4) content code (10 ASCII characters); (5) container weight (kg); and (6) plutonium isotopics. After verifying the input data, NEUT instructs the operator to fire the neutron generator to start an active assay. A passive assay starts automatically when the active assay is finished. When the passive assay is finished, NEUT processes the raw data, archives both raw and processed data, and prints a report of the assay results. Assays of more containers can proceed until 100 assays have been completed. The remote computer mode allows a remote computer (via an EIA-RS232C serial interface) to perform the operator dialogue and otherwise to control program NEUT. This is a full handshake mode, wherein processed data is transferred to the remote computer.

Three options are used to examine processed data from previously performed assays. The "Display data saved in summary data base" will print a report of processed data on the printer. The "Retrieve data in remote computer mode" permits the remote computer to acquire previously processed data. Thus, if the remote computer system is not available, assaying can proceed in the "Assay in standalone mode;" and, when the remote computer becomes available, the assay data can be retrieved without repeating any assays. The "Send data to remote computer" dumps all data in the data base to the EIA RS-232C serial interface with no handshake from the remote computer. This mode is useful for transferring data to a remote computer in instances where control of NEUT by the remote computer is not desired.

Three diagnostic options are also available. These are for collecting, examining, and analyzing specialized raw data. Such diagnostics permit the development and improvement of data-processing algorithms and the isolation of faults in both the hardware and software. The "Recalculate assay from raw data" performs all the assay computations and data archiving that an actual assay does except that the raw data comes from the data base rather than from the assay chamber. This feature is used to test new algorithms on old data. Both the ANEUT and PNEUT standalone modes are used to perform a succession of either active or passive assays and produce a diagnostic report of raw data quantities. The data so acquired can identify system faults and verify the effect of changes in data acquisition hardware and software.

A typical assay sequence consists of the following steps in the order listed:

- Initial configuration
- Initialize summary data base
- Acquire passive background
- Assay in standalone mode
  - 1st run is of known ^{252}Cf sources
  - Up to 99 runs of unknown containers
- Exit to MS-DOS

Once the assaying of containers is started, the procedure need not be continuous. The computer can be powered off and on many times and assaying would continue at the next container to be assayed. If enough time has elapsed that the background may have changed, a new data base should be initialized on another diskette before additional assays are made.

5. Case Example

To date, three assay systems have been installed with the IBM-PC based control equipment described in this paper: a mobile unit[6], a unit at the Rocky Flats Plant in Golden, Colorado, and one in Idaho. Once a system had been set up at the Idaho National Engineering Laboratory (INEL), assay data was produced at a rate faster than could be analyzed by hand. A popular spread sheet (LOTUS 123) was used to keep track of the voluminous data from the INEL installation. A special code (IBM2LOT) was written to convert data in the DBAS files to LOTUS format and keep track of the volume serial numbers of each DBAS file. Approximately 5000 barrels have been assayed and analyzed at this time. Both skilled and unskilled operators have performed the assays. Very little if any data has been lost due to improper operator technique. Additional software has been written that permits use of a Lotus 123 spread sheet as an analysis tool for the relatively large amounts of assay data. Lotus 123 has been used not only to verify and refine the calibrations of the assay systems using the data collected by both LeCroy 3500 and IBM-PC systems from these 5000 barrels, but also to sort difficult assay cases for more sophisticated analyses.

6. Conclusion

The assay system described herein can measure fissile masses as low as 1-mg ^{239}Pu and spontaneous neutron-emitting masses as low as 10-mg ^{240}Pu. This sensitivity can be achieved with substantial quantities of absorber and moderator in the matrix and in the presence of strong alpha-emitting sources. The control-and-data-acquisition system consists of a simple and inexpensive computer system that is a de facto standard in the United States. In addition, the system archives the assay data for later retrieval and analysis. Furthermore, operation of the assay system does not require highly skilled operators.

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TRU WASTE ATTRIBUTE MEASUREMENTS USING A COMBINED PASSIVE AND ACTIVE NEUTRON ASSAY SYSTEM

John T. Caldwell

This write-up provides students a brief, self-contained description of a waste assay technique that uses passive and active neutron counting. It includes the theory behind making assay measurements, a description of how the hardware functions, and provides practical operating instructions.

I. THEORY OF ASSAY

The Los Alamos Assay System (LAAS) is designed to provide accurate and sensitive assays for transuranic (TRU) isotopes in large containers of waste. However, because of the wide variability and heterogeneity common in nuclear waste, such assays are often attribute measurements. The definition of TRU literally is "beyond uranium." What is usually meant by "TRU isotopes," however, is any long-lived alpha-particle emitter that is found in hazardous amounts--at least this is the way it is used in the TRU waste business. Commonly, the TRU isotopes of concern are those of plutonium and americium. In some wastes arising from special programs, one can also find significant amounts of other long-lived alpha particle emitters such as ^{233}U and ^{235}U.

In all cases, we can divide the TRU isotopes into three categories.

<u>Category 1</u>. Those isotopes that can be fissioned with thermal neutrons (i.e., neutrons whose kinetic energy $\approx$ 0.025 eV); ^{239}Pu is a principal example.

<u>Category 2</u>. Those isotopes that don't fission when bombarded with thermal neutrons, but do fission spontaneously at a rate that can be measured; the most important example of this category is ^{240}Pu, which is found in almost all plutonium waste.

Category (3). Alpha-emitting isotopes that don't fission at a measurable rate either spontaneously or with thermal neutrons but that do produce a random flux of neutrons through the (α,n) reactions with the matrix; the most common matrix is ^{241}Am.

The LAAS is a neutron assay system developed at Los Alamos. It is actually two assay systems built in one housing that can perform both the passive and active neutron measurements, thus providing information about all three categories of TRU isotopes. The active measurement determines the amount of Category 1 isotopes. These isotopes (e.g., ^{239}Pu, ^{241}Pu, ^{235}U) are also called fissile isotopes, because thermal neutrons cause them to fission.

The passive neutron measurement determines the amounts of both the Category 2 and 3 isotopes. Spontaneous fission (Category 2) isotopes produce bursts of neutrons at an average rate of 2 to 3 neutrons per fission. On the other hand, Category 3 isotopes don't emit fission neutrons, but they do produce neutrons by the action of alpha particles interacting with matrix materials such as oxygen. These "alpha-neutron," or simply (α,n) events are very different in nature from the spontaneous fission neutrons. The (α,n) neutrons are produced randomly, one at a time, rather than in multiples. If one then does a "time correlation" (also called neutron time coincidence or simply neutron coincidence) measurement of the passive neutrons, one can separate out the time-correlated emissions from random emissions and thus arrive at values for each of the two categories of passive-neutron-emitting TRU isotopes.

TABLE I

ATTRIBUTE MEASUREMENTS IN WASTE USING NEUTRONS

Category	Attribute	Typical Isotopes	Measurement Techniques
1	Fissile	^{239}Pu, ^{235}U	Neutron interrogation, differential dieaway
2	Spontaneous fission	^{240}Pu	Passive neutrons, time correlated
3	Nonfission (alpha reactions)	(α,n)	Passive neutrons, random emission

II. OVERVIEW OF SYSTEM OPERATION

In this section, we will discuss briefly the LAAS and its operation. The LAAS consists of four major subsystems.

1. The assay chamber, which includes the graphite and polyethylene neutron moderating materials, the ^{3}He proportional counters that are located in the assay chamber walls, the polyethylene and borated polyethylene shielding materials, the barrel rotation assembly, etc.

2. The counting and digital electronics (preamps are located in the assay chamber walls, amplifiers, and all digital electronics are located at the control console).

3. The data acquisition system including peripherals, which is located at the control console.

4. The neutron generator and its supporting equipment. The "head" of the neutron generator unit is located in the assay chamber; part of the support equipment resides on top of the assay chamber, and the rest, including controls, is at the control console.

The LAAS has been designed to perform both passive and active assay measurements with high counting efficiency and with good sensitivity. A complete assay consists of an active measurement and a passive measurement in sequence. First, a barrel is loaded into the assay chamber and the barrel rotation unit is started. (A waste crate is loaded for the crate size systems.) The active measurement is started by turning on the neutron generator control unit and the power supplies. The operating program, called SNEUT, is readied and the data-taking sequence initiated. (We will discuss the details of performing these operations in Section IV.)

After the active count is completed (in usually less than 1 minute), SNEUT automatically starts the passive run. The operator can vary the length of the passive run by how he sets up the program during the initial SNEUT/operator interaction. Typical count times will be about 400 seconds for the barrel counter units and 1000 seconds for the large crate units.

After the passive run is completed, the data will be recorded automatically on a floppy disk. At the same time, a summary of the fully analyzed results from both the passive and active measurements will be printed. In a functioning installation, the operator can transfer these results to the facility data management system via the automatic data transmittal that is available with the LAAS LeCroy 3500 data acquisition system. The operator will now have in his possession the analyzed record indicating whether the barrel (or crate) is non-TRU (less than 100 nCi/g) or TRU (greater than 100 nCi/g). Note that this constitutes an attribute determination. The operator will also have information as to whether or not the barrel (or crate) has passed the WIPP waste acceptance criteria (WAC) for criticality safety and heat generation. This assay will ensure that only bonafide TRU is sent to WIPP (i.e., only waste that is more than 100 nCi/g).

TABLE II

SELECTED WIPP ACCEPTANCE CRITERIA

Fissile content	less than 200 g (55-gal drum) plutonium equivalent
Total TRU isotope inventory	greater than 100 nCi/g
Thermal power	less than 0.1 W/ft^3

The operator now can remove the barrel (or crate) from the assay chamber, take whatever action is appropriate, and load in another barrel (or crate) for assay.

III. DESCRIPTION OF HARDWARE

The LAAS 55-gal barrel-size assay chambers have a sliding door that opens either vertically (Fig. 1) or horizontally (Fig. 2). The preamps are located in accessible positions in the side walls and in the top and bottom. Figures 3a and 3b show cross sections through the assay chamber and the configuration of graphite, polyethylene, and detectors. The cadmium-shielded tubes are used for the active measurements, and both the bare and cadmium-shielded tubes are used for the passive measurements.

Figure 3b shows the cross section through the vertical tubes. As can be seen, each side of the system has one shielded package of ^{3}He detectors (three tubes each). There are also individual shielded packages for the top and bottom of the system, for a total of six shielded packages. Each of these is serviced by a single preamp, amplifier, and discriminator. The same is done with each

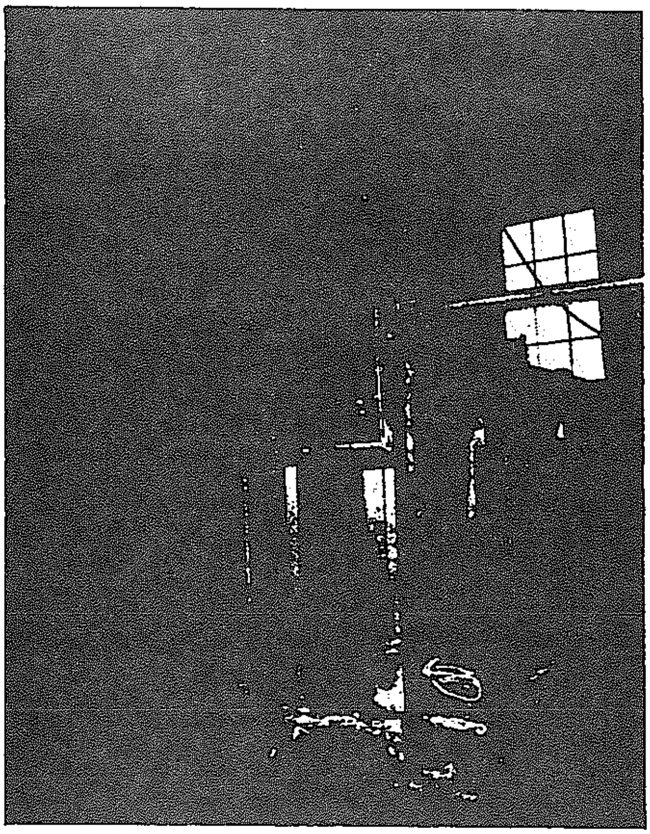

Fig. 1.
LAAS 55-gal. drum size system with vertical sliding door. Two of these assay systems were implemented late in 1984--one at the Stored Waste Examination Pilot Plant (SWEPP) at the Idaho National Engineering Laboratory in Idaho Falls, Idaho and one at Savannah River Plant in Aiken, South Carolina.

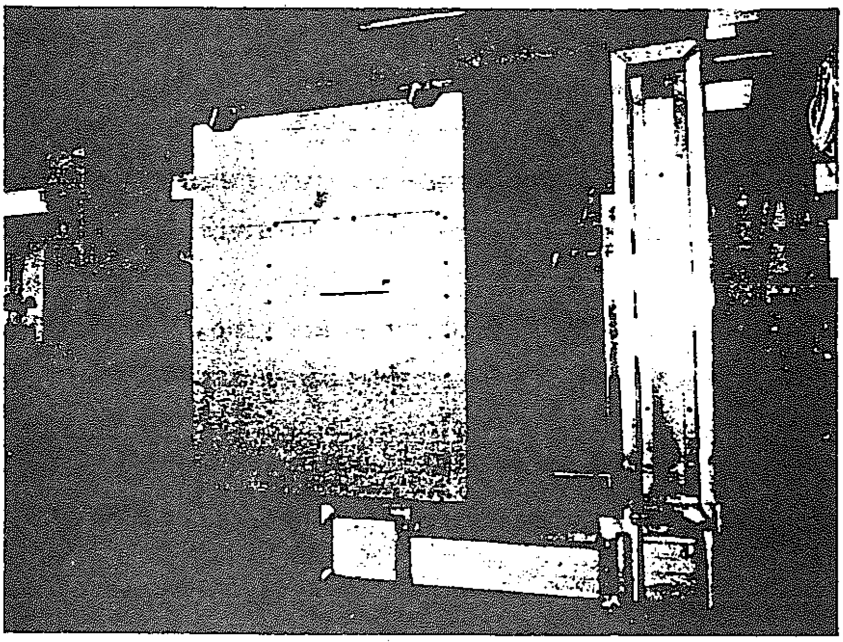

Fig. 2.
LAAS 55-gal. drum size system under construction. This unit will have a horizontal sliding door and a semi-automatic drum-loading system. It is scheduled for implementation at Hanford early in 1985.

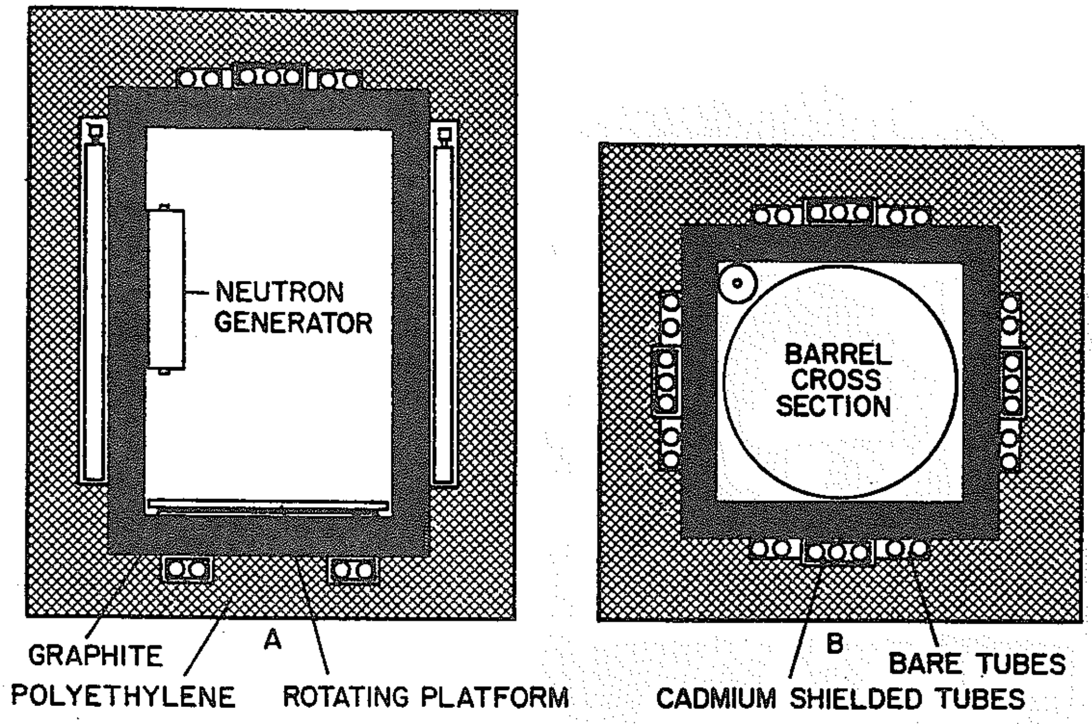

Fig. 3.
A typical disposition of ^3He proportional counters, neutron generator, and materials in a 55-gal. size LAAS. There are two types of detection packages: cadmium shielded (active) and bare (passive).

of the "bare" tube packages (four tubes in each bare package). In addition (see Fig. 4), each of the two flux monitors has a separate counting electronics channel. Thus, there are 14 separate sets of preamps, amplifiers, and discriminators for each assay system.

It is important to remember that each set of three or four proportional counters is monitored separately and the counts from each set can be printed out. Thus, if electrical problems are affecting the system, it is quite easy to isolate the problem if it is due to malfunction of a single tube, preamp, or amplifier. Figure 5 shows an example of the routine printout obtained during a

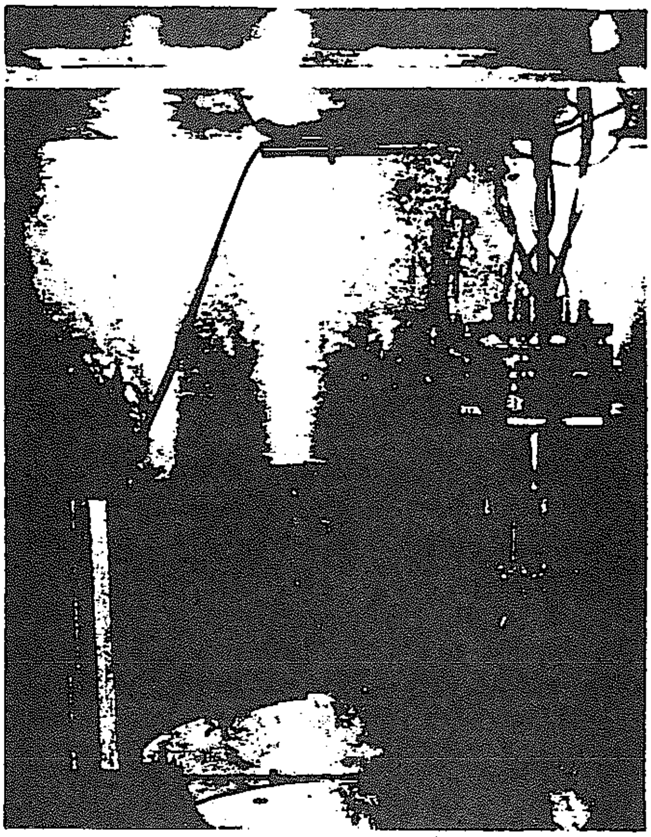

Fig. 4
Details of the typical disposition of the two flux monitors within a LAAS. The "cavity" flux monitor is at the top center of the assay chamber, and the collimated "barrel" flux monitor is in the left side corner. Also shown is the neutron generator head unit in the right side corner.

passive neutron run, illustrating the individual sets. If one of the sets is malfunctioning, the effect on count rate for a standard source or background will be apparent in this printout.

Figure 6 shows the control console electronics. In the racks on the left side of the photograph, one can see the amplifier, discriminator units, and logic pulse-conditioning units.*

The basic approach in the passive assay system is as follows: two independent passive systems are being used. The primary one consists of all the ^{3}He counters in the entire assay system (43 in the latest version of the barrel system, 66 for the Rocky Flats crate system) summed to provide what we call the system "totals" signal. The various logic and gate units provide the time correlation analysis mentioned in an earlier section.

---

*For anyone interested, there is available a more detailed description of the electronics.

```
PNEUT OF 11-11-84--EG & G IDAHO
RUN    14    DRUM 743-1696    13:47:31    11/20/93
GATE CORRECTION FACTORS  1.001480 (70 USEC GATES)   .991460 (250 USEC GATES)
FOLLOWING DATA HAS BEEN BACKGROUND CORRECTED BY BACKGROU   13:59:42  11/19/93    RUN     0

COUNTING TIME IS      425.16 SECONDS

     DETECTOR      COUNT       RATE          DETECTOR      COUNT       RATE

BARE DOOR 500      2818.       6.63      SHLD DOOR 501      497.       1.17
BARE RGHT 502      2929.       6.89      SHLD RGHT 503      535.       1.26
BARE BACK 504      2591.       6.09      SHLD BACK 505      481.       1.13
BARE LEFT 506      2813.       6.62      SHLD LEFT 507      569.       1.34
BARE TOP  508      1249.       2.94      SHLD TOP  509      223.        .52
BARE BOTH          1246.       2.93      SHLD BOTH 511      301.        .71
FLUX MONITOR        -14.       -.03      2ND FLUX MONI       18.        .04
SYSTEM TOTALS RATE        38.23       SHIELDED TOTALS RATE        6.13  (FROM PARTS)
NEUTRON COINCIDENCE

SHIELDED TOTALS                          4606.+/-        67.87
SYSTEM TOTALS                           25554.+/-       159.86
1ST N 250 USEC GATES                    24685.
1ST N 70 USEC GATES                      4557.
RANDOM 70 USEC GATES                  4251629.
RANDOM 250 USEC GATES                  425163.
1ST N GATED 70 USEC TOTALS                 48.
RANDOM GATED 70 USEC TOTALS              3258.
1ST N GATED 250 USEC TOTALS               862.
RANDOM GATED 250 USEC TOTALS             6405.

RANDOM COINCIDENT NEUTRONS/250 USEC GATE    .14936E-01
RANDOM COINCIDENT NEUTRONS/70 USEC GATE    .76743E-03

250 USEC GATE LIVE TIME          418.94 SEC
70 USEC GATE LIVE TIME        424.83 SEC
NET COINCIDENT NEUTRONS/250 USEC GATE    .19984E-01+/-    .12041E-02
NET COINCIDENT NEUTRONS/70 USEC GATE    .97658E-02+/-    .15204E-02

SYSTEM TOTALS RATE      38.228    +/-    .43911
SHIELDED TOTALS RATE    6.1294    +/-    .19117

NET COINCIDENT 250 USEC GATE NEUTRONS/LIVE TIME   .55471E-01+/-    .89168E-01
NET COINCIDENT 70 USEC GATE NEUTRONS/LIVE TIME  -.19688E-02+/-    .22817E-01


REDUCED VARIANCE
Y=   .11797E-01    Q=    .51695E-05
PASSIVE MASS FROM COINCIDENCE COUNTING
USING 70 USEC GATE    -.49221E-01 GRAMS
USING 250 USEC GATE    .10086      GRAMS
USING RED VAR Q     1.4009     GRAMS
SELECTED BY COUNT RATE     .10086      GRAMS
```

Fig. 5
The initial portion of the SNEUT passive "long" printout showing the detailed bare and shielded package count rates. The basic passive coincidence information is also shown.

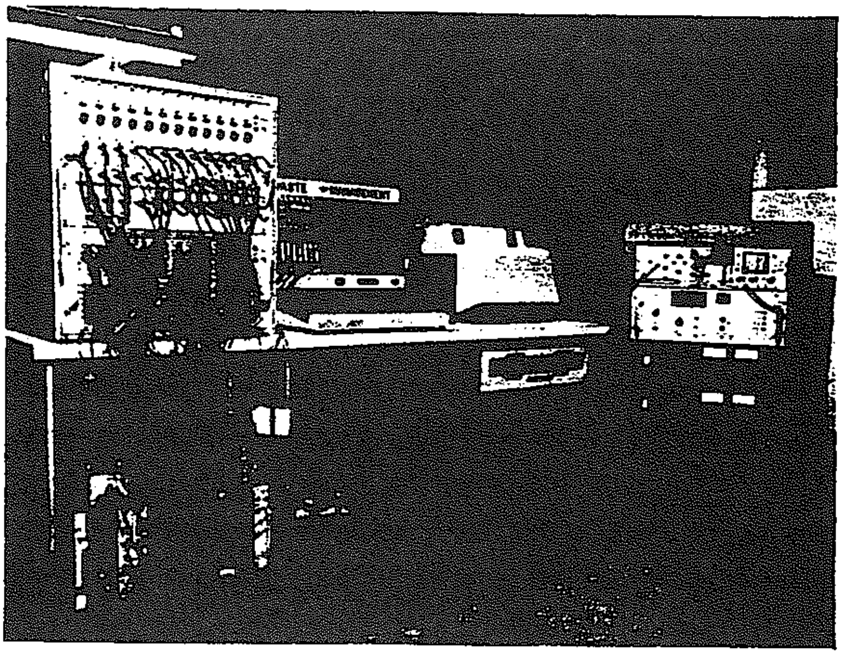

Fig. 6
A typical LAAS console electronics set up. In most current installations, the complex array of cables has been enclosed so as to minimize accidental disconnections.

The secondary passive system utilizes only the shielded totals signal that functions during the passive measurement as a secondary signal that is used for very high count rate situations when the primary system suffers from dead time and accidental coincidence effects that render it unreliable. In such situations, the less efficient shielded totals signal is actually more reliable. For the more usual low or moderate count rate situations, the system totals signal provides the most sensitive and accurate data. During the active measurements, only the shielded counts are used.

In the far right of Fig. 6, the neutron generator controller unit is shown with its monitoring scope and two high-voltage power supplies. The pulse-forming network chassis resides on top of the assay system and the Zetatron 14-MeV neutron tube and transformer housing reside inside the assay chamber.

There are several controls on the controller chassis; however, only a few are involved with routine operation. As a general simple rule to follow, turn on the equipment in the sequence top to bottom (i.e., turn the oscilloscope on first, the controller chassis second, the source high-voltage power supply (HVPS) next, and the target HVPS last). The reverse order should be followed to power down. About a 5-minute warmup time is required to first power up. If all voltages are properly set (a detailed setup and checkout procedure will be provided separately), a typical operation starts, after adequate warmup, with the RUN/SOURCE ONLY toggle switch in the SOURCE ONLY position. Then press the START SEQUENCE BUTTON. When the system is ready to produce neutrons, a characteristic scope tract will appear on the oscilloscope screen. This characteristic wave form for the generator tube ion source is called "the strike." Move the toggle switch to the run position to generate neutrons. After the data-taking run is completed, return the toggle switch to the SOURCE ONLY position.

IV. ASSAY PROCEDURE

This section will describe a step-by-step procedure for performing an assay using the LAAS. The operator will direct the entire assay from the system console using an operating program called SNEUT. SNEUT--and all the other software required to carry out assays--resides on a data floppy disk. Up to 25 records of combined passive and active assays can be written on that disk. A passive background run is also required for each disk. Let us go through this initial setup stage, including the background run.

Step 1. Insert a properly formatted disk containing SNEUT and supporting software into the A disk drive.

Step 2. Simultaneously press the HERE IS and RESET keys. The menu should appear on screen with the caret (>) on the BOOT DISK SYSTEM line.

Step 3. Press the START key. This will cause --DOS 3500 VERSION 4.0--A> to appear on the screen.

Step 4. Type in the five letters SNEUT, then press the RETURN key. This causes SNEUT to be read from the disk. From this point on, SNEUT operation is interactive; the operator chooses from option lists or answers questions. The main menu for SNEUT should now be on the screen.

Step 5. Select the option on the list by typing ST and pressing the RETURN key. You have selected INITIAL CONFIGURATION, which shows you the current list of setup and output options. You can modify these as required for your assay conditions.

Step 5a. Normally, the routine selections are:
Print option = SH (short printout)
Active assay = YE (YE = Yes)
Passive assay = YE
Passive background correction = YE
Passive count time = 400 seconds (drum system) or 1000 seconds (crate system)

Step 5b. To change an option, type the indicated letters. (For example, type PC and press RETURN to change the passive count time. When SNEUT requests the count time, type 400, 1000, or whatever count time you wish and press RETURN). After each change in these options, the current set is displayed. When you are satisfied with the current set, type NO and press RETURN . This will return you to the main menu.

Step 5c. As stated above, each disk must have a passive background run recorded. To do this, type PB and press RETURN. You will be asked to verify that the assay chamber is empty. It is important to do this, because having an accurate background is crucial to all assays. Once you answer YES to this question, the passive background starts automatically and runs for the length of time you previously selected for the passive count time.

Step 6. The main menu returns to the screen after the background run terminates. You have now finished the setup portion for a single disk. Up to 25 data runs can now be acquired with the much simpler procedure described below. To take data, type AS and press RETURN. This selects the "assay in stand-alone" mode.

Step 7. Now SNEUT asks a series of questions relating to the assay data runs.

Step 7a. The first question is RUN NUMBER? The run numbers should all follow in sequence. A total of 25 can be put on each disk. The first drum (or crate) to be assayed should have run number 1.

Step 7b. ENTER PRIMARY IDENTIFICATION. You can enter up to 15 characters of identification. This should include at least the basic drum (or crate) identification number.

Step 7c. ENTER SECONDARY IDENTIFICATION. You again enter up to 15 characters to use for additional identification, such as building of origin or date of waste packaging.

Step 7d. ENTER CONTENT CODE. Many sites, such as Rocky Flats and Idaho National Engineering Laboratory, have detailed numerical waste content codes associated with combustible wastes, miscellaneous metals waste, etc. This is the place to enter such information. Enter only the three-digit content code number, as this number may be used in the assay algorithm.

Step 7e. ENTER CONTAINER WEIGHT IN KG. It is important to obtain the container weight before the assay so that it can be entered here for permanent records and for possible use in the assay algorithm. Be careful to enter only a numerical value for the weight.

Step 7f. IS THE CONTAINER INFORMATION CORRECT? Your responses to the previous five questions are displayed for you to check and to correct if necessary. Then answer YES.

Step 8. After you answer YES to the last question, SNEUT moves on to the active assay. It displays the active assay setup information on the screen and instructs the operator to FIRE 2000 NEUTRON GENERATOR PULSES (or 5000, or whatever you entered).

Step 8a. As mentioned in Section IV, there are four separate units, all in the same rack, to be warmed up before an active pulsed neutron run can be made. Turn these on one at a time, top to bottom. For routine operation, only the bottom one, the target HVPS, need be turned off after each active run. Of course, for extended down times (e.g., overnight), all four units should be turned off. Note that all other electronics (e.g., counting electronics, LeCroy 3500) can be left on overnight and over weekends.

Step 8b. After the warmup period (5 minutes to start the day), verify the nominal voltages on the major components within the Zetatron. Monitor them with the rotary switch to the left side of the the controller chassis. Read the voltages from the LED display in the chassis. The source HV should be 500 V. The target HV should be 450 V, and the RES (reservoir) should be 4.55 to 4.58. Small departures from these nominal values are of no consequence. Large departures may require fine adjustments. These readings don't need to be checked for each run; once or twice during a day is sufficient.

Step 8c. Check that the required number of pulses is being shown on the pulse counter of the controller unit. This will normally be 2000 for the drum counter and 5000 for the crate counter.

Step 8d. Press the START SEQUENCE button on the controller chassis. When the system is ready to produce neutrons, the STRIKE wave form will appear. When it does appear, push the toggle switch up to the RUN position. Neutrons will now be produced.

Step 8e. When the active run is completed, SNEUT will automatically start the passive run. Information appears on the screen to let you know that the active run is over.

Step 9. At the end of the passive run, the data are displayed and printed out. SNEUT will now ask you, DO YOU WISH TO PERFORM ANOTHER ASSAY? An answer of YES repeats this sequence, starting at step 7. Enter run number, etc.

Step 10. Remember that only 25 runs can be written on a single disk. If this limit is reached, place a fresh formatted disk in the disk drive--don't forget to take another passive background for the new disk.

V. CALIBRATION

We recommend that calibration be done frequently using a standard container (drum or crate) filled with a matrix in which are placed, in fixed locations, a ^{252}Cf source and a standard uranium sample. This standard container should be painted some conspicuous color so it will not be confused with regular waste drums.

Assay the standard calibration container at least once every working day using the same procedure as for regular waste drums. This measurement/control procedure will provide a regular, automatic record of the system's operational status, which should remain constant if the system is operating normally. The result of the active assay on the standard should be constant day after day, with appropriate allowance for statistical errors. The passive assay will show the expected ^{252}Cf source decay with its half-life of 2.65 years. Allowing for source decay, the passive assay should also be constant day after day.

If the daily assays are not constant, they indicate a problem that will require attention. Although small changes in system sensitivity will not cause difficulties, any change is really indicative of some problem such as high-voltage power-supply drift, discriminator threshold drift, preamp or amplifier gain shift. Thus, from the point of view of long-term operational success, it is wise to find out as soon as possible what is causing the system to produce nonconstant measurement control assays. It is the nature of these neutron assay units to be very constant in performance over long-term periods, if all components are in working order and are properly adjusted.

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参考資料－4

OPERATION AND LIFE OF THE ZETATRON: A SMALL NEUTRON GENERATOR FOR BOREHOLE LOGGING

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OPERATION AND LIFE OF THE ZETATRON: A SMALL NEUTRON GENERATOR FOR BOREHOLE LOGGING

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SUMMARY

The zetatron is a small sealed-off D-T accelerator being used for pulsed neutron production in Uranium borehole logging experiments. The tube utilizes a zirconium gas reservoir, Penning type ion source, and up to 130 kV ion extraction with ion beam focussing and secondary electron supression. The mixed D-T beam is incident on a scandium film target and produces in the pulse mode up to 8×10^6 neutrons/sec peak rate at a source current of 2.5 A and target voltage of 130 kV. In the borehole logging application the tube, operating at 100 pps, dissipates an average power of about 10 watts. Life testing has demonstrated an average tube functional life of 148 hours and a shelf life in excess of one year. Life limitations include a steady degradation in neutron output of about 50% in 100 hours and eventual tube puncture fractures that are catastrophic.

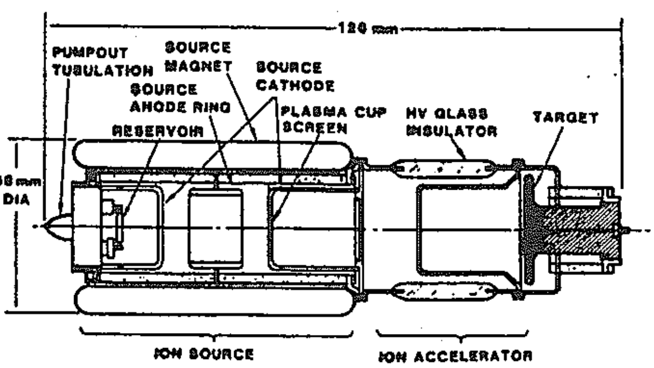
INTRODUCTION

The predecessor of the Zetatron neutron generating tube came into being at the request of NASA for a small neutron source for use in activation analysis. The tube continued its life in a second version as a highly repeatable neutron source for calibrating neutron detectors[2] and measuring neutron attenuation in materials. The most recent version has found application in uranium borehole logging[3,4,5], portal security monitoring[6], and transuranic assaying[7]. Its physical size (Fig. 1) is 38 mm diameter with source magnet in place by 126 mm long. It is operated in a pulse mode on both the ion source and the accelerating voltage. The tube utilizes the D-T reaction, accelerating a mixed beam from the ion source onto an occluded gas target, also of a D-T mixture. Details of the tube operation, life experience and application follow. Some life-limiting mechanisms are discussed.

TUBE DESCRIPTION

The tube is shown in cross-section in fig. 1. The gas reservoir is mounted on one end and consists of a porous steel tube, 0.9 mm diameter, filled with 20 mg of powdered zirconium that has been hydrided with a 50/50 mixture of deuterium and tritium. Passing a current of 3.8 $\pm$0.1 A through the reservoir will provide the heat necessary to release the D-T gas and reach the tube's operating pressure of approximately 2 Pa.

The ion source of metal/ceramic construction has a "Penning" geometry operating in a reflex discharge mode using a permanent magnet to provide a 600 gauss axial field. In present applications, the source is pulsed with a 2500 Volt, 20 µs pulse through an impedance that provides 1.5 to 2.5 A source current. The 2500 volt amplitude provides a rapid striking of the source arc (< 5 µs) which then is maintained with a voltage drop of approximately 400 Volts. The generated ions are extracted from the secondary cathode by the pulsed electric field formed by the cylindrical accelerator and accelerated across the gap to strike the target at typically 120 keV energy. The target consists of a 5 micron thick, 20 mm diameter, scandium deutero-tritide film deposited on a molybdenum



ZETATRON

Fig. 1 The Zetatron tube in cross-section with tube structures and functions identified.

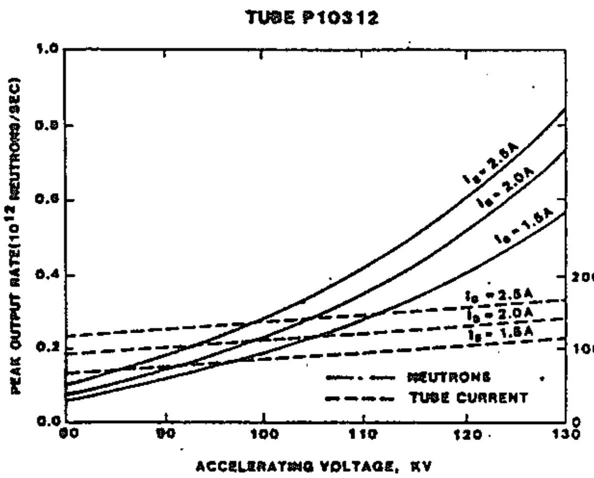


Fig. 2 Typical Zetatron characteristic curves of peak neutron output rate and peak tube currents. data taken at 10 kV increments of accelerating voltage.

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substrate and is electrically insulated from the accelerator permitting the target to be biased to suppress secondary electrons.

Typical characteristics of tube current and neutron output as a function of accelerating voltage and source current are shown in fig. 2. As the D-T reaction predicts, the output is strongly dependent upon the accelerating voltage. Across the 80 to 130 kV range the output varies nearly as the fourth power of the voltage whereas the tube current is linearly dependent. Both neutron output and tube current are linearly dependent on source current.

LIFE EXPERIENCE

Table I summarizes the life test results of nine tubes that were life tested operating at a pulse rate of 100 pps with a peak source current of 2 A and peak accelerating voltage of 120 kV. This life testing is conducted during laboratory off-hours overnight and weekends under computer control. A single shot neutron output is sampled every 24 seconds, and if it meets minimum requirements the test continues. Every 16 minutes 5 pertinent tube waveforms including neutron output are digitized, and an example of that output is displayed in fig. 3. The neutron output is measured using a lead activation/scintillator system[9]. Although one tube failed in 22 hours, which we believe was due to overpressure operations, the average life for the nine tubes was 148 hours. One tube reached 350 hours and was still operating with an output of 1.5×10^6 neutrons/pulse. Neutron output degrades slowly to approximately 50% in 100 hours as is shown in the typical life test results in fig. 3. Catastrophic end of life was most frequently a puncture fracture of the glass wall at the target end of the tube, five of the nine tubes having failed in that mode. Of the remaining four tubes, three were terminated for postmortem examination and one appeared to suffer from a contaminated gas reservoir. It should be mentioned that no tritium has been detected external to the tube nor in the tester dielectric fluid as a result of the tube punctures.

Also in our personal experience with many zetatron tubes — no external tritium has been observed.

TUBE ID	RUN TIME (HRS)	REASON FOR TERMINATION
P10067	100	POSTMORTEM
P10173	168	POSTMORTEM
P10212	210	GLASS PUNCTURE
P10213	22	GLASS PUNCTURE
P10246	73.8	GLASS PUNCTURE
P10288	101.3	GLASS PUNCTURE
K103	350	POSTMORTEM
K109	142.5	GLASS PUNCTURE
K110	164.5	RESERVOIR FAIL.

AVER. 148

$\therefore$ # pulses = (148)(3600)(100) = 53.3×10^6
hrs. sec/hr pps

TABLE I. Summary of Zetatron life test results. Prefix P tubes were fabricated by integrated contract with General Electric Neutron Devices Dep't. Prefix K tubes were fabricated under a transfer of Technology contract with Kaman Sciences Corp.

APPLICATION IN BOREHOLE LOGGING

A typical application for this tube is its use as the neutron generator in borehole logging in uranium exploration[3,4,5]. To this end, the goal for the tube and its driving circuit components was 50 hours minimum life at a minimum output of 10^6 neutrons/pulse and pulse rate of 100 pps. As can be seen from the life test results (Fig.3), the output rate starts at about

OUTPUT VS. RUN TIME
TUBE K109

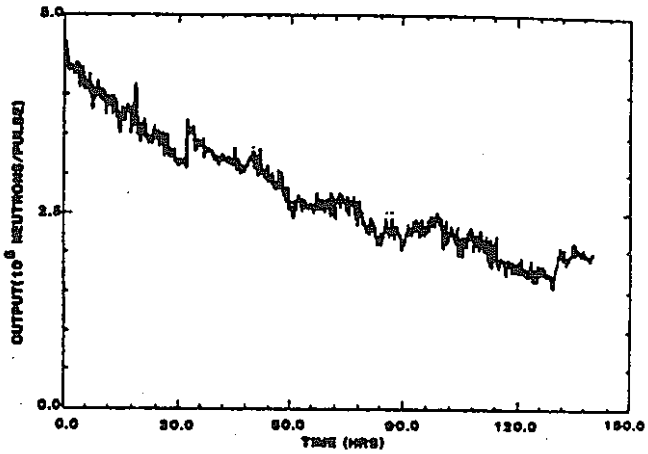

Fig. 3 Typical life test record of a Zetatron tube.

4.5×10^6 neutrons/pulse and remains well above the desired minimum through its life. The short term (4 hours) shot-to-shot deviation computes to be less than 5% which is also considered acceptable.

In logging applications with the circuitry used by this laboratory the power dissipated in the tube at a 100 pps repetition rate is 2.6 watts in heating the gas reservoir, 2.0 watts in the ion source, and 5.4 watts in accelerating the ions and electrons. To minimize the power requirements of the high voltage section we have attempted to reduce the tube current by biasing the target of the tube using resistors between the target and accelerator. Fig. 4 displays the tube current dependency on the bias resistance as the resistance is increased up to 100 Kohms. Total tube current is reduced by about a factor of 2.4. Further increase in the size of the bias resistor yields only a slight additional reduction. We have found that the tube performs satisfactorily with the target floating, implying a self-limiting bias action such as a gas discharge is in effect in the target region.

This tube has been assembled with a current limiting resistor and high voltage pulse transformer and overcast with a modified epoxy resin into a tube-transformer-assembly (TTA). Five such assemblies have been tested in the same life tester and mode used to life test tubes. Table II summarizes the test results to date. There were two discouragingly short run times, namely, on TTA's 630 and 4206. But both were identifiable as solvable process problems, not design faults. In no case was the tube the cause of failure. In fact the tube, in such instances of failure, is recoverable and can be re-used.

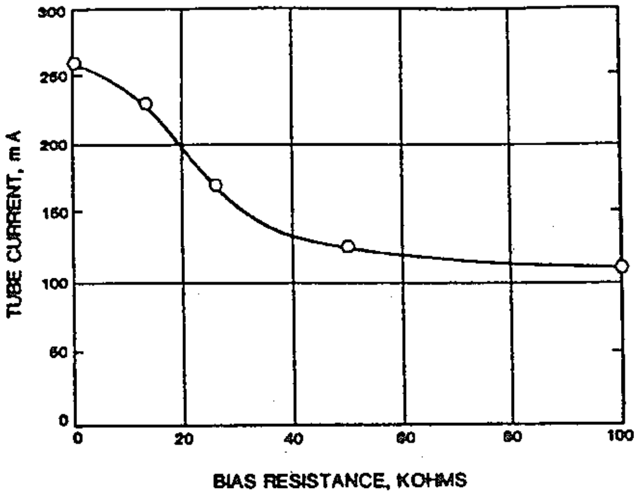

Fig. 4 Tube current as a function of target bias resistance.

TTA ID	RUN TIME (HRS)	REASON FOR TERMINATION
626	58.2	POTTING PUNCTURE
629	140.	POTTING PUNCTURE
630	8.	XFMR FAILURE
4206	0.2	XFMR FAILURE
4207	45.9	STILL RUNNING

TABLE II. Summary of TTA life test results.

LIFE LIMITATIONS

Several life limiting factors were of concern to us including helium generation, reservoir capacity, target degradation, and metal sputtering. The following paragraphs describe our life testing experience in these areas.

Helium Generation. With approximately 800 μgm of tritium in the tube at birth it was thought that ^{3}He generated by radioactive decay would be a problem in that the tube would reach 17 Pa in about 7 days assuming 100% of the ^{3}He was released. We have seen no evidence of this in tubes over one-year old indicating that a very large percentage of the ^{3}He is not released. The tube that had run for 350 hours was opened for postmortem analysis and only 0.5 μgm of the 70 μgm of ^{3}He generated, less than 1%, was found as free gas. The balance of the gas was found to still be in the reservoir and target. This is clear evidence that the ^{3}He will not be released from the reservoir at a significant rate even with repeated temperature cycling experienced in normal operations. Life limitation due to free ^{3}He buildup does not, therefore, appear to be a problem in times less than one year. We are accumulating tubes with greater age and conducting experiments with tubes backfilled with ^{3}He to provide a basis for life projections beyond our one year experience.

Reservoir Capacity. Reservoir characteristics on all tubes with the exception of the life tested tube, K110, have remained stable throughout their shot lives and shelf lives up to 18 months. That one exception, K110, shows evidence of the reservoir being contaminated in that it has become very slow in re-gettering evolved gas but it does not appear to have insufficient capacity. We, therefore, conclude that the capacity is more than adequate and are developing smaller capacity reservoirs to reduce the tritium inventory.

Target Degradation. Postmortems of life tested tubes indicated that a circle approximately 3 mm in diameter in the center of the target is completely depleted of deuterium-tritium. The depletion of the target is believed to be due to sputtering and implantation of impurity ions and is most likely the major contributor to the steady degradation in output. Beam modeling[10] of this tube indicates this area of the target receives a very large percentage of the beam and implies that a lower degradation rate would be realized with a less strongly focussed beam. Tube redesigns are being pursued to verify the theory and improve tube performance.

Metal sputtering. The postmortems showed considerable material deposition on the glass walls of the accelerator section. It is believed that this deposition leads to enhanced surface breakdowns which induce glass damage and eventual tube puncture, the major life limiting factor in life testing.

CONCLUSION

The Zetatron tube has been demonstrated as a neutron generating tube small enough to be used in a uranium borehole logging application with a high neutron output, low shot-to-shot deviation, and an adequately long shot life. Shelf life does not appear to be less than one year.

ACKNOWLEDGEMENTS

We would like to gratefully acknowledge the contribution of R.D.Volk of this laboratory for his direction in the early development phase of this project and the assistance of R.A.Kubach of General Electric Neutron Devices Department in resolving assembly and processing problems.

The uranium borehole logging program is sponsored by the Department of Energy, Office of Uranium Resources and Enrichment. As in most programs of this type a transfer of technology to private industry is envisioned. Part of that technology transfer involved developing a competent vendor for the Zetatron. After competitive bidding a contract was awarded to Kaman Scientific Corporation of Colorado Springs, Colorado, for the fabrication of Zetatrons, pulse transformers, and Zetatron/transformer assemblies. Some of the results on Kaman-made Zetatrons was reported in this paper. Other details of the technology transfer will be reported at a seminar tentatively scheduled for mid-1981.

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PAJARITO SCIENTIFIC CORPORATION

INFORMAL REPORT PSC-88Ø11

CALCULATION AND MEASUREMENT OF PULSED 14 MEV NEUTRON GENERATOR NEUTRON AND GAMMA RAY DOSE RATES

By Dr. John T. Caldwell

September 23, 1988

I. Introduction

The calculation and measurement of pulsed 14 Mev neutron generator fast neutron and gamma ray dose rates require proper modeling to account for the pulsed nature of the observed radiation on exterior surfaces. For instance, to obtain the biologically equivalent fast neutron dose rate produced by a pulsed 14 Mev source in units of millirem per hr, one must first determine the effective pulsing rate in pulses per hour of operation. In the case of the PNC Tokai passive-active neutron assay unit the PNC/JGC system specifications call for an assay rate of one per 10 minutes, or 6 assays per hour. Assuming the system as delivered by Pajarito Scientific Corporation will meet these specifications, and that the USDOE system standard of 2000 pulses per unit assay is utilized by PNC Tokai in their operations, one can calculate the pulsing rate to be 12000 per hr during active assay operations. (Of course, when the assay system is not in use, the pulsing rate will be zero. Thus, the overall weekly, monthly or yearly average dose rates will be less than the active assay dose rates, in proportion to the active assay time divided by the total time.)

The specifications of the zetatron pulsed neutron generator system call for a nominal output of one million (1x 10E6) 14 Mev neutrons per pulse, with a nominal pulsing rate of 50 pps utilized until 2000 pulses have been accumulated. Thus, the calculated hourly pulsing rate during active operations results in a neutron source of 12000x1,000,000= 1.2x10E10 n/hr. The chatacteristic angular distribution of these 14 Mev D+T neutrons is isotropic to better than +/- 5%. That is, all source neutron are emitted in equal amounts in the direction of all solid angles. The PNC Tokai unit assay chamber door is approximately 100 cm from the internally located 14 Mev source. For the purposes of this calculation it will first be assumed that the door is open so that a <u>maximum possible</u> dose rate may be calculated in order to evaluate the potential health hazard in the unlikely event safety interlock systems fail and the generator fires with the door open.

Table I shows the neutron dose equivalent for a range of neutron energies, taken from the radiation standard handbook, NBS No. 63, 1988 edition, p. 130. In this case a total fluence of 1.4x10E6 n/cm2 of 14 Mev neutrons are required to produce a dose of 0.1 rem (100 millirem). In our calculation for 100 cm separation the calculated hourly fluence is 1.2x10E10/(4x3.14x10,000)= 9.55xE4 n/cm2/hr. Scaling by the data in NBS No.63 results in a calculated dose rate of (9.55x10E4/1.4x10E6)x100= 6.8 millirem/hr 100 cm from the unshielded source, at the assay system door location.

Since it is proposed to exclude personal from room P-303 by an interlock in the room door closure mechanism, it is important to calculate the dose rate at the room door location also. This is approximately 400 cm from the source. In this case the calculated dose rate is 0.68x(100)2/(400)2=0.43 millirem/hr.

In the actual operational situation, of course, approximately 35 cm of polyethylene and graphite shielding is interposed between the 14 Mev neutron source and the calculational points at 100 cm and 400 cm. Standard fast neutron shielding calculations (ex, NBS No. 63) indicate this shielding wall produces considerable 14 Mev neutron attenuation and associated lowering of external fast neutron dose rates. The tables in NBS No. 63 indicate this will be about a factor of 3. The actual dose rates have been measured in several USDOE passive-active neutron assay systems utilizing zetatron 14 Mev systems. These measurements verify the calculations presented here. For example, the Savannah River Plant NDA system, during acceptance tests in 1985, was measured with standard TLD fast neutron dosimeters by the SRP radiation health physics department. These measurements indicated a 35 millirem fast neutron dose produced on the external assay chamber walls by 180,000 zetatron pulses. Scaling this measurement to the PNC Tokai system operations of 12000 pulses per hr results in a measured dose rate of approximately 35x(12000/180000)=2.3 millirem/hr on the external wall surface, or about a shielding factor of 3 (unshielded dose rate calculated to be 6.8 mr/hr.)

Thus, using the measured and calculated factor of three dose rate attenuation due to the assay chamber walls, one obtains a final external assay chamber wall (100 cm) dose rate of 2.3 millirem/hr. and the corresponding room P-303 door (400 cm) fast neutron dose rate of .14 millirem/hr.

The accompanying gamma ray dose rate may be estimated by assuming each original 14 Mev neutron will produce about 1 Mev of prompt gamma radiation through inelastic collisions with drum and wall materials. (Most of the original 14 Mev of kinetic energy is lost in radiationless elastic collisions, primarily with H atoms.) Thus, the assay chamber door (100 cm) unshielded gamma ray source, assuming 1 Mev gamma rays on the average, will be 1.2x10E10 gamma rays per hr. Using NBS No. 63 conversion tables, this results in a calculated 1 Mev gamma ray dose rate of .09 millirem/hr. The corresponding calculated dose rate at the room door loacation (400 cm) is .0056 millirem/hr. The 35 cm of assay chamber wall shielding results in even lower gamma ray dose rates being observed.

Finally, it is possible to estimate fast neutron dose rates produced by fission of Pu-239 in waste drums being measured in the assay system. From standard source measurements in the USDOE assay systems one determines that the standard 2000 pulse assay of 1 gm of Pu-239 produces 1.8x10E5 fast fission neutrons. Assuming the worst case scenario for the PNC operations of 6 drums in one hour, each containing 30 gm Pu-239--one calculates (using NBS No. 63) a fission spectrum (2 Mev average neutron energy) dose rate unshielded at 100 cm of .0003 millirem/hr, or negligible compared to the 14 Mev neutron dose rate.

TABLE I, NEUTRON EQUIVALENT DOSE FACTORS

2. NEUTRON DOSE EQUIVALENT (From NBS No. 63)

Neutron Energy (MeV)	0.1 rem (n/cm^2)	0.1 rem/40 h (n/cm^2 per sec)
Thermal	96x10^6	670
0.0001	72x10^6	500
0.005	82x10^6	570
0.02	40x10^6	280
0.1	12x10^6	80
0.5	4.3x10^6	30
1.0	2.6x10^6	18
2.5	2.9x10^6	20
5.0	2.6x10^6	18
7.5	2.4x10^6	17
10	2.4x10^6	17
10 to 30	1.4x10^6 [a]	10[a]

[a]Later data suggest that these values are applicable up to 400 MeV.

3. ISOTOPES WITH LOWEST MPC'S IN AIR IN ORDER OF MPC (Basis, 40-h/week worker exposure) Source, AEC Manual Chapter 0524.

Alpha Emitters

MPC (μCi/cm^3)	Isotopes
6 x 10^{-13}	^{248}Cm
1 x 10^{-12}	^{231}Pa
2 x 10^{-12}	^{249}Cf
	^{251}Cf
	^{238}Pu
	^{239}Pu
	^{240}Pu
	^{242}Pu
	^{244}Pu
	^{230}Th
4 x 10^{-12}	^{237}Np
5 x 10^{-12}	^{250}Cf
	^{254}Cf
	^{245}Cm
	^{246}Cm
	^{247}Cm
6 x 10^{-12}	^{241}Am
	^{243}Am
	^{243}Cm
	^{228}Th
9 x 10^{-12}	^{244}Cm
2 x 10^{-11}	^{252}Cf
	^{254}Es
3 x 10^{-11}	^{226}Ra
	^{232}Th
	Th-nat[c]
	^{232}U
6 x 10^{-11}	U-nat[c]

Beta Emitters[a]

MPC (μCi/cm^3)	Isotopes
2 x 10^{-12}	^{227}Ac[b]
6 x 10^{-12}	^{242m}Am[b]
4 x 10^{-11}	^{228}Ra[b]
9 x 10^{-11}	^{241}Pu[b]
1 x 10^{-10}	^{210}Pb[b]
4 x 10^{-10}	^{255}Es[b]
8 x 10^{-10}	^{253}Cf[b]
	^{230}Pa[b]
9 x 10^{-10}	^{249}Bk[b]
1 x 10^{-9}	^{90}Sr
2 x 10^{-9}	^{129}I
4 x 10^{-9}	^{154}Eu
5 x 10^{-9}	^{254m}Es[b]
	^{125}I
	^{59}Fe
6 x 10^{-9}	^{210}Bi[b]
	^{144}Ce
	^{106}Ru
8 x 10^{-9}	^{126}I
9 x 10^{-9}	^{60}Co
	^{131}I
	^{22}Na

See footnotes at end of table.

See footnotes at end of table.

reuter stokes

He-3 PROPORTIONAL COUNTER

SPECIFICATIONS

REUTER-STOKES MODEL NUMBER P4-1624-205

MECHANICAL

Maximum Diameter	2.03 in.
Maximum Overall Length	27.56 in.
Maximum Sensitive Length	24.00 in.
Connector Type	HN

MATERIALS

Body	304 SS
Detector Insulation	Al_2O_3
Connector	Brass, Silver Plated.
Connector Insulation	Teflon
Fill Gas	He-3 - CO_2
Fill Pressure	30 psia

MAXIMUM RATINGS

Voltage	3500 V
Temperature (See Note 1)	100°C

OPERATING CHARACTERISTICS

Environmental

Operating Temperature Range	-40 to +100°C
Humidity (non-condensing)	0 - 100%

Nuclear

Thermal Neutron Flux Range (See Note 2)	to 500 nv
Thermal Neutron Sensitivity	147 cps/nv ± 5%
Background	< 1 cps

Electrical

Resistance	> 10^{12} Ω @ 25°C
Capacitance	10 pf
Operating Voltage Range	900 to 1500 VDC

OPTIMA 860
POWERED CAMAC CRATE

dsp Technology Inc.

The Optima 860 has set the industry standard for powered CAMAC crates.

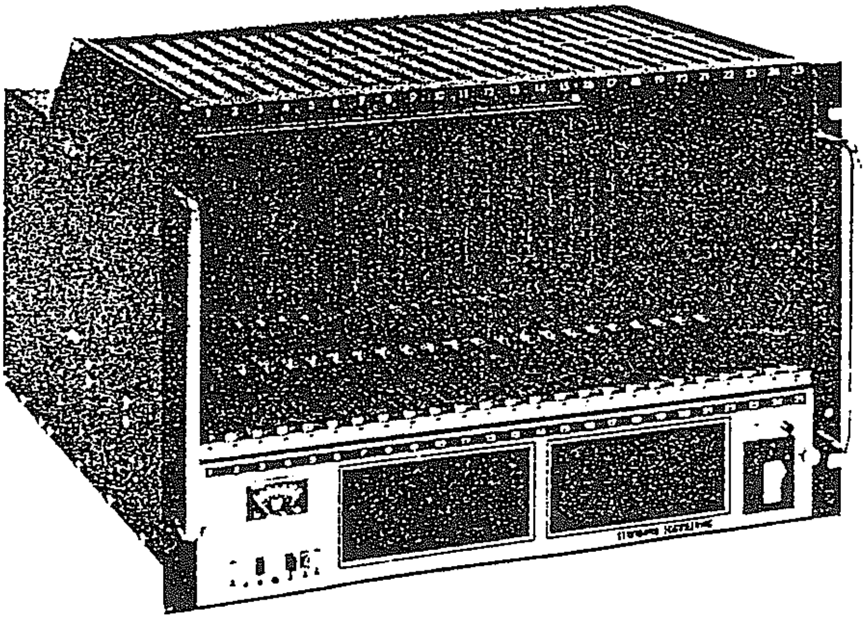
The Optima 860 continues to be the world's largest selling powered CAMAC crate based on quality, reliability and performance at a very competitive price.

Power Features:

- ±6V 60 Amps shared current (to 50 A max on either +6V or -6V)
- ±24V 9 Amps shared current (to a 6A max on either +24V or -24V)
- ±12V to 3 Amps on either +12V or -12V (current shared with ± 24V)
- 550 Watts total output power

General Features:

- Reinforced steel crate construction with cast aluminum, non-galling, electroless nickel-plated module guide rails for long and rugged life.
- Modular power supply has demonstrated superior reliability and performance by 21,000 hour mean-time-between-failure in field experience.
- Integral support shelf and quick release mechanism allows no-tool, single-handed power supply removal.
- Slide-out three-fan blower tray for module ventilation features low-noise, venturi-type, ball-bearing fans (over 400 CFM rating) combined with large air intake openings provides unmatched standard cooling capacity for enhanced module performance and maximum life.
- Convenient front panel metering and test jacks for output voltages and currents.
- Provides thorough protection on all circuits and outputs:
  - Overvoltage Protection
  - Overcurrent Protection
  - Thermal Protection



- Exceeds all CAMAC (IEEE 583-1982) and ESONE report, EUR 4100e, Type CP-1 power supply requirements for:
  - Regulation
  - Ripple and Noise
  - Long-Term Stability

Description:

The Optima 860 CAMAC (IEEE-583, Computer Automated Measurement and Control) powered crate is an assembly made up of three basic units: the crate, the power supply, and the blower drawer.

The blower section and power supply can be removed without disturbing or removing the crate from the rack, or dismantling cabling from the modules.

DSP Technology, Inc. 48500 Kato Road, Fremont, CA 94538-7338 (415) 657-7555 TLX 283608

Description (cont.)

The crate houses the removable blower tray, supports the power supply, and incorporates the Dataway bus. The crate assembly is constructed of reinforced steel and precision-cast aluminum upper and lower guides for long, rugged, trouble-free use. Precision-cast and machined, full-depth, self-centering module insertion guide rails insure smooth insertion, removal, and the proper alignment of CAMAC modules with their Dataway connectors.

Only the highest quality components go into the modular construction of SEC's power supplies. Rigorous quality assurance procedures and standards are followed throughout the manufacturing process including a 48-hour burn-in and final test under load conditions. The power supply has a 21,000 hour MTBF (mean-time-between-failure) in field performance. Additional features include "at-a-glance" fuse visibility and a separate cooling system. Further, the power supply's modular construction assures ease of maintenance and minimum downtime should repair become necessary. All outputs of the power supply are protected from overvoltage and overcurrent.No damage to the power supply results if there is a continuous short-circuit. A thermal warning and a thermal cutoff switch protect the power supply whenever the internal temperature exceeds a safe limit.

The slide-out fan tray features three low-vibration, low noise, venturi-type ball-bearing fans rated at 134 CFM each.

- Large air-intake opening
- High-adhesion dust filters easily removable without tools
- Front metering and test jacks for voltage and current
- Crate power switch

TECHNICAL SPECIFICATIONS:

Mechanical:

- Width: 19.0 in. (rack mount)
- Height: 12.3 in.
- Depth: 21.6 in. behind front panel (22.4 in. overall)
- Weight: 88 lb. (shipping weight: 101 lb.)

Electrical:

- Input: 115V +10%/ -12% Vac, 50 - 60 Hz nominal (standard)
- Output voltage and current:
  ±6V at 0A to 50A,
  current shared to 60A
  ±24V at 0A to 6A,
  current shared to 9A
  ±12V at 0A to 3A,
  current shared to 24V output
- Output Power: 550W at 25°C derated to 500W at 50°C

NOTE:

OUTPUT POWER is the sum of two groups of output voltages multiplied by their respective load currents. The sum of the two groups must not exceed 550W at 25°C. and must be derated to 500W at 50°C.

GROUP 1 OUTPUTS: +6V, +24V, and ±12V combined output power must not exceed 400W at 25°C. The ±12V output currents are considered +24V outputs for power calculations.

GROUP 2 OUTPUTS: - 6V and - 24V combined output power must not exceed 400W at 25°C and 375W at 50°C.

- Overvoltage Protection: All output voltages are protected by silicon control rectifiers (SCR) that can be activated by ac line transients, large load-current changes, or internal circuit failure.
- Overcurrent Short Circuit Protection: The outputs of the power supply are protected against over-current and short circuits by fold-back type current-limiting circuits which cause the output current to be reduced to a low, safe value

- Thermal Protection:
  Two thermal switches are mounted on each of the two heat sinks. One thermal switch on each heat sink disconnects the main power; one turns on the overload indicator on the front panel of the blower.
- Metering: ±6V and ±24V and current
- Test Jacks: All output voltages
- Line and Load Regulation:
  ±0.2% for ±6V and ±24V outputs
  2.0% for ±12V output
- Ripple and Noise: 10mV peak to peak, 50MHz bandwidth, 50mV on ±12V outputs
- Transient Recovery: 50 microseconds, typical
- Long-term Stability: Less than 0.2% for 24 hr. after 1 hr. warm-up (constant ambient temperature)
- Temperature Coefficient: 0.02%/°C maximum

Environmental

- Operating Temperature: 0°C to 50°C (32°F to 122°F) at less than 90% relative humidity

HOW TO SPECIFY THE OPTIMA 860

ORDER BY MODEL: OPTIMA 860 plus eight-digit number which includes options:

8 2 0 _ _ 2 1 _

First blank:

1 — Standard CAMAC construction
3 — NIM holes — Jacking screw holes for NIM modules in both upper and lower guide grills.
5 — Viking blocks (set of 12) blocks accommodate 36-pin Viking card-edge connectors (REQUIRED WITH SEC's QUANTROL E-200 SERIES ADC MODULES)
6 — NIM holes and Viking blocks

Second blank:

5 — 115 Vac, 50-60Hz (standard)
6 — 230 Vac, 50Hz
8 — 100 Vac, 50Hz

Last blank:

1 — Standard (NO STATUS BIT)
2 — Status Monitor Bit: Optional monitoring of all outputs and thermal overloads. Provides single relay contact closure on exceedences.

OPTIMA Wired Crate only: P/N 82860001
OPTIMA Crate Power Supply only (PS860/0) P/N 82005200

Polyolefin Comparison

10 Rating = Excellent

Material	Abrasion Resistance	Chemical Resistance	Impact Resistance 73° F	Impact Resistance 0°F	Coefficient of Friction	Temperature Limits	Thermal Expansion	Ease of Fabrication	Water Absorption	Load	Formability	Stiffness
High Density Polyethylene (HDPE)	6	6	8	8	7	7	6	8	10+	6	9	7
Low Density Polyethylene (LDPE) Homopolymer	3	7	9	8	6	5	6	8	10+	5	9	3
Ultra High Molecular Weight Polyethylene (UHMWPE)	10+	8	10+	10+	10+	7	7	4	10+	7	3	7
Polypropylene Homopolymer	4	10+	5	5	7	8	8	8	10+	9	8	9
Polypropylene Copolymer	5	9	8	8	7	5	8	8	10+	7	9	9

This guide rates one polyolefin against another to indicate relative strengths. The guide is based on the experience of Poly-Hi personnel and is subjective. Poly-Hi assumes no liability for incorrect applications or selections.

FEDERAL SPECIFICATIONS
ALUMINUM MILL PRODUCTS

QQ-A-200	**General Specification for Extruded Rod, Bar, Shapes, Tube and Wire.**
QQ-A-200/3	**2024** Extruded Rod, Bar, Shapes, Tube and Wire.
QQ-A-200/8	**6061** Extruded Rod, Bar, Shapes, Tube and Wire.
QQ-A-200/9	**6063** Extruded Rod, Bar, Shapes, Tube and Wire.
QQ-A-200/11	**7075** Extruded Rod, Bar, Shapes, Tube and Wire.
QQ-A-200/15	**7075-T76, T76510, T76511** Extruded Rod, Bar and Shapes.
QQ-A-200/16	**6061-T6** Extruded Structural Shapes.
QQ-A-225	**General Specification for Rolled, Drawn, or Cold Finished Wire, Rod, Bar & Shapes.**
QQ-A-225/1	**1100** Rolled, Drawn, or Cold Finished Wire, Rod and Bar.
QQ-A-225/2	**3003** Rolled, Drawn, or Cold Finished Wire, Rod and Bar.
QQ-A-225/3	**2011** Rolled, Drawn, or Cold Finished Wire, Rod and Bar.
QQ-A-225/4	**2014** Rolled, Drawn, or Cold Finished Wire, Rod and Bar.
QQ-A-225/5	**2017** Rolled, Drawn, or Cold Finished Wire, Rod and Bar.
QQ-A-225/6	**2024** Rolled, Drawn, or Cold Finished Wire, Rod and Bar.
QQ-A-225/7	**5052** Rolled, Drawn, or Cold Finished Wire, Rod and Bar.
QQ-A-225/8	**6061** Rolled, Drawn, or Cold Finished Wire, Rod and Bar.
QQ-A-225/9	**7075** Rolled, Drawn, or Cold Finished Wire, Rod and Bar.
QQ-A-225/10	**6262** Rolled, Drawn, or Cold Finished Wire, Rod and Bar.
QQ-A-250	**General Specification for Sheet & Plate**
QQ-A-250/1	**1100** Sheet and Plate
QQ-A-250/2	**3003** Sheet and Plate
QQ-A-250/4	**2024** Sheet and Plate
QQ-A-250/5	**ALCLAD 2024** Sheet and Plate
QQ-A-250/6	**5083** Sheet and Plate
QQ-A-250/7	**5086** Sheet and Plate
QQ-A-250/8	**5052** Sheet and Plate
QQ-A-250/9	**5456** Sheet and Plate
QQ-A-250/10	**5454** Sheet and Plate
✓QQ-A-250/11	**6061** Sheet and Plate
QQ-A-250/12	**7075** Sheet and Plate
QQ-A-250/13	**ALCLAD 7075** Sheet and Plate
QQ-A-250/24	**7075-T76 and T7651** Sheet and Plate
WW-T-700	**General Specification Drawn Tube Seamless**
WW-T-700/1	**1100** Drawn Tube, Seamless
WW-T-700/2	**3003** Drawn Tube, Seamless
WW-T-700/3	**2024** Drawn Tube, Seamless
WW-T-700/4	**5052** Drawn Tube, Seamless
WW-T-700/5	**5086** Drawn Tube, Seamless
WW-T-700/6	**6061** Drawn Tube, Seamless
WW-T-700/7	**7075** Drawn Tube, Seamless

ALUMINUM ALLOY DESCRIPTIONS
Sheet and Plate Non-Heat Treatable

1100 (QQ-A-250/1)

"Commercially pure aluminum." Minimum specified aluminum content of 99%. Satisfactory results in terms of anodizing. Welding, brazing, and soldering can be accomplished by all the processes used to join any aluminum alloy. Machining is generally more difficult than with other aluminum alloys. Outstanding workability. End uses for 1100 in the various tempers include such items as spun hollow ware, decorative parts of appliances and radio cabinets, heat exchanger fins, small motor covers, small ash trays and novelties, chemical equipment, kitchen items such as measuring spoons, insulating panels, heat reflectors and general sheet metal work.

3003 (QQ-A-250/2)

Corrosion resistance is good to excellent. Finishing produces excellent results when mechanical finishes are required. May discolor (brownish cast) when anodized. May be welded by all the processes used to weld aluminum. Soldering presents no problems. Brazing requires special techniques. Machining is difficult. Strength is increased to approximately 20% higher than 1100 due to addition of manganese. Workability is excellent. Typical end uses include awnings, fish boxes, food and chemical handling and storage equipment, furniture (lawn chairs, carts, trays, etc.), heat exchangers, highway sign panels, mobile home siding, trailer roof sheeting, ventilators, flashing and spun parts.

5005

Corrosion resistance is excellent. Finishing produces superior results in that all standard finishes may be applied. Anodizing in particular produces a pleasing appearance. Color match between anodized 5005 sheet and 6063 extrusions is good. Welding may be done by any method used to weld other aluminum alloys. Brazing requires extra care. Solderability not as good as 1100 or 3003. Machining is difficult. Strength about equal to 3003. Workability is generally comparable to 3003 except in severe drawing. Typical end uses include architectural applications, kitchen utensils and equipment, mobile home siding, outdoor signs, patio covers and reflectors.

5052 (QQ-A-250/8)

Corrosion resistance is excellent. More resistant to attack from salt water than 1100. Finishing may be accomplished by all standard methods. However anodizing produces a yellow cast. Welding may be done by any method used to weld other aluminum alloys. Brazing and soldering are difficult. Machining is difficult. High strength. Excellent for structures subject to much vibration. Workability is effected by relatively high magnesium. Typical end uses include aircraft fuel tanks, boats (hulls and components), bulk food processing equipment, chemical drums, fertilizer tanks, pressure vessels, storage tanks, tank trailer bodies and sheet metal products.

ALUMINUM ALLOY DESCRIPTIONS
Sheet and Plate Heat-Treatable

6061 (QQ-A-250/11)

Corrosion resistance is good. For additional corrosion resistance may be clad with alloy 7072 which has a nominal 1% zinc content. Finishing may be done by any of the standard finishing practices. To anodize extra care in removing heat treat stains must be exercised. Welding is readily accomplished by any of the normal processes. Brazing and soldering are feasible, but trial runs may be necessary to establish best method. Machining produces good results. A strong tough alloy 6061 is the least expensive and most versatile of the heat-treatable alloys. Generally recommended that drawing be attempted in the T4 or 0 condition. Typical end uses include aircraft landing mats, aircraft water storage tanks, architectural sections and billboard sign panels.

2024 (Bare-QQ-A-250/4) (Clad-QQ-A-250/5)

Corrosion resistance in the heat-treated condition is only fair. Corrosion resistance is improved by cladding. Finishing can be accomplished by any of the processes applicable to aluminum alloys, however, anodizing will not provide a consistent color match. Welding by resistance methods is feasible. Fusion not normally recommended. Brazing and Soldering not recommended. Machining is relatively good in either 0, T4 and T351. Good strength-weight ratio. Workability in the annealed condition is good. Typical end uses include aircraft skins, cowls and structurals, truck trailer structurals and side panels.

7075 (QQ-A-250/12) (Clad-QQ-A-250/13) (Clad 1 Side-QQ-A-250/18)

Corrosion resistance when heat treated and fully aged is only fair. Improved corrosion resistance is obtained by cladding. Anodic coatings are applied for corrosion protection only. Welding by the fusion method gives inferior results. Resistance welding with clad material in T6 gives good results. Resistance welding of bare material not recommended. Brazing and soldering not recommended. Machining requires high surface speed of the cutter for good results. One of the highest strength alloys commercially available. Workability extremely low in the T6 temper. Typical end uses include aircraft skins, cowls and structures.

NOTE

Aluminum alloy descriptions contained on this page and on the preceding page are stated in general terms only. For more specific detailed information on these alloys or other aluminum alloys contact your Gate City representative.

COLOR CODES

ALLOY STEEL:

4130	Blue & Gray
4140/Annealed	Brown & Gray
4140/Heat Treated	Brown & Gold
4340/Annealed	Red & Yellow
T1	Red & Black
Abrasion Resistant	Gray

Stainless Steel:

Type 303	Red
Type 304	Gold
Type 304 L	White
Type 316	Green
Type 321	Black
Type 347	Pink
Type 410	Blue
Type 416	Yellow
Type 440 C	Gray
15-5 PH	Orange
17-4 PH	Brown

Drill Rod:

Oil Hardening	Red
Water Hardening	Green

Aluminum:

2011-T3	Brown
2014-T3	Gray
2024-T3&4	Red
3003-H14	Gold
5052-H32	Green
6061-T6	Blue
7075-T6	Black

9/23/88

1. PSC acceptance criteria and inspection items.

Ans.
PSC will use US standard "military specifications" (mil-spec) and/or US federal/DOE standard specifications and acceptance criteria for all items (electronics, Al, CH_2, etc) for which such standards and acceptance criteria exist. A few examples are included for your examination.

For some items PSC will establish more stringent or testing intensive criteria. The graphite is one such material. We will require it to meet strict standards by the vendor's signed statement and also pass an independent testing laboratory's evaluation as well. [this is a multielemental ppm level quantitative impurity assay]

Generally speaking, PSC will subject all materials to an initial visual inspection for any obvious defects, followed by PSC's own operational tests. All our testing results as well as the results from vendor's testing will be documented and made available for inspection.

参考資料－7

September 6, 1988

Pajarito Scientific Corporation
322 Kimberly Lane
Los Alamos, New Mexico 87544

Subject: Seismic Analysis JGC/PNC NDA System

Dear Mr. Caldwell:

Attached are seismic calculations for the JGC/PNC Non-Destructive Analysis System. The bases for these calculations are a 0.36 g ailowable for the system itself and 1.0 g allowable for the anchor bolts.

The following three configurations were investigated to assess the integrity of the design:

1. Aggregate System Weight on Base Plate   K = 1.0
2. Side Module-Separation from Unit   K = 0.36
3. Open Position-Door on Rails
   A. Force Perpendicular to Travel   K = 1.0
   B. Force Parallel to Travel   K = 1.0

A bolt analysis was performed to determine the suitability of the bolts chosen for the various configurations. At the time this report was written, the number and the location of anchor bolts as well as the joint design had not been finalized; therefore, Option 1 and Option 2 in the anchor bolt calculations were explored. The 'Vendor Supplied' threaded assemblies utilize screws rather than bolts. If a failure should occur, it is preferable for the screw to fail under tension rather than have either the internal or external threads strip. The Required Length of Engagement calculations determine the engagement of mating threads sufficient to carry the full load necessary to break the screw. Mechanical properties for these calculations are based on a Low to Medium Carbon-Steel for the external threads and on 6061-T6 Aluminum for the internal threads. I will collaborate with the mechanical designer to ensure the final design meets your client's seismic criteria.

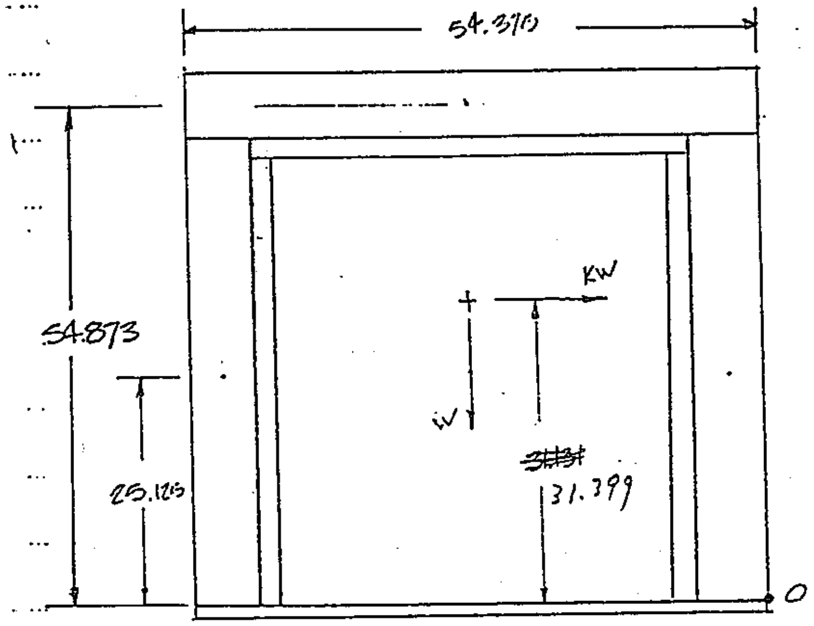
Yours truly,

Timothy T. Fife

Timothy T. Fife, P.E.
117 Azure Drive
Los Alamos, New Mexico 87544

COLORADO REGISTERED
TIMOTHY THOMAS FIFE
22615
PROFESSIONAL ENGINEER

1. AGGREGATE ON BASE PL K=1



$$\bar{Y} = \frac{\Sigma(WY)}{W} = \frac{2\times 25.125(1040+1525)+54.873(1371)}{6501}$$

$$\bar{Y} = 31.399'$$

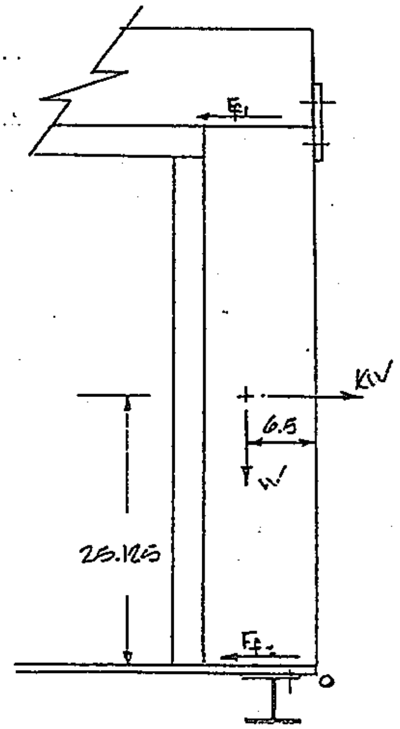
$$\Sigma M_o \Rightarrow (6501)\left(\frac{54.370}{2}\right) + F_b(52.257) = (6501)(31.399) \Rightarrow \underline{F_b = 524^{\#}}$$

F_b: RESULTING FORCE ON THE ANCHOR BOLTS DUE TO A LATERAL LOAD EQUAL TO THE WEIGHT OF THE UNIT

THIS ASSUMES AN ANCHOR BOLT LOCATION THRU THE FLANGE OF THE SUPPORT STRUCTURE.

THERE WILL BE NO SHEAR LOAD ON THE ANCHOR BOLTS BECAUSE $\mu_s = 1.05$ (REF. STAND. HANDBOOK MECH. ENG. Al/Al DRY)

... 2. SIDE MODULE - SEPARATE FROM THE UNIT $K = .36$



$W = 1525$

$KW = 550^{\#}$

FRICTION FORCES OPPOSING SEISMIC OR LATERAL LOAD

TOP MODULE ON SIDE : F_{f1}

$N = \frac{W}{3}$, $\mu_s = 1.05 \Rightarrow F_{f1} = \left(\frac{1371}{3}\right)(1.05) = 480^{\#}$

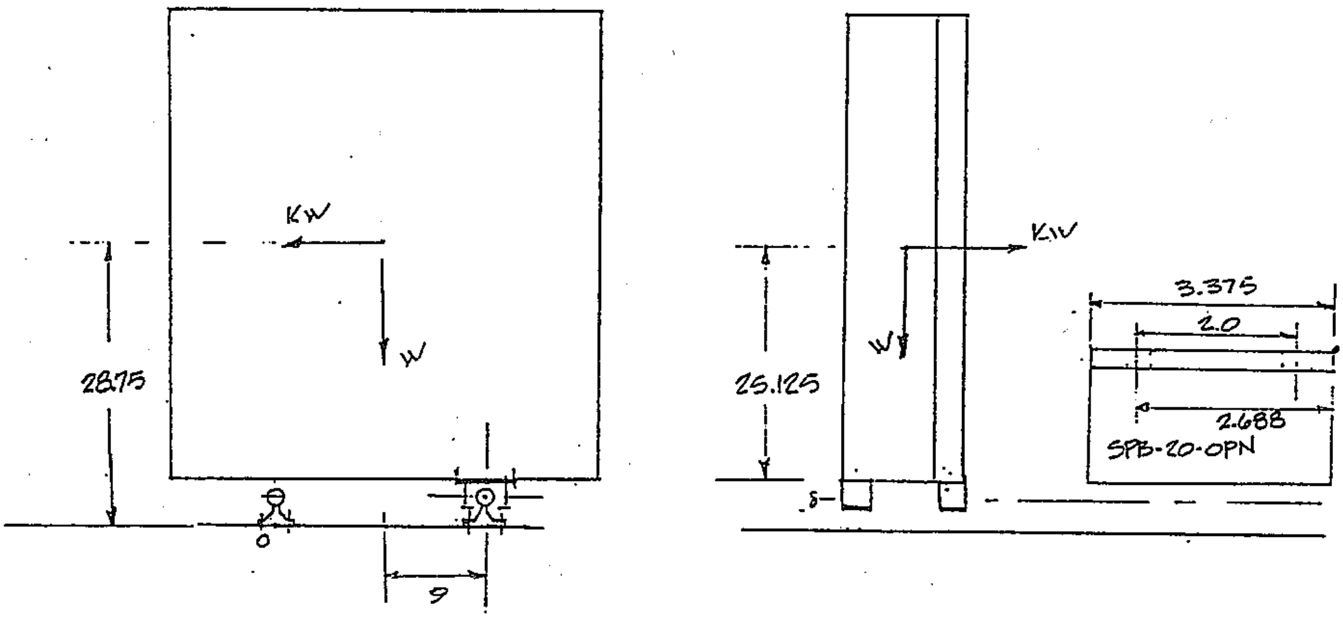
SIDE MODULE ON BASE : F_{f2}

$N = 1525$, $\mu_s = 1.05 \Rightarrow F_{f2} = (1525)(1.05) = 1601.25$

$\Sigma F \Rightarrow$ FRICTION IS SUFFICIENT TO COUNTER ACT A LATERAL SEISMIC LOAD OF .36g

HOWEVER STRAP PLATES WILL BE BOLTED AT THE MODULE INTERFACE HOLDING THE MODULES TOGETHER THUS RESISTING LATERAL LOADS AND TRANSMITTING THE FORCES TO THE ANCHOR BOLTS

3. OPEN POSITION : DOOR ON RAILS K=1



⊥ TO RAILS:

$$\Sigma M_o \Rightarrow (1040)(9) + F_b(13) = (1040)(28.75) \Rightarrow F_b = 1{,}141^{\#}$$

DUE TO BOLT HOLE PATTERN OF THE SHAFT SUPPORT RAILS, Fb DUE TO A LATERAL LOAD WILL BE DISTRIBUTED AMONG 2 ea. ¼-20 SCREWS MINIMUM.

⇒ FORCE/SCREW = 570.5# WHEN SEISMIC LOAD ⊥ TO RAILS

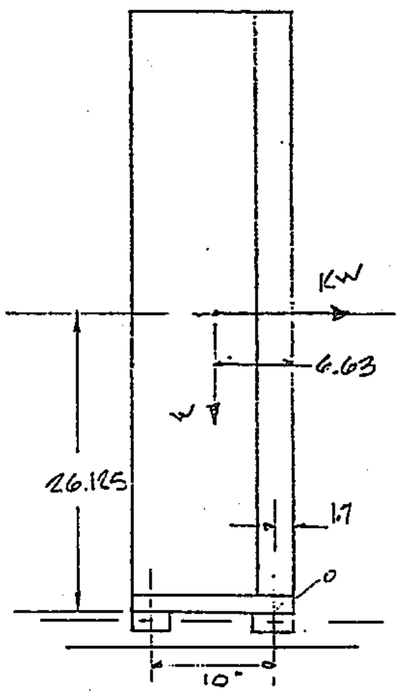
SHEAR STRESS ON RAILS

$F = 1040^{\#}$

$A = 2\,(3.375 \times .563) = 3.797\,in^2$

$\tau = \dfrac{F}{A} = 274\ psi$ — INCONSEQUENTIAL

3. CONTINUED



// TO RAILS

$\Sigma M_b \Rightarrow (1040)(4.603) + F_b(10) = (1040)(26.125) \Rightarrow F_b = 2238^{\#}$

DUE TO THE BOLT HOLE PATTERN IN THE SPB-20-OPN PILLOW BLOCK THE LOAD WILL BE DISTRIBUTED AMONG 8 EA. #10x24 SCREWS

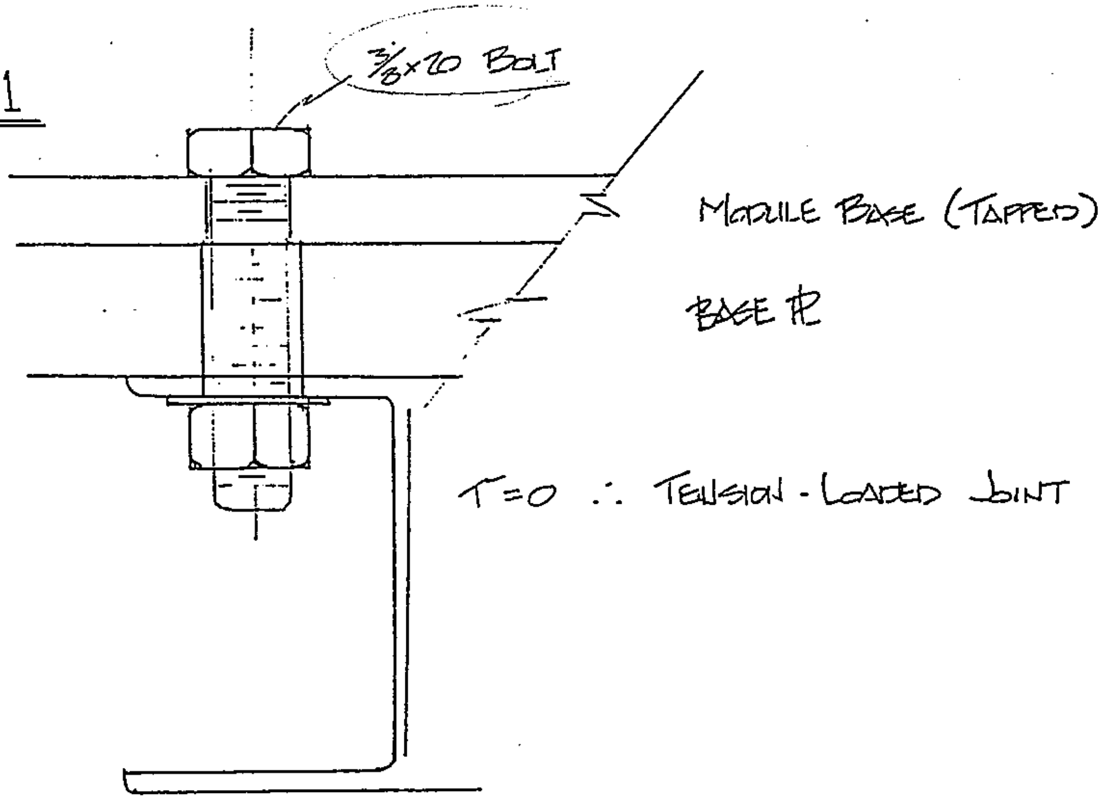
FORCE / SCREW = 280# WHEN SEISMIC LOAD // TO RAILS

BOLT ANALYSIS: ANCHOR BOLTS TO BASE PL.

- ALL OF THE CASES INVOLVE SCREWS RATHER THAN BOLTS, TAPPING INTO 6061-T6 Al PL. TO ACHIEVE A SHEAR STRENGTH OF THE INTERNAL THREADS EQUAL TO THE TENSILE STRENGTH OF THE SCREW THE LENGTH OF ENGAGEMENT SHOULD APPROACH 2.5 TIMES THE NOMINAL DIA OF THE SCREW. (REF. MACHINERY'S HANDBOOK, 20TH ED.) THIS CRITERIA POSES A PROBLEM W/ THE ANCHOR BOLTS TO THE BASE PL.

RECOMMEND 3/8 x 16 UNC BOLTS TO ANCHOR TO THE BASE PL.

OPTION 1



OPTION 1 CONT.

LOAD REQ'D TO BREAK THREADED PORTION OF BOLT : P

$$P = S\left[.7854\left(D - \frac{.9743}{n}\right)^2\right]$$

S = ULT. TENSILE STR. = 74KSI
D = BASIC MAJ. ϕ = .375
n = N° OF THREADS/in = 16

$$P = 74{,}000\left[.7854\left(.375 - \frac{.9743}{16}\right)^2\right]$$

$= \underline{5{,}728^{\#}}$ ⇒ FACTOR OF SAFETY = 11.0 FOR A SINGLE ANCHOR BOLT

OPTION 2: 3/8-16 TAPPED INTO BASE OF THE MODULE, CLASS 2

n = 16 – N° OF THREADS
K_n max = .321 – MAX MINOR ϕ INTERNAL
E_s min = .3287 – MIN PITCH ϕ EXTERNAL
Le LENGTH OF ENGAGEMENT

$$Le = \frac{2A_t}{\pi K_{n\,max}\left[\frac{1}{2} + .57735n\left(E_{s\,min} - K_{n\,max}\right)\right]} = \frac{.155}{.576}$$

$= .269''$ FOR EQUAL TENSILE STR. OF INT. & EXT. THREADS

Q = JLe = REQ'D LENGTH OF ENGAGEMENT, DIFFERING MATERIALS

$$J = \frac{A_s \times \sigma_T \text{ EXT. MAT.}}{A_n \times \sigma_T \text{ INT. MAT.}}$$

OPTION 2 CONT.

$$A_S = \pi n \, Le K_{n\,max} \left[\frac{1}{2n} + .57735 (E_{s\,min} - K_{n\,max}) \right]$$
$$= .155$$
$$A_n = \pi n \, Le \, D_{s\,min} \left[\frac{1}{2n} + .57735 (D_{s\,min} - E_{n\,max}) \right]$$
$$= .223$$

$D_{s\,min} = .3643$ - MIN MAJ ϕ EXT.

$E_{n\,max}$ .3401 - MAX PITCH ϕ INT.

$$J = \frac{(.155)(14)}{(.223)(45)} = 1.144$$

$Q = J\,Le = (1.144)(.269) = \underline{.308''}$ = REQ. LENGTH OF ENGAGEMENT TO PREVENT STRIPPING OF INT. THREAD.

IF FAILURE OF A THREADED ASSEMBLY SHOULD OCCUR, IT IS PREFERABLE FOR THE SCREW TO BREAK RATHER THAN HAVE EITHER THE EXT. OR INT. THREAD STRIP ∴ THE LENGTH OF ENGAGEMENT OF MATING THREADS SHOULD BE SUFFICIENT TO CARRY THE FULL LOAD NECESSARY TO BREAK THE SCREW.

IF OPTION 2 IS CHOSEN THE $\frac{1}{2}''$ MODULE BASE SHOULD BE TAPPED THRU TO ENSURE REQUIRED LENGTH OF ENGAGEMENT.

BOLT ANALYSIS: RAILS TO BASE PL.

$F = 570.5^{\#}$

$A_t = \frac{\pi}{4}\left(.250 - \frac{.9743}{20}\right)^2 = .032 \text{ in}^2$

$\sigma_t = \frac{F}{A_t} = \underline{17{,}925 \text{ psi}}$

PROOF STRENGTH FOR A LOW-MEDIUM CARBON STEEL BOLT = 55,000 psi

$\Rightarrow$ <u>FACTOR OF SAFETY = 3</u>

REQ'D LENGTH OF ENGAGEMENT:

$\frac{1}{4}$-20 CLASS 2 FIT: $n = 20$

$K_{n\,max} = .207$	MAX MINOR ϕ INT
$E_{s\,min} = .2127$	MIN PITCH ϕ EXT
$D_{s\,min} = .2408$	MIN MAJ ϕ EXT
$E_{n\,max} = .2223$	MAX PITCH ϕ INT.

$$L_e = \frac{(2)(.032)}{\pi(.207)\left[\frac{1}{2}+(.57735)(20)(.2127-.207)\right]} = .174''$$

$$A_s = \pi(20)(.174)(.207)\left[\frac{1}{40} + .57735(.2127-.207)\right] = .064$$

$$A_n = \pi(20)(.174)(.2408)\left[\frac{1}{40} + .57735(.2408-.2223)\right] = .094$$

$$J = \frac{(.064)(74)}{(.094)(45)} = 1.120$$

$Q = J L_e = \underline{\underline{.195'' = \text{REQ'D LENGTH OF ENGAGEMENT}}}$

BOLT ANALYSIS: SPB-20-OPN PILLOW BLOCK TO CARRIAGE

$F = 280^{\#}$

$A_t = \frac{\pi}{4}\left(.19 - \frac{.9743}{24}\right)^2 = .018 \text{ in}^2$

$\sigma_t = \frac{F}{A_t} = \underline{15{,}971 \text{ psi}}$

PROOF STRENGTH = 55,000 psi $\Rightarrow$ <u>FACTOR OF SAFETY = 3.4</u>

REQ'D LENGTH OF ENGAGEMENT:

#10-24 CLASS 2 FIT: $n = 24$

$K_{n\,max} = .156$ - MAX MINOR ϕ INT

$E_{s\,min} = .1619$ - MAX PITCH ϕ EXT

$D_{s\,min} = .1818$ - MIN MAJ ϕ EXT

$E_{n\,max} = .1672$ - MAX PITCH ϕ INT

$$L_e = \frac{(2)(.018)}{\pi(.156)\left[\frac{1}{2} + .57735(24)(.1619 - .156)\right]} = .126''$$

$$A_s = \pi(24)(.126)(.156)\left[\frac{1}{48} + .57735(.1619 - .156)\right] = .036$$

$$A_n = \pi(24)(.126)(.1818)\left[\frac{1}{48} + .57735(.1818 - .1672)\right] = .051$$

$$J = \frac{(.036)(74)}{(.051)(45)} = 1.171$$

$Q = J L_e = \underline{\underline{.148 = \text{REQ'D LENGTH OF ENGAGEMENT}}}$

参考資料－8

IDAHO NATIONAL ENGINEERING LABORATORY

IDAHO NATIONAL ENGINEERING LABORATORY

The Idaho National Engineering Laboratory (INEL), situated on 890 square miles of the vast sagebrush desert between Idaho Falls and Arco in southeast Idaho, is the setting for some of the most advanced energy research in the world.

Established in 1949 as the National Reactor Testing Station, the INEL contains the largest concentration of nuclear reactors in the world. Over the years, 52 reactors, most of them first-of-a-kind facilities, have been built at the INEL. Fourteen of them are operating or operable, the others having been phased out after completion of their research missions.

The INEL has a long tradition of pioneering new advances in science. For example, in 1951, one of the most significant scientific accomplishments of the century occurred at the INEL—the first use of nuclear fission to produce electricity.

The first pressurized water reactor and boiling water reactor prototypes were built and operated at the INEL in the 1950's. One of the boiling water reactor experiments, BORAX-III, was the first to light an American town—Arco, Idaho—in 1955.

Today, the INEL is a leading center for nuclear safety research, defense programs, nuclear waste technology, and developing advanced energy concepts.

The Site's name was changed to the Idaho National Engineering Laboratory in 1974 to better characterize current programs, which now include research for engineering and nonnuclear, as well as nuclear, energy programs.

In addition, work is conducted in fusion energy, low-head hydropower, industrial energy conservation, environmental research, strategic and critical materials, computer code development, materials testing, and advanced instrumentation development.

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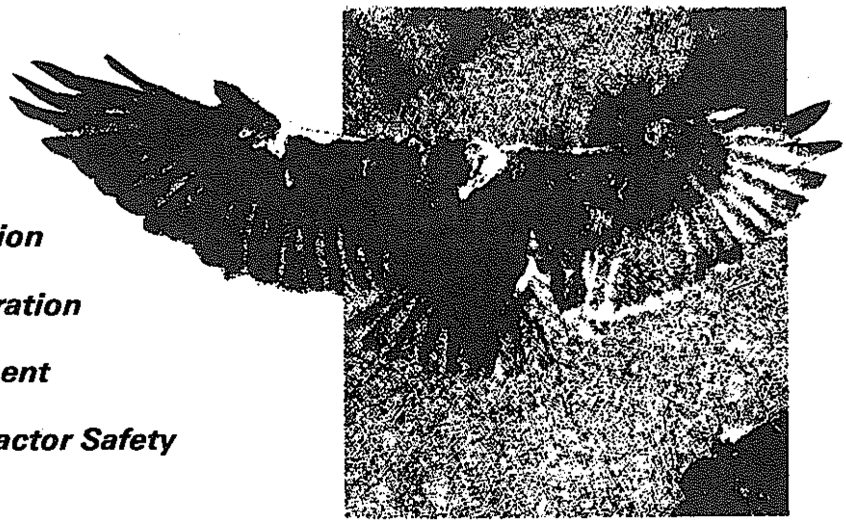


IDAHO NATIONAL ENGINEERING LABORATORY

ADMINISTRATION

The U.S. Department of Energy Idaho Operations Office (DOE-ID) administers programs at the INEL and provides support services for INEL programs administered by DOE's Chicago and Pittsburgh field offices. The INEL Site has nine operating areas with a total property value of about $3.6 billion. During 1986, INEL programs cost about $778 million. Costs for 1987 were approximately $806 million. Estimated costs for FY 1988 amount to $810 million.

In addition, DOE-ID administers research work at the following locations:

— Three Mile Island (TMI) Project Office near Harrisburg, Pennsylvania, where INEL researchers are studying the damage caused by the accident in 1979 at the TMI Unit 2 commercial reactor.

Research is aimed at removing the core, studying core materials, immobilizing the waste, and determining the accident's sequence of events. These activities are providing information for improving light water reactor plant safety, reliability, and operability, and for advancing the technology of facility decontamination, and waste handling, transportation, and disposal.

Knowledge gained from TMI research is being disseminated to others engaged in design, construction, operation, maintenance, review, and regulation of commercial nuclear power plants.

— West Valley Project Office, near Buffalo, New York, where researchers are demonstrating safe and effective methods for solidifying and disposing of high-level radioactive liquid wastes.

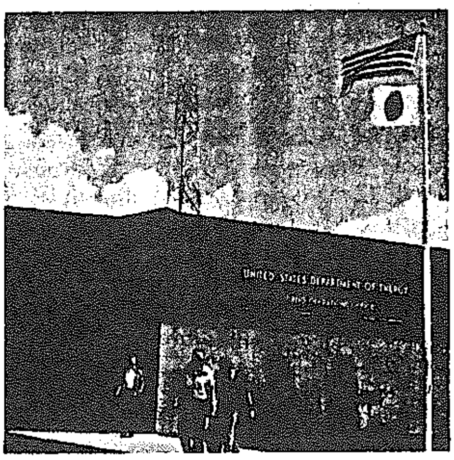
A commercial fuels reprocessing facility, operated at the West Valley Site from 1966 to 1972, generated some 600,000 gallons of liquid waste which is stored in underground tanks. Before solidification can begin, almost every aspect of nuclear waste management will be demonstrated. This will include the decontamination of a nuclear facility by developing tools and equipment to meet special requirements and techniques, the conceptual design of solidification equipment within existing facilities, and disposal of low-level and transuranic wastes.

Spent nuclear fuel management, consolidation, and transportation are also being done at the West Valley facility.

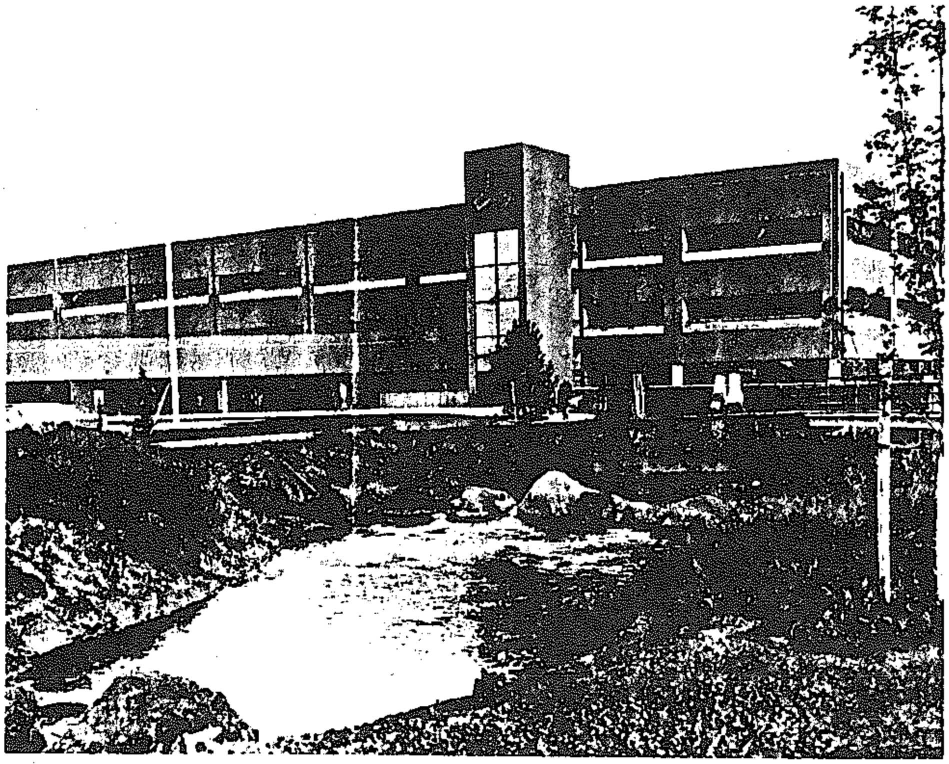
— Butte Project Office in Butte, Montana, where researchers at the Magnetohydrodynamics Component Development and Integration Facility (MHD-CDIF) are developing equipment to generate electricity from coal-fired gases. Research is aimed at determining if electricity from coal or coal-derived fuel can be produced at a greater efficiency rate than electricity from conventional fossil-fuel-powered plants.

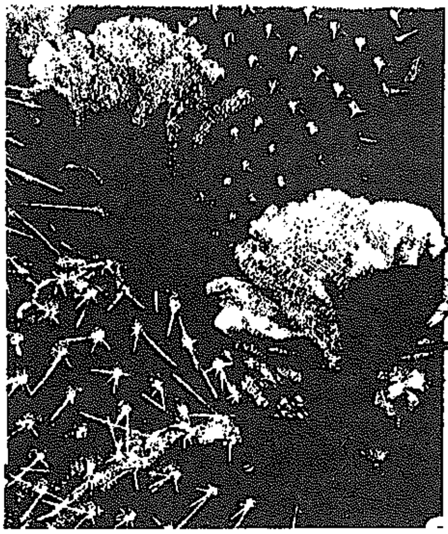
— Denver Support Office in Denver, Colorado, which manages DOE grants and energy conservation programs in the Rocky Mountain region. These programs are in such areas as low-income home weatherization, state energy conservation, energy extension services, and energy conservation for schools, hospitals, and public buildings.

— Grand Junction Area Office in Grand Junction, Colorado, which manages portions of the National Waste Terminal Storage Program, Uranium Mill Tailings Remedial Action Programs, and the Uranium Leasing Program. This office provides geotechnical expertise in support of long-term waste storage, uranium mill tailings cleanup, and federal uranium lease management.



The DOE-ID headquarters building is located in Idaho Falls





IDAHO
NATIONAL
ENGINEERING
LABORATORY

The Willow Creek Building

EMPLOYMENT

The INEL employs about 10,500 people, or about 2.5 percent of Idaho's total workforce. While about 350 people work for the DOE-ID operations office, most workers are employed by private firms under contract to DOE. These include Argonne National Laboratory, Protection Technology, Inc., EG&G Idaho, Rockwell-INEL, Westinghouse Electric Corporation, and Westinghouse Idaho Nuclear Company.

Construction management services are provided by MK-Ferguson of Idaho Company. Project construction employment ranges between 500 and 1000, and $35-$50 million is awarded annually in construction contracts, primarily to Idaho firms.

About 8,000 workers staff the nine operating areas on the INEL Site. Some administrative, scientific support, and nonnuclear laboratory programs are housed in Idaho Falls in seven buildings: the DOE-ID Headquarters, the Willow Creek Building, two Technical Support Buildings, the Remote Office Building, the Computer Science Center, and the INEL Research Center.

The INEL work force comprises the largest concentration of technical professionals in the northern Rocky Mountain region. More than 1,300 employees hold engineering degrees. About 600 have science degrees, mostly physical science, and more than one employee in three has a college degree.

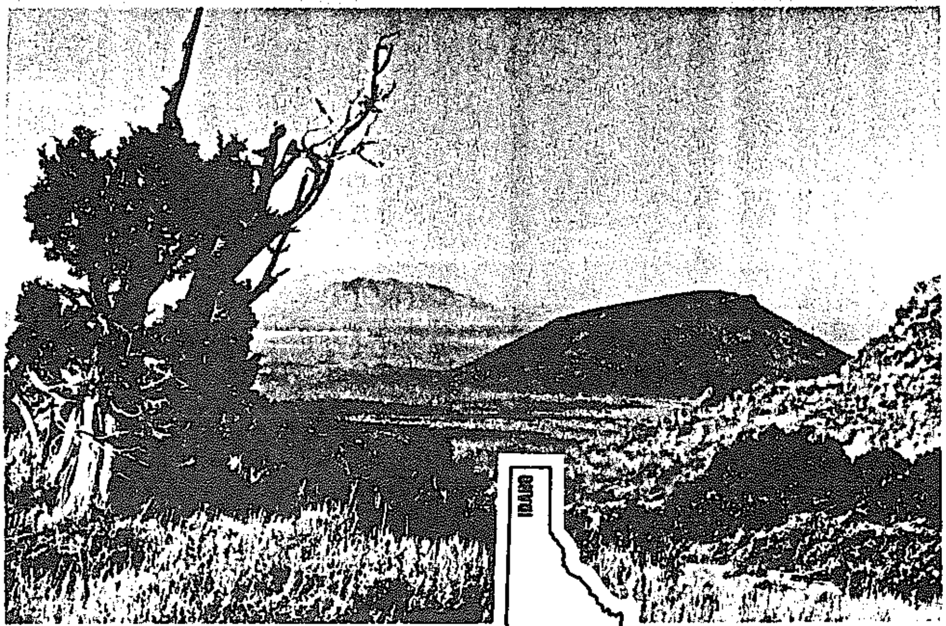
Approximate INEL payroll during FY 1987 was $370 million. Directly and indirectly, the INEL generated approximately $794 million in wages and salaries.

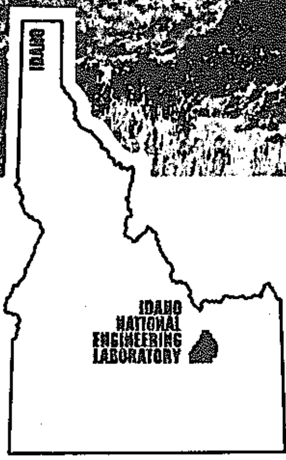
Employees live in more than 30 adjacent communities, with the heaviest concentration in Idaho Falls. Bus service is provided to the Site from the major communities.

The INEL and its employees significantly influence Idaho's social and economic welfare. Many employees are active in civic and community affairs, helping to improve the quality of life in the area.

ENVIRONMENT

The INEL Site covers nearly 570,000 acres or 890 square miles of dry, cool desert at an elevation of almost one mile above sea level. The land is bordered on the north and west by snow-capped mountains and on the south by three towering buttes.





The site is located in southern Idaho on a high desert plain.

The Site is nearly 39 miles long from north to south and about 36 miles wide at its broadest part. The Site boundaries nearest to populated areas are 22 miles west of Idaho Falls, 23 miles northwest of Blackfoot, 44 miles northwest of Pocatello, and 7 miles southeast of Arco.

Annual precipitation has averaged only 8.5 inches in the last 15 years.

Underlying the area is a huge natural underground reservoir of water in basaltic lava rock called the Snake River Plain aquifer.

The average annual temperature at the Site is 42°F, with extremes of 103°F and -47°F.

RADIOLOGICAL AND ENVIRONMENTAL SCIENCES LABORATORY (RESL)

DOE-ID laboratory scientists study and monitor water, air, soil, and area farm produce throughout an area of about 5,000 square miles. Data obtained from this work substantiate that INEL operations are safe for Site employees and the public. Monitoring results are reported quarterly to the Environmental Protection Agency (EPA), the State of Idaho, DOE-ID, and the various INEL contractors.

RESL is recognized internationally as one of the most advanced laboratories performing radiation safety work. The laboratory is renowned for its pioneering work in the fields of radiation monitoring devices, ultrasensitive methods for radiochemical analysis, and radiation safety research and development.

RESL maintains direct traceability to the National Bureau of Standards. This credential is necessary for RESL to test the quality of programs conducted by other laboratories. An example of this testing is the DOE Laboratory Accreditation Program for Personnel Dosimetry in which RESL tests DOE and DOE-contractor laboratories to assure that results of the dosimetry programs meet specified standards.

The laboratory's specialized individual health protection functions include personnel and environmental radiation dosimetry (to detect external radiation) and whole body counting and radiochemical analysis (to measure possible internal deposition of radionuclides in workers). Records are maintained on each employee's radiation exposure history.

The 570,000-acre INEL Site is one of two National Environmental Research Parks. All lands within the INEL boundaries comprise a protected outdoor laboratory where scientists from DOE, other federal and state agencies, universities, and private research foundations conduct ecological studies. The RESL staff coordinates the various research projects which have included wildlife use of burned areas, bobcat density and dispersal, use of honeybees and lichens as indicators of pollutants, effects of harvester ants on

ENVIRONMENTAL MONITORING

Since the early years of the Radioactive Waste Management Complex (RWMC) operation, an active environmental monitoring program has been maintained to measure the potential hazards associated with waste located at the RWMC. Air, water, soil, plants, and animals are monitored at and near the RWMC for the presence and effects of radioactivity. Also, supporting special studies are performed as needed, and environmental radionuclide transport pathways are evaluated to determine if additional monitoring activity is required in a particular area.

Recent monitoring results have shown that organic chemicals are being released from waste that was deposited in the ground prior to 1970. There are also indications that some radioactive material may have seeped into the ground from transuranic waste disposal sites established before 1970.

Although the monitoring results indicate no potential harm to the environment, actions have been taken to further increase monitoring activities and to develop plans to prevent contamination of the Snake River Plain aquifer.

ENVIRONMENTAL COMPLIANCE

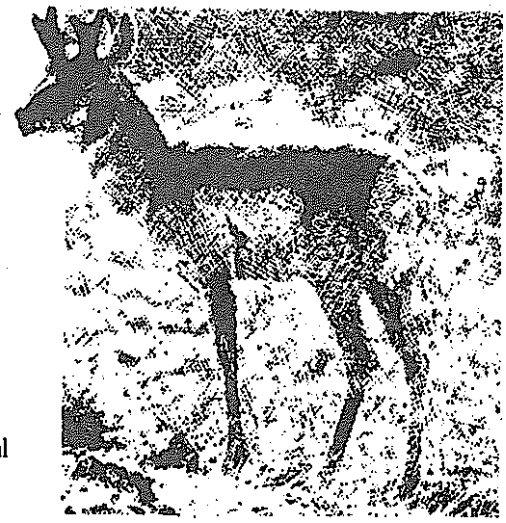
To ensure the safety of people who work at or live near the INEL Site and to protect the environment, all INEL operating facilities are regulated by DOE orders and EPA and State regulations. All new facilities strictly adhere to the National Environmental Policy Act, which requires environmental analyses ranging from environmental evaluations to environmental impact statements, depending on the complexity and potential impact of the facility to the environment.

A thorough evaluation of past waste disposal sites and practices at the INEL has resulted in changes in the ways of disposing waste, and plans are pending to perform remedial actions on certain waste disposal sites that might pose future environmental threats.

movement of contaminants, use of vegetation to control moisture above buried waste, and long-term vegetation changes.

In addition, RESL serves as a reference laboratory, providing a variety of services to Nuclear Regulatory Commission programs. The services include chemical, environmental, dosimetry, and data analyses for nuclear power plants, fuel fabrication facilities, and fuel processing plants. Also, RESL provides equipment, supplies, personnel training, and various radiochemical standards to the NRC regional offices.

Working in conjunction with, but independent of, DOE scientists and technicians are a number of geological, hydrological, and meteorological experts employed by the U.S. Geological Survey and the National Oceanic and Atmospheric Administration.



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WATER REACTOR SAFETY RESEARCH

Water reactor safety research has been conducted at the INEL since 1955 when the first reactor safety studies were conducted in the Special Power Excursion Reactor Test No. 1 (SPERT-1) Facility.

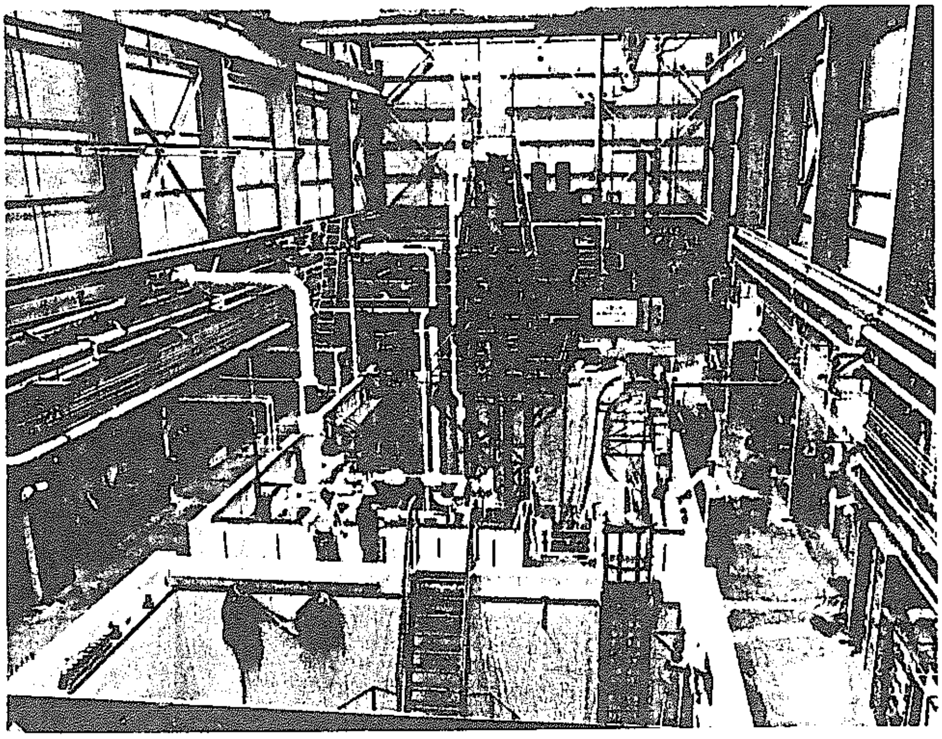
Early research focused mainly on the so-called "runaway power" accident. Testing in four SPERT reactors led to an understanding of several natural mechanisms that resist runaway power conditions. SPERT tests also led to the development of improved reactor safety control systems.

Other major contributions to water reactor safety research have been made by the Power Burst Facility (PBF) and the Loss-of-Fluid Test (LOFT) Facility. Tests conducted at these facilities have provided information about fuel and systems behavior during various hypothetical accident scenarios.

EG&G Idaho conducts the INEL's water reactor safety research programs for DOE and NRC. Some programs are funded partially by foreign governments or private institutions.

The purpose of these programs is to study the behavior of nuclear power plants, nuclear fuel, and reactor safety systems during off-normal or accident operating conditions. Equipment, procedures, and operator performance are studied to reduce the possibility of reactor accidents and to reduce the extent of damage should an accident occur.

This research is conducted in accordance with the nation's policy to continue seeking information to resolve safety issues related to reactor design, licensing, and operation.



Semiscale Facility located at the INEL Test Area North

NUCLEAR REGULATORY COMMISSION (NRC) PROGRAMS

The INEL provides support to both the research and licensing areas of the NRC. This support is for NRC research efforts to provide the technical basis for rulemaking and regulatory decisions and to provide technical assistance in licensing decisions. Key elements of NRC-sponsored research are concentrated in experiment analysis; development, assessment, and application of analytical computer codes for describing reactor systems and fuel behavior; and age-related safety issues. The major goal of this combined program is to understand various phenomena and assess analytical models, thus providing a means of predicting reactor behavior during operational transients and postulated accidents.

NUCLEAR REACTOR REGULATION

The INEL provides assistance to the NRC for nuclear reactor regulation in such areas as operating reactor licensing, regional reactor operator licensing examinations, operating license reviews, generic safety assessments, unresolved safety issues, human factors program issues, and risk assessment.

INSPECTION AND ENFORCEMENT

The INEL assists NRC in inspection and enforcement by studying quality assurance and quality control programs at selected operating sites and by integrated design inspections at nuclear power reactors under construction. In addition, the INEL provides technical support for NRC inspections of manufacturers who supply basic reactor components.

The INEL conducts drills for NRC Emergency Response Operations Centers to test NRC response to incidents at nuclear power plants.

TECHNICAL INTEGRATION CENTERS

In 1985, the INEL began consolidating the many projects it administers for the NRC Office of Nuclear Regulatory Research into a few broad technical missions to facilitate the integration of research work. Four areas were selected: light water reactor (LWR) thermal-hydraulics, severe accident behavior, reactor aging safety research, and risk data systems.

LIGHT WATER REACTOR THERMAL-HYDRAULICS

Research in this area includes developing the Nuclear Plant Analyzer (NPA) and maintaining and improving existing NRC thermal-hydraulic computer codes.

The NPA system is being developed to assist NRC in analyzing plant responses to actual reactor accidents. The system combines graphic representations of nuclear power plants with various computer codes and data sources to provide new analysis tools for accident evaluation.

Maintaining and improving NRC thermal-hydraulic computer codes are major efforts, also. These codes are used to predict sequences of events during hypothetical reactor accidents and the potential consequences. For example, INEL researchers developed the Reactor Excursion Leak Analysis Package (RELAP) code series, which is used extensively by nuclear regulatory agencies and industries in the U.S. and abroad. NRC uses RELAP codes and other codes developed and improved at the INEL for licensing power reactors and for establishing regulations and standards. The codes are also used for engineering and evaluating reactor systems and designs.

The experimental element of this program addresses the effects of inadequately understood thermal-hydraulic phenomena. Support is provided to the two-dimensional/three-dimensional (2D/3D) experimental effort to gain an understanding of refill and reflood behavior, emergency core cooling system performance, and data for assessing the thermal-hydraulic codes. Support is also provided to the Rig Of Safety Assessment (ROSA)-IV and Integral System Test Programs. These experiments will provide data for developing, assessing, and determining uncertainties in thermal-hydraulic computer codes.

SEVERE ACCIDENT BEHAVIOR

This mission integrates the NRC's severe accident research, assists in providing the technical basis for resolving source-term questions, and provides the basis for research in accident management, equipment/instrument qualification, and man-machine interface.

The Severe Accident Code Integration and Development element provides a mechanistic, fast-running computer code that models severe accident progression within the primary coolant system up to, but not including, vessel breach.

Through the use of computer codes and experimental data, the severe accident sequence analysis and research integration element will perform tasks that include providing technical guidance for experimental and code development efforts and providing pressurized water reactor and boiling water reactor severe accident sequence analyses.

The final severe fuel damage test was performed at the Power Burst Facility in February 1985. Postirradiation examinations and analyses are presently under way on this and previous severe fuel damage experiments to provide an understanding of damage mechanisms, intermediate and final damage states of fuel, retained fission products, and fission product chemical forms.

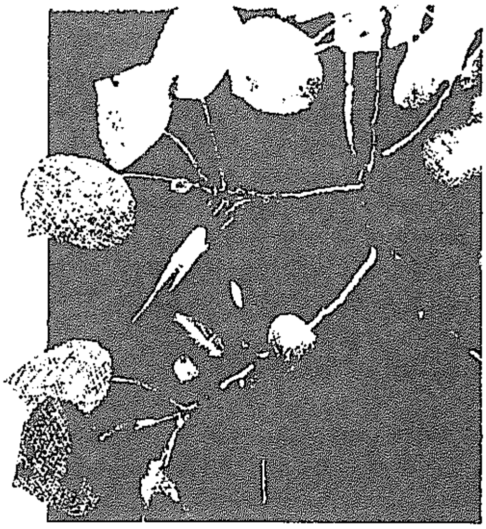
REACTOR AGING AND SAFETY RESEARCH

The NRC is increasing its efforts to understand the physical deterioration of nuclear power plants so that adequate criteria can be developed to ensure safe operation of older plants.

To support the NRC in this area, the INEL is establishing improved criteria, requirements, and methodologies to qualify current and future equipment and systems; establishing criteria concerning aging to maintain adherence to technical specifications; and developing criteria to quantify the characteristics of reactor aging.

RISK DATA SYSTEM

This INEL mission provides information to improve Probabilistic Risk Assessments (PRAs) and to obtain information from PRAs to enhance the regulatory process. INEL researchers use licensee event reports and other data to identify basic causes of component failure in the safety systems of reactors and to identify risk-dominant sequences.



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BREEDER REACTOR TECHNOLOGY

The INEL is the leading center for experimental breeder reactor research in the U.S. The research is conducted at the Argonne National Laboratory-West (ANL-W) complex by the University of Chicago under contract to the Department of Energy-Chicago Operations Office.

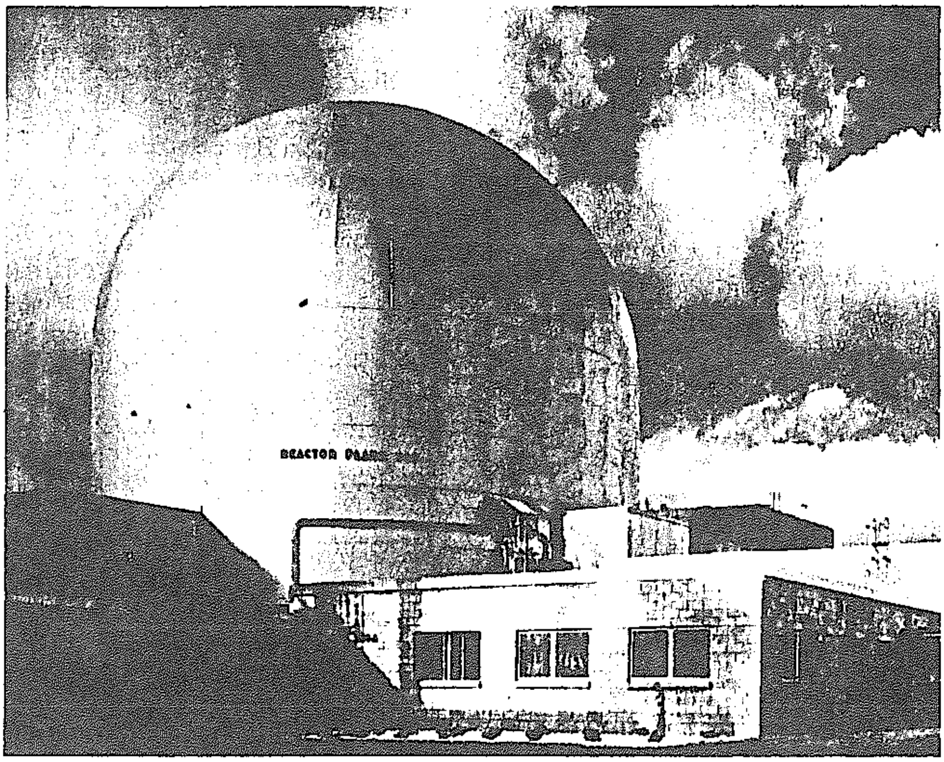
A breeder reactor is a nuclear power plant that, like other power facilities, produces heat for generating electricity. Unlike any other power plant, however, the breeder also produces new fuel—more than it uses—as it operates. In a breeder reactor, uranium-238, the nonfissionable but more abundant part of natural uranium, is converted into plutonium-239, a man-made fissionable element that can be used as a reactor fuel.

During the past several years, scientists from Argonne National Laboratory have developed an innovative design for an advanced nuclear power plant, called the Integral Fast Reactor (IFR). The plant appears to be inherently safe and less expensive to build and operate than previous reactors. It also has a closed, self-contained fuel cycle that appears to be economical and highly resistant to diversion.

Much of the technology for the IFR is based on the Experimental Breeder Reactor II (EBR-II), the first pool-type liquid-metal reactor. Metallic fuel was developed as the driver fuel in EBR-II. Technology developments of the past decade have radically improved the outlook for metallic fuels and a pyrochemical fuel cycle. For example:

- Metallic fuel can be designed now for very superior irradiation performance;
- Recent metallurgical processing discoveries and developments have radically altered both the process itself and the outlook for major breakthroughs in fuel and blanket processing;
- Accumulating evidence from relatively recent theoretical and experimental investigations indicates superior inherent safety characteristics for metallic fuel.

The Experimental Breeder Reactor (EBR-II), the first pool-type liquid-metal reactor



FACILITIES

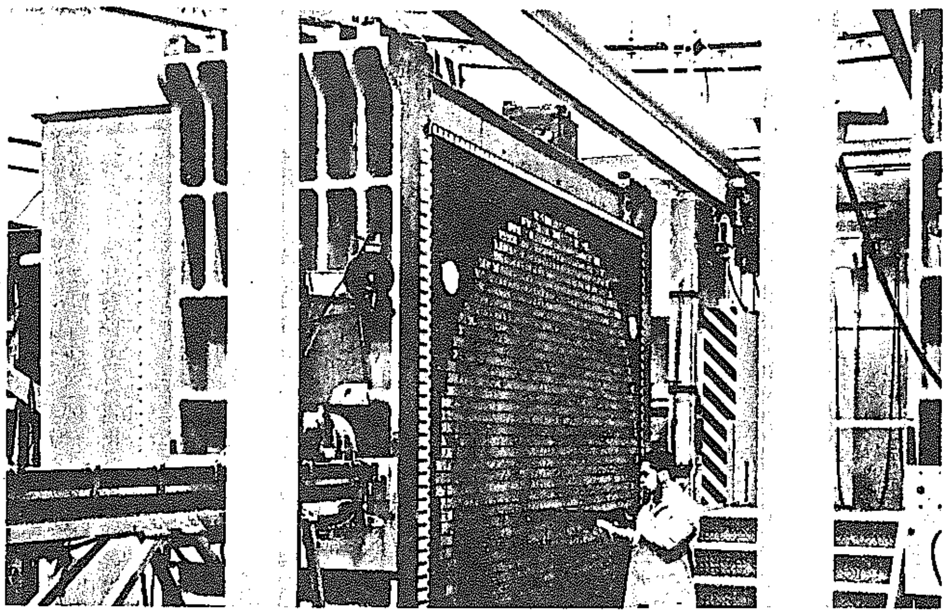
The ANL-W complex includes seven major operating areas, five reactors, and two fuel examination facilities.

Experimental Breeder Reactor II (EBR-II) was originally built to demonstrate the feasibility of operating a liquid-metal-cooled fast-breeder-reactor power plant with on-site fuel reprocessing. The facility also has been used to irradiate reactor fuel and structural material samples, testing their durability in breeder reactor environments. This information helps to improve fuel and material performance for future breeder reactors.

Operating as a power plant, EBR-II has generated more than two million megawatt hours of electrical energy through 1987. EBR-II power is fed to the INEL grid, supplying some 40 to 50 percent of the INEL's electrical power needs.

The Hot Fuel Examination Facility (HFEF) comprises two adjacent hot cell facilities. Each facility contains two heavily shielded hot cells, one with inert gas atmosphere (argon gas), the other with air atmosphere. Each facility also includes unshielded repair areas, laboratories, and equipment rooms. HFEF/North provides highly reliable alpha-gamma containment and protection. This facility can accommodate radioactive-material-filled casks weighing up to 50 tons and irradiated experiments up to 24 inches in diameter and 30 feet long. HFEF/South has similar capabilities for handling smaller components and assemblies. It is currently being modified to demonstrate pyrometallurgical reprocessing and refabrication of metallic fuels for liquid-cooled reactors. Operation is scheduled to begin in 1990.

HFEF/North offers a broad range of services for highly radioactive nuclear fuel assemblies and reactor components. These services include pretest and posttest handling and examination of irradiation experiments. Handling operations include remote assembly, disassembly, and reassembly of the irradiation experiments. Examinations include neutron radiography and tomography, precision gamma-scanning, profilometry and other dimensional measurements, weight determinations, visual and photographic examinations, eddy-current nondestructive testing, specimen gas sampling, and specimen metallographic examinations (including use of optical and scanning electron microscopes).



Technician loading fuel plates in one half of the ZPPR core

The Neutron Radiography Facility (NRAD), located in HFEF/North, is a nondestructive examination tool. Using two collimated neutron beams, produced by a 250-kilowatt reactor, NRAD produces neutron radiographs of internal conditions of highly irradiated test specimens without physically cutting into the specimen. Additionally, the reactor is used as a neutron source for isotope production, activation analysis, and radiation effects on materials.

The Zero Power Physics Reactor (ZPPR) provides reactor physics data for any type of fast neutron spectrum reactor, from tiny space-power reactors to huge breeder reactors. The reactor being studied is built in a large lattice framework that is split at the center. Each reactor is built full-size with the proper reactor fuels and other materials, so that extrapolation from the zero-power measurements to full-power conditions is readily achievable. Depending on the size of the reactor, the core, blankets, reflectors, plena, shields, and external reactor components and instrumentation can be loaded into the lattice framework. After the two lattice halves are driven together, ZPPR is brought to a low power, critical state by control rods. Heat removal is by air flow over the fuel elements. ZPPR's first nuclear operation was in 1969.

The Argonne Fast Source Reactor (AFSR) is used to calibrate instruments and to study fast reactor physics, augmenting the ZPPR research program. The facility began operating in 1959 as a neutron source used in developing improved instruments and techniques. AFSR has a design power of one kilowatt.

The Transient Reactor Test (TREAT) Facility is a uranium-oxide-fueled, graphite-moderated, air-cooled reactor designed to produce short, controlled bursts of nuclear energy. The purpose is to simulate accident conditions leading to fuel damage, including melting or even vaporization of test specimens, while leaving the reactor's "driver" fuel undamaged. Such tests provide data on fuel-cladding damage, fuel motion, coolant-channel blockages, molten-fuel/coolant interactions, and potential explosive forces during an accident. These data help in determining the consequences of accident conditions, refining computer simulations of reactor accidents, and, ultimately, designing reactors with greater inherent safety.

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TEST REACTOR AREA

The Test Reactor Area (TRA), operated by EG&G Idaho, is among the world's largest and most sophisticated materials testing complexes. The TRA complex houses extensive facilities for studying the effects of radiation on materials, fuels, and equipment.

ADVANCED TEST REACTOR

The Advanced Test Reactor (ATR) is located at TRA. The main purpose of the ATR is to simulate a prototypical reactor test environment for the study of radiation on materials.

ATR produces an extremely high neutron flux (up to 1×10^{15} n/cm^2/sec) for testing the durability of reactor fuels and materials. These tests determine how fuels and materials—relatively unaffected in more conventional environments—react when bombarded with streams of neutrons and gamma rays in high pressure and high temperature conditions. Data that would normally require years to gather from ordinary reactors can be obtained in weeks or months by using ATR's high neutron flux capability.

The 250-megawatt reactor core has a unique four-leaf-clover design. The four-lobed core delivers a wide variation of power levels to nine main test spaces. Each main test space, or loop, has its own separate environment apart from the main core of the reactor. This allows nine individual experiments to be conducted simultaneously. Additional smaller test spaces surrounding the loops allow even more experiments to be conducted independently.

In recent years, available space in the reactor has been utilized to produce radioisotopes to support the increasing needs of medicine, industry, and research. ATR has achieved new milestones in the production of the following radioisotopes:

— Cobalt-60, for use as a source for cancer radiation therapy and process irradiators

— Gadolinium-153, for use in dual-photon absorptiometry diagnostic equipment

— Nickel-63, for use in the production of "beta cells" for long-lived, more reliable power sources for control circuits

— Iridium-192, for use as an industrial radiography source.

ADVANCED TEST REACTOR CRITICAL FACILITY

The Advanced Test Reactor Critical (ATRC) Facility is a low-power (5 kW maximum power), full-size nuclear duplicate of the ATR, designed to provide physics data in support of the ATR program. Typically, the ATRC provides data for

— Reactivity effects of the insertion and removal of experiments using nuclear duplicates of the ATR tests

— Experiment void reactivities

— Capsule experiment reactivities using prototypes

— Thermal and fast neutron flux distributions

— Gamma heat generation rates

— Fuel loading requirements.

The ATRC also tests new ATR design and experiment concepts.

TEST REACTOR AREA HOT CELL FACILITY

The TRA Hot Cell Facility supports ATR and also Waste Management programs. Three hot cells within the facility are equipped with remotely operated machine tools, measuring instruments, and master-slave manipulators. This equipment is used for metallurgical study and irradiated sample testing.

NUCLEAR PHYSICS RESEARCH PROGRAMS

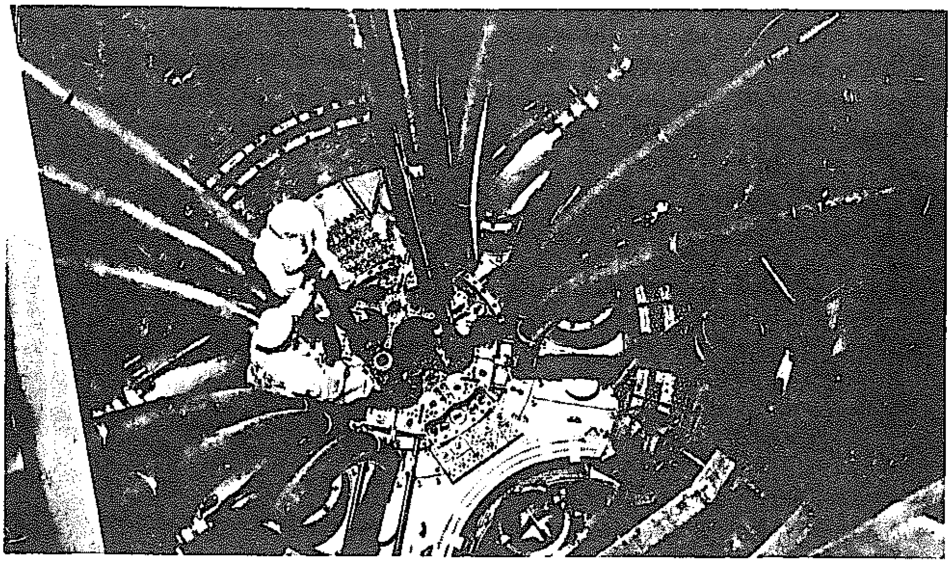
Special nuclear physics research programs are conducted for DOE at TRA laboratories to measure and evaluate the physics parameters required for advanced reactor concepts. These programs include

— Measuring the radioactivity and decay properties of radionuclides important to fission and fusion energy programs

— Participating in an internationally coordinated research program to improve the base of precise decay data for important radionuclides

— Developing improved techniques to measure the properties of radionuclides

— Evaluating data to incorporate in the Nuclear Data Sheets, which contain the recommended values for the decay properties of radionuclides.



The 250 megawatt ATR core

ADVANCED REACTIVITY MEASUREMENT FACILITY AND COUPLED FAST REACTIVITY MEASUREMENT FACILITY

The Advanced Reactivity Measurement Facility (ARMF) and the Coupled Fast Reactivity Measurement Facility (CFRMF) are two small pool reactors that are operated at maximum power levels of 100 kW and produce neutron flux levels up to 1×10^{10} n/cm^2/sec.

The ARMF and CFRMF reactors provide irradiation services for research and materials testing that include neutron radiography, nondestructive hydrogen determination, uranium-235/uranium-238 fuel assay, fast neutron damage, activation analysis, and reactivity measurements.

RADIATION MEASUREMENTS LABORATORY

The Radiation Measurements Laboratory (RML) is a modern, well-equipped radioanalytical laboratory specializing in qualitative and quantitative measurements of alpha, beta, gamma, and neutron radiation. It develops and uses state-of-the-art radiation measurement instrumentation and analysis techniques to support a broad variety of INEL programs.

RADIOCHEMISTRY LABORATORY

The EG&G Idaho radiochemists at TRA are responsible for radioanalytical support to the Radiation Measurement Laboratory (RML) and numerous independent R&D areas as well.

Investigations are under way to produce and purify medical radioisotopes, study the effects of radiation on hazardous wastes, and partition actinides from highly radioactive fission product wastes for long-term management of nuclear wastes. In addition, custom analysis of such interesting radioisotopes as Iodine 129, Telerium 141, Strontium 141, and environmental level alpha emitters inspires the development of improved techniques at this facility. Nine laboratories and a special chemical hot-cell cave provide facilities for these efforts.

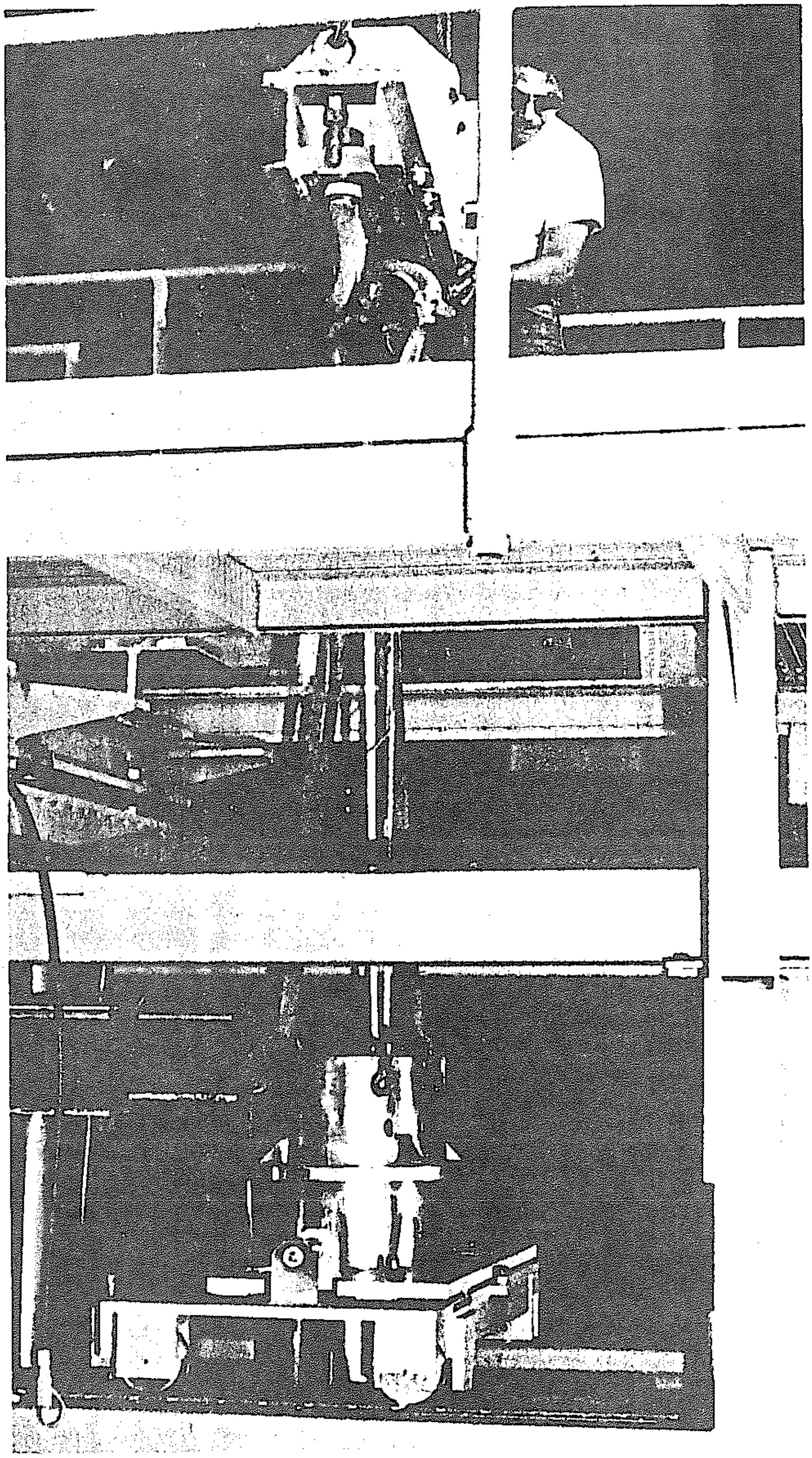
THE POWER BURST FACILITY (PBF)

Located southeast of TRA, the Power Burst Facility (PBF) is a one-of-a-kind facility. It is the only reactor in the world that can perform rapid power changes on the order of milliseconds, perform severe rod fuel burst tests, and simulate loss-of-coolant accidents within a special assembly that fits inside the main reactor core.

The initial mission for PBF was in the testing of light water reactor fuel rods under representative accident conditions. Data from these tests were used to develop and validate fuel behavior computer codes for the Nuclear Regulatory Commission. These tests were completed in 1985, and the reactor was placed on standby status pending new programs.

Although officially on standby status, the PBF reactor, because of its unique capabilities, is currently being considered for use in several defense-related programs and for use in a brain cancer treatment program called Boron Neutron Capture Therapy (BNCT). The BNCT Program would be used to treat patients with glioblastoma multiforme—a form of brain cancer that kills approximately 4,000 people per year in the United States.

BNCT uses neutrons to bombard and kill cancerous cells. The treatment requires an intense source of epithermal neutrons, or neutrons in between the low-energy level produced by thermal reactors and the high-energy level produced by fast reactors. With modification, the PBF reactor could be made to produce epithermal neutrons to treat patients in approximately 10 minutes, and in sufficient quantity to treat hundreds and possibly up to thousands of patients per year.



PBF is a one-of-a-kind facility

IDAHO NATIONAL ENGINEERING LABORATORY

IDAHO CHEMICAL PROCESSING PLANT

Since 1953, the Idaho Chemical Processing Plant (ICPP) has recovered uranium from spent (used) nuclear fuel assemblies, largely from government-owned reactors. The fuel processed to date totals more than 24,300 kilograms of uranium-235 valued, at today's prices, at more than one billion dollars. Secondary ICPP functions are to recover valuable rare gases and develop improved fuel processing and waste management methods. Westinghouse Idaho Nuclear Company (WINCO) operates the ICPP for the Department of Energy Idaho Operations Office and employs more than 1400 people.

The ICPP was the first facility in the free world to

— Reprocess highly enriched pure uranium on a production basi;

— Process breeder reactor fuels, such as the EBR-I Core I

— Dissolve spent nuclear fuel assemblies continuously on a routine schedule

— Use fixed and soluble neutron poisons for criticality control

— Store high-level acidic liquid waste in stainless steel tanks

— Solidify (calcine) highly radioactive waste liquids on both plant and production scales

— Store solidified, high-level radioactive waste in stainless steel storage bins

— Operate a dry storage facility for irradiated fuels

— Operate a radioactive rare gas recovery plant in conjunction with a fuel recovery process

— Show that multiple head-end processing of various fuels is compatible with common second- and third-cycle purification

— Process graphite-base fuels as a demonstration.

ICPP facilities include spent fuel storage and reprocessing areas, a waste solidification facility, calcine storage bin sets, a waste disposal building, remote analytical laboratories, service waste percolation ponds, and a coal-fired steam-generating facility providing process and space heat.

URANIUM RECOVERY

When a reactor is operated, uranium atoms in the fuel split, or fission, to create energy. Fissioning also creates radioactive waste products (fission products) inside the fuel elements. After a time, but before all the uranium atoms are consumed, the radioactive fission products build up, causing inefficient use of the fuel. At this point, spent fuel elements are removed from the reactor, and new fuel elements are installed. Test reactor fuel elements may be replaced when only 25 to 35 percent of the fissionable uranium has been spent. The valuable unused uranium is recovered from the spent fuel and used in new fuel elements.

ICPP operations recover uranium from reactor fuels encased, for example, in aluminum or stainless steel alloys or graphite. Radioactive waste products in the fuel are separated from the uranium during processing.

The ICPP processes highly enriched fuels—those where the fissionable uranium-235 isotope concentration has been artificially increased from the 0.7 percent found in natural uranium ore to between 20 and 90 percent.

Plant operations also help recover the valuable inert gas, krypton-85, which is used in many nonnuclear industries, such as electronics for flaw detection.

The processing sequence begins when spent government reactor fuel assemblies are brought in shielded casks to the ICPP via rail or truck. The assemblies are removed from the casks for underwater storage in fuel receiving and storage stations. Graphite-matrix fuel assemblies and liquid-metal reactor assemblies are stored above ground or below ground in special dry storage facilities.

Following storage and cooling, the fuel assemblies are transferred to fuel processing operations. Here, the assemblies are dissolved in liquid acid solutions containing nuclear poisons for criticality control. The acid type depends on the fuel and its cladding. The ICPP has multiple processes to dissolve several fuel types.

Once dissolved, the fuel, in a now aqueous solution, passes through an organic solvent, which selectively extracts the uranium and separates it from the radioactive fission products. The uranium is then stripped from the organic solvent by a different aqueous solution and further purified by passing through two additional solvent-extraction cycles using another organic solvent.

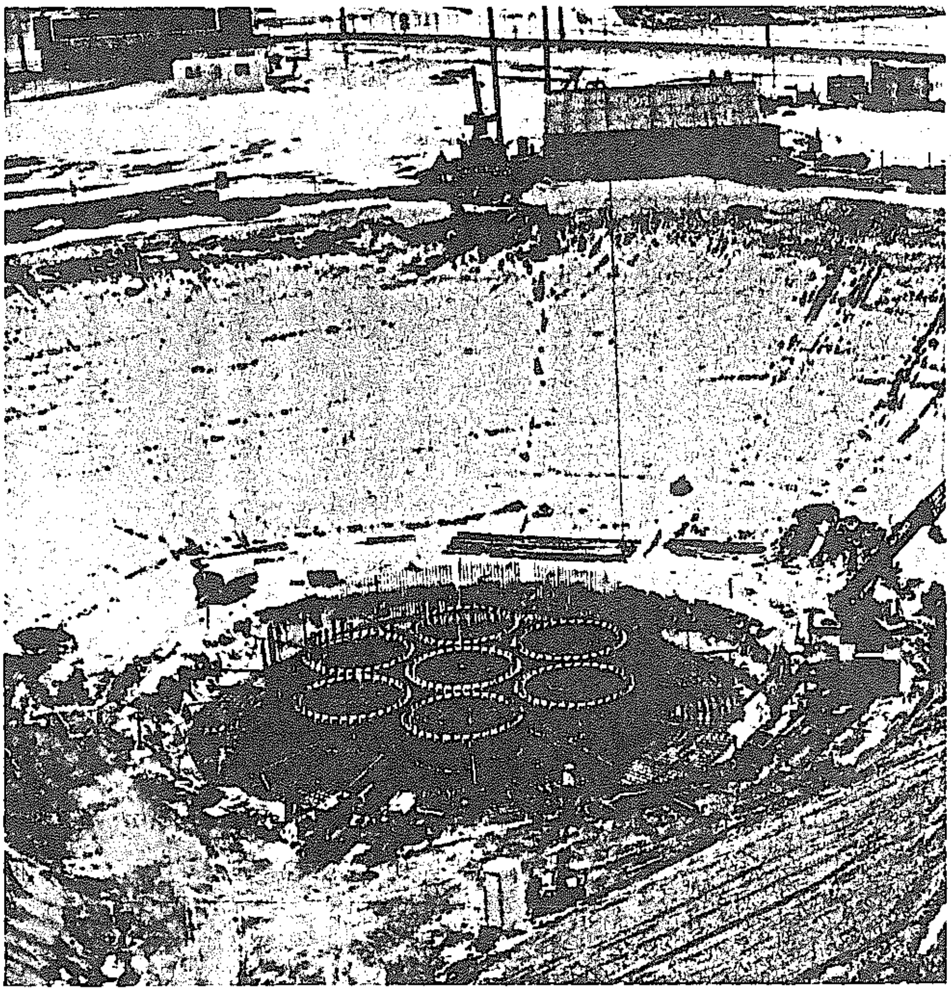
The purified uranium product is uranyl-nitrate solution, essentially free of all fission products and other impurities. The uranyl-nitrate is denitrated to uranium-oxide powder and transported to government facilities at Oak Ridge, Tennessee. Further processing follows, and the uranium is ultimately remanufactured into reactor fuel.

ICPP FACILITIES

WASTE CALCINATION

Since 1963, the ICPP has processed highly radioactive acidic liquid waste from its operations into a safer solid form requiring about one-eighth the storage space. The first Waste Calcining Facility, originally designed as a pilot demonstration plant, operated as the nation's first plant-scale production facility for more than 18 years. During that time, more than 4 million gallons of waste liquid were converted into solid form.

The process begins with the waste solutions being sprayed as a fine mist into a chamber containing heated granules about the size of coarse sand. A controlled airflow through the material has the effect of "fluidizing" and circulating the granular bed within the chamber. Heat evaporates the water and deposits dissolved aluminum and fission-product nitrates as coatings on the granules. As the particle size increases, the violent agitation continuously chips small fragments off the granules. Some fragments are carried as fine dust into the calciner's off-gas cleanup system while some remain in the chamber to form seed for new granules.



Calcined solid storage site under construction

The New Waste Calcining Facility became operational in 1982, and is the nation's first full-scale facility to convert highly radioactive liquid waste to solid form. In addition to increasing the ICPP's radioactive waste management capabilities, this facility incorporates new technologies in the areas of fluidized-bed calcination, off-gas cleanup, remote operations, and decontamination methods.

Solid, nonfragmented granules are carried by airflow through a shielded underground tube into stainless steel storage bins located inside concrete vaults.

There are six sets of calcine bins, and a seventh set is nearing completion. New storage facilities are needed about every three years to meet production requirements. All calcine is retrievable to allow further processing or offsite transport. Research continues to evaluate the feasibility of various compaction and encapsulation alternatives to provide permanent storage.

The ICPP is not intended to serve as a permanent high-level waste storage facility. Eventually, the waste will be moved to a federal repository.

FLUORINEL DISSOLUTION AND FUEL STORAGE FACILITY

The new Fluorinel Dissolution and Fuel Storage Facility (FAST) augments and improves existing and aged ICPP operations. The multi-level, 120,000-square-foot facility receives, stores, and dissolves used nuclear fuel assemblies from government-owned reactors. The uranium recovery payback is expected to cover the facility's construction costs by the early 1990s.

Fuel casks, brought to FAST by rail or truck, are first cleaned and then put in an unloading pool. While underwater, the casks are opened and fuel assemblies are removed by cranes and transferred into one of six fuel storage pools. While stored, decay heat from fission products inside the fuel assemblies is released and captured by facility processes to help heat the building. When ready for dissolution, the fuel assemblies are transferred underwater to the dissolver cell.

Once dissolved, the aqueous solution is piped to other ICPP process areas where the usable uranium is chemically extracted from the fission products (see Uranium Recovery section).

FAST operations employ versatile and remotely replaceable equipment throughout. This maintenance system keeps employee radiation exposure low and reduces facility downtime.

REMOTE ANALYTICAL LABORATORY

The Remote Analytical Laboratory (RAL) at the ICPP incorporates the most modern and sophisticated analytical chemistry techniques to support analyses of process operations at the ICPP. The laboratory complex includes a hot cell for analyzing highly radioactive substances. The cell is 50 feet long, 20 feet wide, and has 17 master/slave manipulators and 9 shielded viewing windows.

Encapsulated samples from processing facilities are transferred to the RAL via an air-actuated pneumatic transfer system at speeds up to 50 feet per second. This rapid sample transfer system minimizes the time between taking the samples and performing the analyses while reducing radiation exposure to workers. Results from the analyses are relayed between the RAL and the operating facility by interactive computer systems.

The new RAL replaces an earlier analytical chemistry facility whose useful function had become limited after more than 30 years of service.

FUEL PROCESSING FACILITY

Construction has started on the final facility necessary to bring about a complete upgrade of the ICPP. Excavation for the Fuel Processing Facility began in April 1986. Completion of construction is now scheduled for 1993. The Fuel Processing Facility (FPF) will increase uranium recovery by more than a factor of four over present capability. The FPF also will allow increased on-stream availability through remote maintenance and equipment replacement and will permit more positive control of effluent. Safety, security, and rigorous design standards are being incorporated into the design and construction of the FPF.

SPECIAL ISOTOPE SEPARATION PROJECT

The ICPP has been chosen as the DOE's preferred site for the new Special Isotope Separation (SIS) Project. SIS would use high-technology lasers to remove unwanted isotopes from government stocks of fuel-grade plutonium. An official decision concerning the building of SIS is expected in late-1988.

IDAHO NATIONAL ENGINEERING LABORATORY

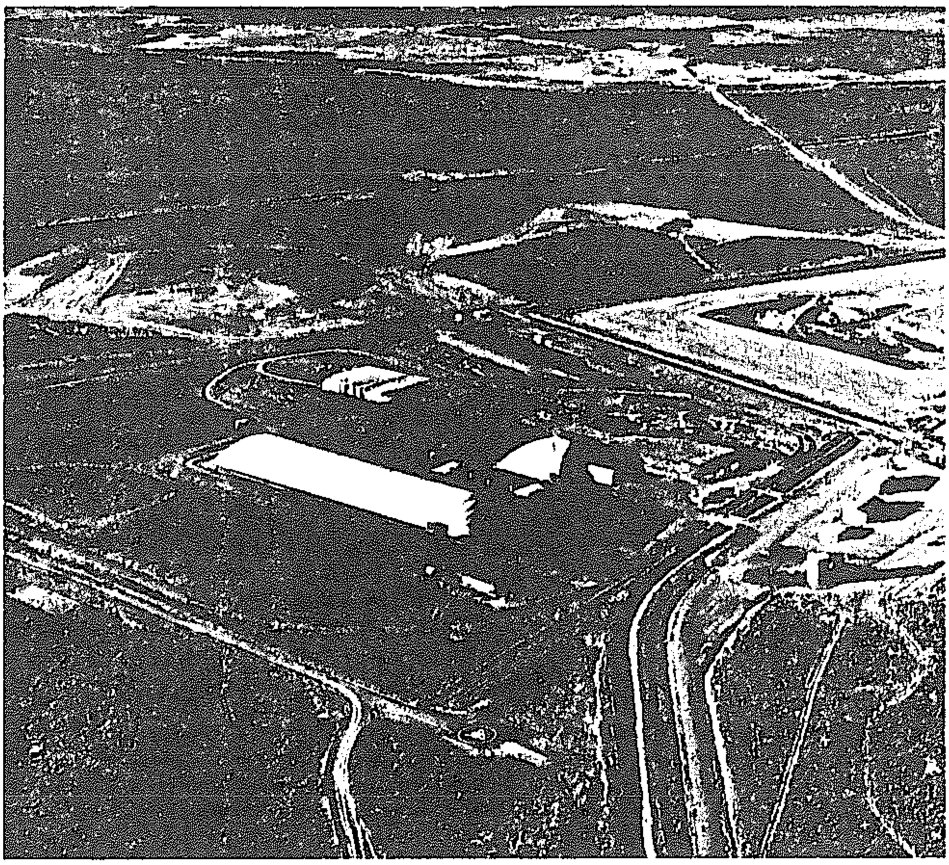
WASTE MANAGEMENT PROGRAMS AT THE INEL

Various strategies of waste storage, processing, and disposal are studied by Waste Management personnel in support of the research and development activities at the INEL. EG&G Idaho operates the facilities built to manage the wastes housed at the INEL for the DOE Idaho Operations Office.

The RWMC receives two basic types of solid radioactive waste:

— Transuranic (TRU) waste is contaminated with transuranic nuclides such as plutonium-239, an alpha emitter, and have a half-life of more than 20 years. TRU waste at the RWMC comes from national defense and research programs at the INEL and other DOE laboratories.

— Low-level waste is contaminated with beta- and gamma-emitting radionuclides. This waste generally has low radiation levels and decays to near background levels in a relatively short time. Low-level waste currently is received at the RWMC almost exclusively from INEL operations.



Aerial view of Radioactive Waste Management Complex

RADIOACTIVE WASTE MANAGEMENT COMPLEX

Storage and disposal of solid radioactive waste generated in national defense and research programs are major areas of work at the Radioactive Waste Management Complex (RWMC).

TRANSURANIC WASTE MANAGEMENT

Solid TRU waste, which includes such items as cloth, paper, plastics, metal, rubber, sludge and concrete, is stored at the RWMC on above-ground asphalt pads or in steel pipe vaults.

Since 1970, TRU waste has been stored on asphalt pads at the Transuranic Storage Area (TSA). The waste is contained in 55-gallon steel drums, fiber glass-covered wooden boxes, or steel bins. The containers are stacked on the pads in cells separated by earthen walls and covered with plywood, tough polyvinyl plastic, and two to three feet of soil. During storage operations, a pad is covered with an air support building to protect workers and the waste containers from weather until the pad is filled and the waste is covered with soil.

The TSA has two pads, each measuring 150 by 750 feet. The first pad was filled and closed in 1975.

TRU waste that must be remotely handled is placed in steel pipe vaults at the Intermediate-Level Transuranic Waste Storage Facility within the TSA. About one percent of the INEL-stored TRU waste is placed in these vaults.

A major goal of the DOE radioactive waste management program is to remove all retrievable TRU waste from the INEL and place it in a federal repository. Starting in 1989, INEL TRU waste is scheduled to be transported to the federal Waste Isolation Pilot Plant (WIPP) in New Mexico for permanent disposal. Before TRU waste can be transported from the INEL, it must be retrieved and examined to verify that the waste and the waste containers meet WIPP waste acceptance criteria.

Two experimental pilot plant facilities have been constructed at the INEL to prepare INEL TRU waste for acceptance at the WIPP.

The Stored Waste Examination Pilot Plant's (SWEPP) primary objective is to nondestructively examine and certify INEL-stored TRU waste for WIPP. SWEPP is the first U.S. facility performing such examination and certification. Examination of waste and containers includes weighing, real-time radiography examination, neutron assay, container integrity examination, and radiological surveys. Waste and containers certified for disposal at WIPP are in interim storage until transportation starts.

Waste containers not meeting WIPP waste acceptance criteria are sent to the Process Experimental Pilot Plant (PREPP) located at the INEL Test Area North. Here, uncertified waste undergoes waste container shredding, rotary kiln incineration, and product immobilization in a grout mixture. The final product, a cemented 55-gallon drum, is then sent to SWEPP for neutron assay and certification, and finally sent to WIPP for permanent disposal. Processing of other waste forms, including hazardous and classified materials, is also being demonstrated by PREPP.

LOW-LEVEL WASTE MANAGEMENT

Existing since 1952, the RWMC has developed technology to safely dispose of the low-level waste (LLW) generated by the INEL.

LLW primarily consists of protective clothing, paper, rags, packing material, glassware, and contaminated equipment. These materials are either contaminated with radioactive nuclides or contain radioactive activation products from nuclear reactions. Most of these materials have low radiation levels, normally around 20 mR/h or less, at three feet from the container surface.

An education and training program is in place at the INEL to help minimize the LLW at the points of generation. Waste is segregated and separated into various categories to best utilize disposal and processing operations.

When received at the RWMC, LLW shipments are surveyed for contamination, radiation levels, and compliance with approved packaging criteria for acceptance. The waste containers are then placed in the Subsurface Disposal Area or in soil vaults depending upon the waste container and radiation level.

As part of a greater confinement technology program, the RWMC is actively pursuing a contaminated-soil grout-development program. This program will improve disposal space efficiency and stabilize LLW disposal sites as they are filled. Completion of this program is expected to extend the life expectancy of the area, minimize the migration of radionuclides, and reduce the maintenance efforts required for unstabilized areas.

To further extend the disposal life of the RWMC, a concerted effort has been made to identify and develop methods to reduce the volume of LLW at the INEL.

The Waste Experimental Reduction Facility (WERF) has been established at the PBF area, nine miles northeast of the RWMC, to process LLW. It is the most comprehensive reduction-scale treatment facility in the DOE system.

At WERF, four methods of volume reduction are used:

— Contaminated metal is sized at a rate of several tons per year to reduce the volume it occupies. The volume reduction averages a rate of 4 to 1.

— Contaminated metal is melted and cast into ingots. This volume reduction is 4 to 1.

— Noncombustibles are compacted, achieving a volume reduction of 5-10 to 1.

— Contaminated combustibles are incinerated, achieving a volume reduction of over 200 to 1.

Another process developed at WERF is the ash solidification process. Grout is added to the ash from the incinerator to stabilize the end product. Waste Engineering Development is developing methods to stabilize unusual waste forms and mixed hazardous waste.

NATIONAL LOW-LEVEL WASTE MANAGEMENT

The INEL is the lead laboratory for the DOE Low-Level Waste Management Program, which provides management and technical assistance to support the disposal of low-level radioactive waste. Established in 1979, the National Low-Level Waste Management Program divides activities into two areas, Defense and Nuclear Energy.

The Defense Program provides a central organization to assess DOE-wide needs, coordinate technology development activities to meet those needs, and promote the use of improved technology at DOE facilities. Specific program objectives are to ensure operation and maintenance of active and inactive DOE low-level waste disposal sites while protecting public health and safety to meet applicable criteria and standards for isolation of low-level waste from the biosphere.

The mission of the Nuclear Energy LLW Management Program is to assist states and regions in establishing an effective nationwide LLW management system. Program objectives are

— To provide technical assistance to states and regions;

— To manage assigned responsibilities for surcharge rebates and unusual volume allocations and provide oversight for LLW management system progress;

— To develop recommendations and plans for ensuring the safe disposal of Greater-than-Class C LLW.

The technical assistance tasks provide technical expertise, information, technology development, and other technical resources to states and regions to support their development of LLW management facilities. The Greater-than-Class C LLW disposal tasks provide for preparation of DOE's recommendations and plans for ensuring the safe disposal of Greater-than-Class C LLW. The Program Integration tasks provide for management, administration, planning, and coordination of Program activities.

SPENT FUEL PROGRAMS

Spent Fuel Programs comprise several research and development programs for spent fuel management being conducted by the DOE at the INEL. The Nuclear Waste Policy Act of 1982 directed DOE to establish a national system for disposal of commercial spent fuel and high-level radioactive waste. The Office of Civilian Radioactive Waste Management was established within the DOE to implement the act. The purpose of Spent Fuel Programs is to demonstrate the feasibility of packaging, transporting, storing, and consolidating aged spent fuel in support of DOE's program for permanent disposal.

The Spent Fuel Storage Cask Testing Program has demonstrated dry-storage cask performance for spent fuel from pressurized water reactors. This work is being performed for the DOE with some activities sponsored under a cooperative agreement with Virginia Power and the DOE.

The objectives of the Program are

— To test and demonstrate dry storage casks containing either unconsolidated or consolidated fuel rods, using a range of cask designs and materials;

— To provide loaded cask performance data that can be used by utilities to facilitate licensing at reactor dry storage installations and that can be used by the NRC for licensing dry storage by generic rule.

The Test Area North (TAN) Hot Shop is used to receive the shipping casks containing spent fuel and to transfer the fuel into storage casks by remote handling. The storage casks are then moved into the TAN Warm Shop for testing of containment, shielding, heat transfer, and design. After the testing is completed, the casks are transferred to a newly constructed test pad near the Hot Shop for long-term monitoring. The monitoring and testing will provide data on the condition of the casks and spent fuel during long-term outdoor storage.

In conjunction with the Cask Testing Program, a Dry Rod Consolidation Technology (DRCT) Project has been completed at the INEL. The DRCT developed and operated semiautomatic remote equipment designed to demonstrate the dry horizontal consolidation of 15 x 15 Pressurized Water Reactor (PWR) spent nuclear fuel assemblies. Forty-eight PWR fuel assemblies were consolidated into twenty-four canisters. The consolidated canisters are being utilized in the DOE Spent Fuel Storage Cask Testing Project to determine cask performance with consolidated fuel. The data collected will directly benefit the Prototypical Consolidation Demonstration Project (PCDP).

The PCDP will demonstrate production-scale rod consolidation of commercial fuel assemblies. The rod consolidation equipment developed during this project will provide a standard for future selection of licensable equipment to be used at high-level waste repositories or a Monitored Retrievable Storage (MRS) facility. This project will consolidate approximately 200 fuel assemblies (approximately 100 Pressurized Water Reactor and 100 Boiling Water Reactor assemblies) in a hot demonstration. The private sector has been involved in the design of the hardware for the consolidation process.

HOT CELLS

The Hot Cells are located at Test Area North (TAN) and at TRA. Operators use specialized tools and state-of-the-art technology for the disassembly and detailed analyses of irradiated materials.

The Hot Shop at TAN is the world's largest hot cell, measuring 51 feet wide by 165 feet long by 55 feet high. Its 7-foot-thick concrete walls and 6-foot-thick windows afford protection to personnel involved in the examination, handling, analysis, or disassembly and assembly of highly radioactive or contaminated assemblies, including reactor systems. Recently, the Hot Shop was utilized for the transfer of spent fuel from the shipping casks to the storage casks.

Located within the same building and immediately adjacent to the Hot Shop are several smaller hot cells and other facilities that supplement the capabilities of the Hot Shop. The TAN Hot Cell is the facility where Dry Rod Consolidation hardware was recently tested. Within the Hot Cell, all work was performed remotely and recorded on video cameras for data collection.

Another specialized feature of the TAN Hot Cells is a Scanning Electron Microscope (SEM). In addition to providing images magnified up to 300,000 times, the SEM can perform both qualitative and quantitative elemental analyses on atomic numbers down to oxygen.

The Water Pit has become the interim storage area for several research and development activities at TAN. Fuel skeletons from the Dry Rod Consolidation Project are stored in the Water Pit. Another specific use is the storage of irradiated materials taken from the Three Mile Island Unit 2 (TMI-2) accident cleanup effort. An estimated 125,000 cubic feet of core debris will be placed in stainless steel canisters at TMI-2 and transported to the INEL for research and storage in the Water Pit. Many samples of the Three Mile Island materials were also examined in the Hot Cells. Data were collected to help determine the composition of the core after the accident.

A Warm Shop, immediately adjacent to the Hot Shop, is used for direct-contact handling of assemblies with low to medium radiation or contamination. Railroad tracks connect this shop to the Hot Shop and other facilities at the TAN area. The Warm Shop is the site for much of the testing and monitoring of the spent fuel storage casks.

The TRA Hot Cells are used primarily for beta-gamma radiation work, e.g., radioactive metallography. One cell contains specialized mounting, grinding, polishing, cleaning, and etching equipment. Other equipment includes two shielded metallographs, monocular and stereoscopic periscopes, thin sectioning saws, and vacuum impregnation equipment.

CASK SYSTEMS DEVELOPMENT PROGRAM

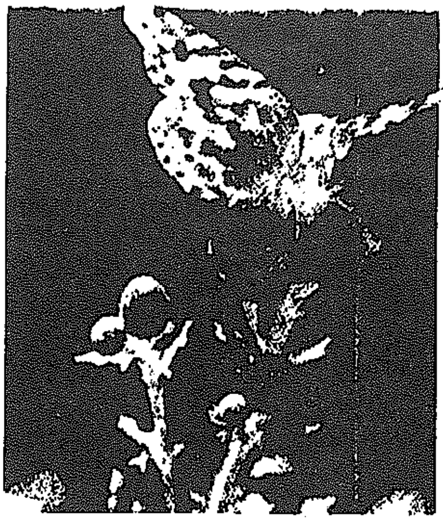
Cask Systems Development Phase I includes the development of prototypical spent fuel and radioactive waste casks required by the DOE for civilian radioactive waste transport. Three initiatives will allow the DOE to ensure the availability of certified casks for the following uses:

- Shipments of spent fuel and consolidated fuel rods from most reactors to facilities in the federal disposal system
- Shipments of spent fuel between an MRS facility and a repository in an integrated disposal system
- Shipment of nonstandard spent fuel and nonfuel materials from reactors or other facilities to federal disposal facilities.

These cask initiatives are respectively referred to as From-Reactor Casks, From-MRS-to-Repository Casks, and Nonstandard Spent Fuel and Components Casks. The term "cask" includes not only the packaging or cask, but also the tie-downs, vehicular conveyance, and associated ancillary equipment.

Cask Systems Development Program (CSDP) Initiative 1 consists of procurement action from private industry which will result in a family of NRC-certified prototype casks. The activity will include cask preliminary design, detailed design and engineering, test model and prototype fabrication (trailer and railcar included), quality assurance inspection, component testing, certification by the NRC, and startup and acceptance testing of the configuration. The activity will be performed for at least two legal-weight truck-transported cask designs. Three designs will be completed for casks which can be transported by rail or barge.

Spent-fuel casks developed under this program will be certified by the NRC. The cask contractor will be responsible for obtaining a valid Certificate of Compliance on behalf of DOE from the NRC for the cask being developed before the prototype cask will be fabricated. CSDP will provide assistance in resolution of broad technical issues that may arise during the design certification process through its procedural agreement with the NRC and through applied-technology tasks sponsored by the DOE Office of Civilian Radioactive Waste Management.



NAVAL REACTORS FACILITY

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Four major installations make up the Naval Reactors Facility (NRF). These are the Submarine Prototype (S1W), the Large Ship Reactor (A1W), the Natural Circulation Submarine Prototype (S5G), and the Expended Core Facility (ECF). NRF is operated for DOE and the U.S. Navy by Westinghouse Electric Corporation under jurisdiction of DOE's Pittsburgh Naval Reactors Office.

It was in the S1W, originally called the Submarine Thermal Reactor, or STR, that the United States Nuclear Navy was born. The project, aimed at freeing naval vessels from their need for refueling at sea or frequent returns to port, achieved success with an initial power run in the Nautilus prototype on May 31, 1953.

A subsequent test simulated a nonstop voyage from Newfoundland to Ireland, submerged and at full power. This proved atomic propulsion of ships was feasible and that the *Nautilus*, long before it set out to sea, could do remarkable things, such as a later accomplishment of subnavigating the polar cap from the Pacific to the Atlantic.

The next logical step was to develop a prototype for surface ships. However, other problems, including the necessity of proving reactors can be teamed up to drive one turbine, remained to be solved. In the late 1950s, the A1W attained criticality and full-power operation of both reactors. The aircraft carrier *Enterprise* and the missile cruiser *Long Beach* were the first ships powered by the A1W-type plants. One of the A1W reactor plants was modified in 1972 to provide test and prototype operation of a new-type reactor design, which has since been used in the newest aircraft carriers like the *Nimitz*, *Eisenhower*, and *Vinson*.

In addition to the major technological studies made at NRF, the three Nuclear Prototype Plant Facilities (S1W, A1W, S5G) are used to provide a comprehensive nuclear plant operational training program. Numerous naval officers and enlisted personnel have received training in operation of nuclear power plants at this site, and continuing training will provide personnel to operate the current and future naval nuclear-powered ships.

S1W PROTOTYPE PLANT

Started in 1953, the S1W Facility still houses the prototype of the *Nautilus*, although the testing program has changed from simulating the *Nautilus* power plant to testing advanced design equipment with prototypes of new systems for current nuclear projects to obtain data for future naval vessel power plants.

A1W PROTOTYPE PLANT

The A1W is a prototype facility consisting basically of a dual pressurized water reactor plant within a portion of a steel hull, built to simulate the aircraft carrier *Enterprise*. All components could withstand seagoing use.

Operational since 1958, this nuclear plant is the first to have two reactors powering one ship propeller shaft. The prototype powers the plant's propeller shaft through a single-geared turbine propulsion unit. New advanced cores and equipment have replaced many of the original components. This plant has the capability of, and is presently operating with, reactor plants of two different reactor designs operating independently. Thus, flexibility and operational experience are provided for both testing purposes and training naval personnel.

S5G PROTOTYPE PLANT

First operated in 1965, the S5G is a prototype pressurized water reactor that can operate in either a forced circulation or a natural circulation flow mode where cooling flow through the reactor is caused by thermal circulation rather than by pumps. Using natural circulation improves plant safety, simplifies plant design, increases reliability, and reduces noise level.

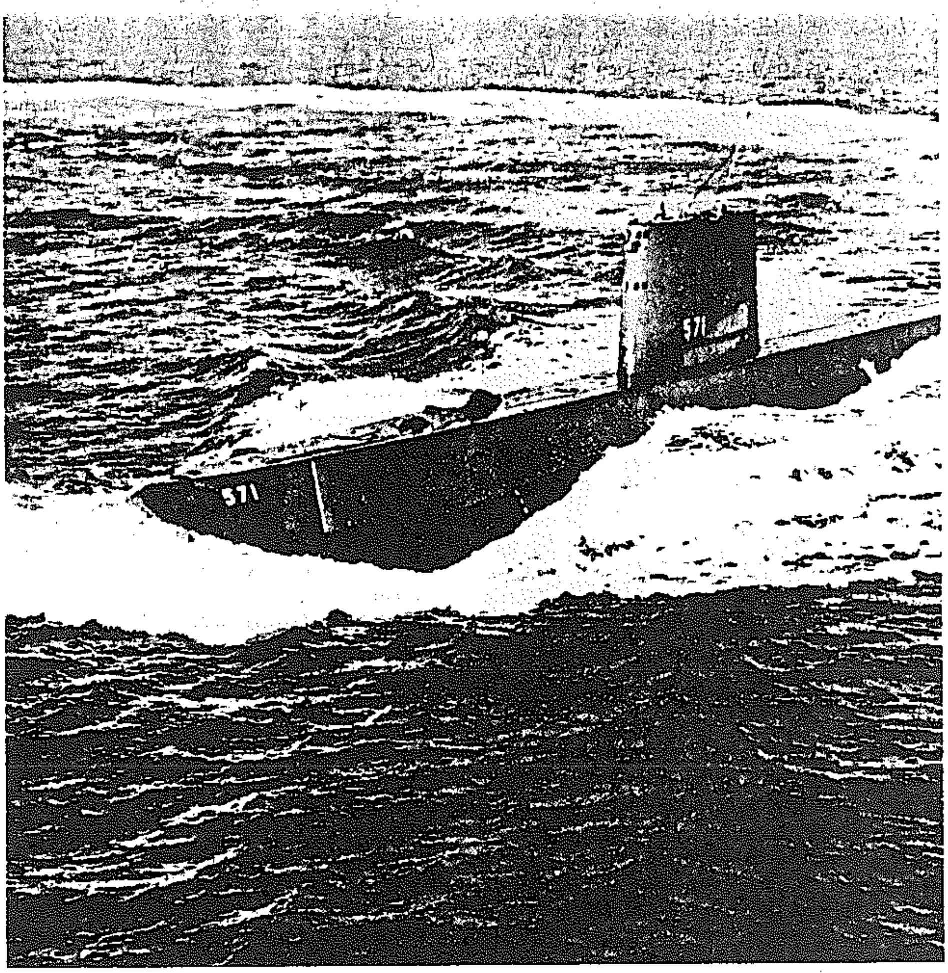
To prove that the new design concepts would work in an operating ship at sea, the prototype plant was installed in an actual submarine hull section capable of simulating the rolling motions of a ship at sea.

EXPENDED CORE FACILITY

The Expended Core Facility (ECF), also operated for DOE and the U.S. Navy by Westinghouse, receives, examines, and prepares naval expended cores for reprocessing.

Another ECF activity is to handle and examine irradiation tests (small-scale representations of current or future core designs) in the INEL's Advanced Test Reactor, providing test information for design personnel at the Bettis and Knolls Atomic Power Laboratories.

Part of the building contains deep, water-filled pits for safe underwater disassembly, examination, and preparation for analysis of radioactive components and irradiation tests. Portions of the disassembled components are sent to hot cells within the building for further examination and testing.



The United States Nuclear Navy was born at the INEL.

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INDUSTRIAL ENERGY CONSERVATION

The INEL Industrial Energy Conservation Program is aimed at finding new ways to help U.S. industries conserve fuel, improve productivity, and improve the nation's competitive position. Industry currently uses about 37 percent of the energy consumed in the U.S. According to some estimates, widespread adoption of successful technologies being developed through the Department of Energy Industrial Energy Conservation Programs at the INEL and elsewhere could conserve more than three quadrillion Btu's each year. This translates into a savings of about $12 billion annually.

The INEL program includes in-house research and development as well as contracting, planning, and coordinating programs performed by universities and industrial firms throughout the U.S. Specific activities include research and development for new waste heat recovery technologies, increased combustion efficiency, industrial heat pump technologies, new automated process control systems, advanced separation processes, waste products utilization, advanced iron and steel production and manufacturing technologies, advanced aluminum reduction and production technologies, and energy-integrated farm practices.

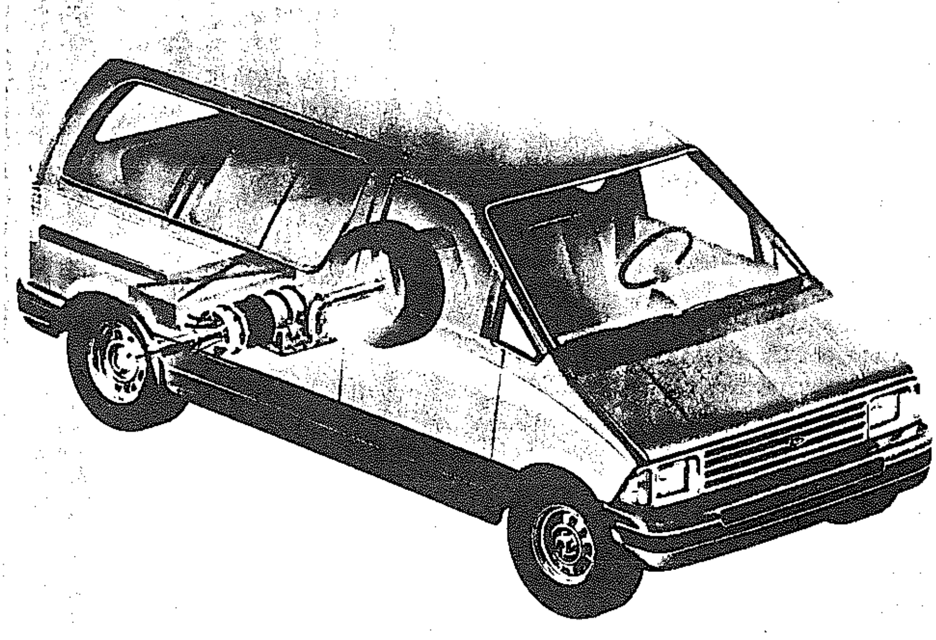
In addition, the INEL has been appointed lead laboratory for the Thermal Sciences portion of the DOE Energy Conservation and Utilization Technologies (ECUT) Program. Program development and research activities in heat and mass transfer as well as thermodynamics and fluid systems are currently under way. The INEL's newly installed CRAY-XMP/24 supercomputer is used extensively to support the analytical development that is a major part of the ECUT program. The INEL also has projects in the Materials and Biocatalysis areas of the ECUT program.

ELECTRIC VEHICLE

The purpose of the DOE Electric and Hybrid Vehicle Testing Program is to test the performance of various battery types and other equipment to help determine the future role that electric vehicles may play in the U.S.

INEL researchers work with the engineering divisions of major automobile manufacturers to test newly designed batteries and drive trains. A computer-controlled dynamometer is used in the laboratory to simulate actual driving conditions.

A prototype rendering of an electric van.



GEOTHERMAL

The INEL is a leading center for geothermal energy expertise and technology development. One major area of activity is the Injection Research Program, which is designed to provide a better understanding of the characteristics of geothermal reservoirs in underground fractured rock structures.

Other areas involve developing and collecting technical information for transfer to various state agencies and to the private sector, researching and developing advanced geothermal power plant concepts, and providing technical support to other government agencies in developing geothermal resources.

HYDROPOWER

The National Small Scale Hydroelectric Technology Development Program, administered by DOE-ID, provides technical assistance and cost-sharing for the development of small-scale hydroelectric projects throughout the U.S. Successful projects have been developed in Arizona, California, Florida, Idaho, Maine, Michigan, New Hampshire, New Jersey, New York, South Carolina, Texas, Vermont, Virginia, Washington, and Wyoming. These projects demonstrate the technical and economic feasibility of small-scale hydroelectric systems.

TECHNOLOGY TRANSFER

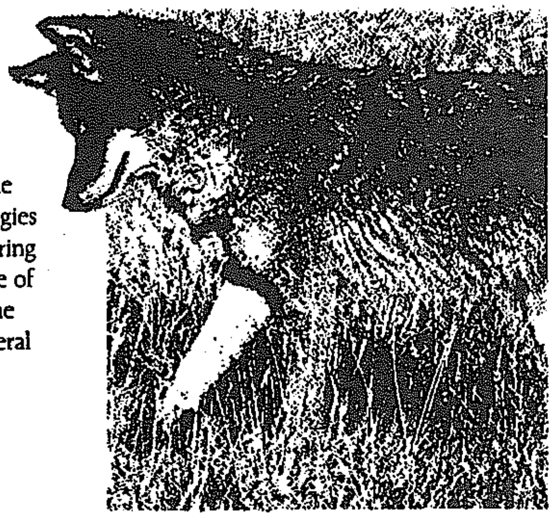
Federally funded research and development programs produce discoveries, products, and processes that have applications outside the confines of the laboratory. Part of the INEL mission is to ensure that these technologies are used by U.S. businesses and industries and by state and local governments. It is in the interest of national productivity and competitive position to encourage domestic businesses to use the resources of the federal laboratory system.

One of those resources is the Office of Research and Technology Applications (ORTA) at the INEL. ORTA coordinates activities that will lead to the greatest possible use of federal technologies in the private and public sectors by offering access to INEL technologies and to those of 300 other federal laboratories through the technology transfer network of the Federal Laboratory Consortium.

FOSSIL ENERGY

Advanced fossil energy research at the INEL is aimed at developing methods to process and effectively utilize the nation's vast fossil fuel resources. One primary area of research is microbial processing of coal to produce a liquid fuel and to reduce the environmental hazards associated with coal combustion. Materials research, which develops advanced materials for fossil applications, is also investigating techniques to fabricate thin films of superconducting ceramics.

Another area of research is the investigation of methods to effectively mine oil shale residues. A new research and development program in enhanced oil recovery has the goal of rapid and efficient expansion of domestic oil production in the event of an international oil crisis.



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CENTRAL FACILITIES AREA

Many services for the entire INEL site are located or headquartered at the Central Facilities Area (CFA). These include the DOE Radiological and Environmental Sciences Laboratory; Security; a site-wide communications system; the power distribution system; onsite railroad facilities; fire, medical, and ambulance services; warehouses; supplies and materials; cafeterias; equipment, vehicle, and labor pools; Site bus services; and a laundry for radioactively contaminated apparel articles.

HEALTH AND SAFETY REQUIREMENTS

DOE-ID is responsible for protecting the property, the environment, and the health and safety of people on or near the INEL. Elaborate safeguards are constantly in effect to minimize any possible hazard from Site activities. Nuclear safety overview, provided by DOE-ID, ensures that INEL nuclear reactors and nuclear facilities operate with the highest degree of safety.

Health and safety interests encompass all phases of safety, including industrial safety, industrial hygiene, fire protection, radiological safety, radiological shipments, environmental protection, reactor safety, nuclear criticality safety, and emergency planning. The program includes developing and enforcing safety policies and standards, conducting evaluations of contractors and facilities, performing review of new designs and safety analyses, providing radiological assistance, carrying out accident investigations, and supplying technical advice and consultation.

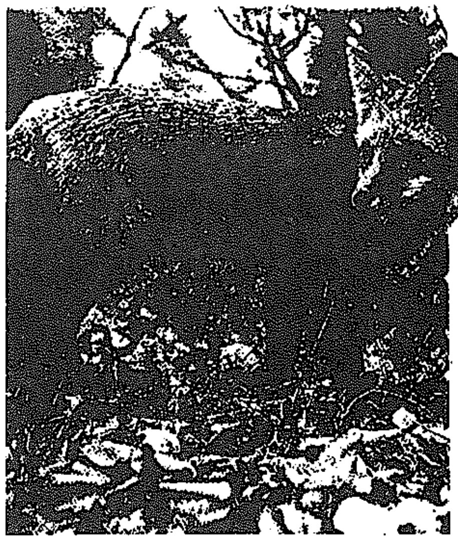
CONSTRUCTION MANAGEMENT

MK-Ferguson of Idaho Company, the INEL Construction Management Services Contractor under prime contract to DOE-ID, completed about 1,930 construction projects between 1979 and 1988. The projects varied in difficulty from the simple installation of concrete pads to the very complex demolition and rebuilding of a process cell. The costs of such projects range from a few hundred dollars to several million dollars.

RADIOLOGICAL ASSISTANCE PROGRAM

The INEL Radiological Assistance Program provides assistance if incidents involving radioactive materials occur in Colorado, Idaho, Montana, Utah, or Wyoming. Special teams, consisting of radiological monitoring personnel, can be dispatched immediately in response to calls concerning radioactive accidents. As part of a nationwide network, the teams can evaluate situations and recommend measures to control radiation hazards in the interest of the public safety.

Whole body counter used as part of individual health protection program



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SAFEGUARDS AND SECURITY

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Responsibility for the INEL's Safeguards and Security Program, an integral part of all INEL operations, is shared by every contractor at the INEL.

DOE Safeguards and Security, the Site protective force, and other INEL security personnel are responsible for protecting government property and classified information; processing and granting clearances and access authorizations for day-to-day operations and visitor controls; providing nuclear material accountability, control, and management; and maintaining a 24-hour Warning Communications Center.

Helicoptors are used for routine and emergency situations

All INEL access locations are controlled by guard stations to verify the identities and purpose of visits of both visitors and employees. Helicopters perform regular Site patrols and are used in security emergencies. No firearms, explosives, illegal drugs, or alcoholic beverages may be transported on INEL land. Cameras and recording devices are allowed only with previously written approval.

INEL trespassing restrictions are strictly enforced, and boundaries are clearly marked where public highways enter federally protected areas. The Federal Aviation Administration recommends that all non-INEL authorized aircraft flights do not fly lower than 2000 feet above ground level.

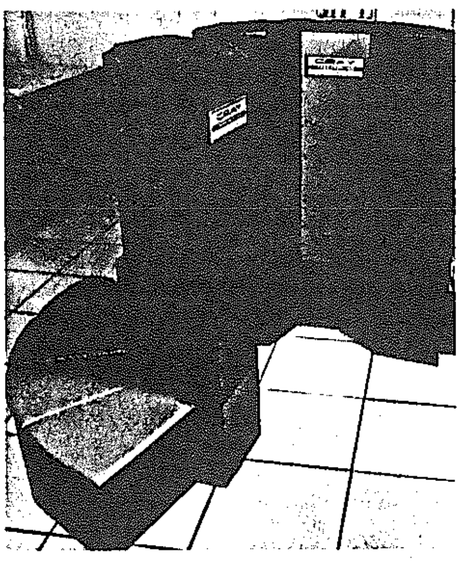
COMPUTER SCIENCE CENTER

The INEL Computer Science Center (CSC) is dedicated to supporting the scientific, engineering, and operations needs of the Department of Energy and other government agencies.

With the recent installation of the CRAY X-MP/24 for enhanced scientific computing and the IBM 3090 to perform business computing, the INEL now serves as one of the leading computing centers in the DOE family.

The INEL computing facility employs about 300 programmers, engineers, computer scientists, hardware designers, and systems analysts. Several areas of expertise are available: computer simulation and modeling, computer graphics and animation, artificial intelligence, systems architecture, Computer Aided Drafting/Computer Aided Modeling (CAD/CAM), an automated office system, and networking and telecommunications, to name a few.

The CSC is recognized as a valuable resource in providing information, state-of-the-art technology, and services to other DOE sites and federal agencies throughout the U.S.



The CRAY X-MP/24

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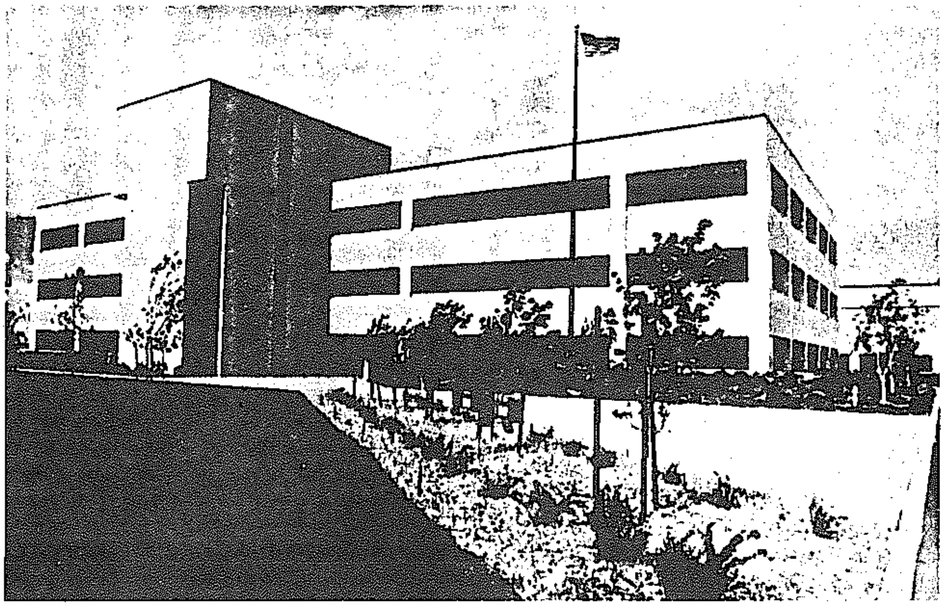
INEL RESEARCH CENTER

The INEL Research Center (IRC), located in Idaho Falls, is the DOE-ID Operations Office's multidisciplinary research center. With a staff of more than 400 scientists, engineers, and support personnel, research is conducted in such wide-ranging disciplines as materials science, chemistry, biotechnology, physical sciences, and environmental sciences. Work in these and other areas supports programs for such organizations as the Department of Energy, the Nuclear Regulatory Commission, the Environmental Protection Agency, the Department of Defense, and the Department of the Interior.

The IRC is a cornerstone for present and future government-based technology programs that are geared toward achieving energy self-sufficiency and improving industrial productivity and competitiveness for the United States. The program spectrum is impressive, embracing such activities as biocorrosion, supercritical fluid chemistry, muon-catalyzed fusion, powder metallurgy, environmental monitoring, and many others. The INEL facility, fostering close multidisciplinary cooperation among scientists, engineers, and program managers, is clearly a pivotal force in the pursuit of advanced research and development in energy-related programs and other facets of technology that are critical to the country.

The INEL Research Center (IRC)



The IRC is a focal point for INEL research and development activities. With its 70 individual and versatile laboratory units, it is instrumental in bringing to the INEL new customers, new programs, and new challenges. In addition to the work conducted for a wide variety of federal agencies, cooperative programs are under way with private industry, universities, and nonprofit organizations. The IRC also has become the focal point for the INEL's thrust in transferring technology to the private sector as well as fostering extensive university cooperation in a number of research efforts.

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NEW INITIATIVES

The INEL continues to expand its program base and has received new initiatives as industry, other government agencies, and the worldwide scientific community recognize the unique facilities, capabilities, and expertise available here. Defense and aerospace programs are becoming more important to the INEL as are cooperative research projects with universities and private industry.

In addition to the Department of Energy and the Nuclear Regulatory Commission, the INEL has new programs sponsored by the Department of Defense, the National Aeronautics and Space Administration, the Department of Interior, and the Environmental Protection Agency.

The DOE-ID Operations Office has been named as project office for the Multimegawatt Space Power Program, a Strategic Defense Initiative (SDI) Project. The INEL has also been designated as the preferred site for the Special Isotope Separation Project.

The INEL is a strong contender for siting or involvement in various national programs, such as

— Global Atmospheric Monitoring and Control

— TACS, an electronic engineering prototype of a field communication center for the Air Force

— Tactical Source Regional Simulator

— Medical reactor research for the application of reactors to the health field

— Nuclear rocket propulsion for space programs

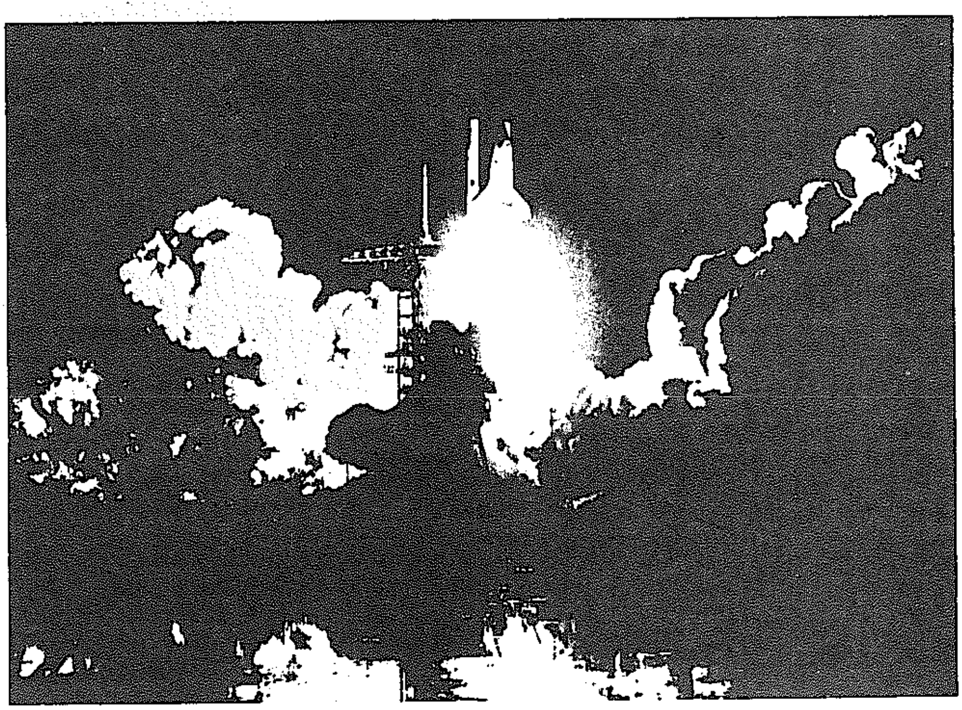
— Norfolk Shipyard Power System Upgrade for the Navy

— McClellan AFB and Air Training Command waste management projects

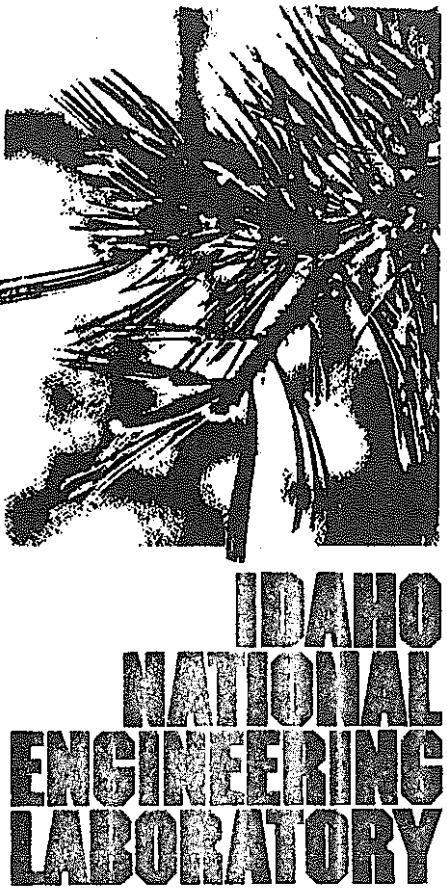
— Materials research for the Army Laboratory Command

— Data architecture conversion support for the Army Corps of Engineers

— Small Modular Advanced Reactor Test facility



INEL has new programs sponsored by NASA



HISTORY OF FACILITIES AND PROGRAMS

EXPERIMENTAL BREEDER REACTOR-I (EBR-I)

EBR-I, the first reactor built at INEL, began operation in 1951 and was decommissioned early in 1964 for lack of further assignments. This reactor produced the first usable electricity from nuclear heat in December 1951. In 1953 the facility confirmed that a nuclear reactor designed to operate in the high-energy neutron range is capable of breeding (creating more fuel than its operation consumes). The reactor was unmoderated and used sodium-potassium alloy (NaK) as coolant and enriched uranium (largely uranium-235) as fuel. A blanket of uranium-238 around the core provided the "fertile" material in which breeding took place. The liquid-metal coolant permitted the neutron energies to be kept high to promote fissionable material breeding. In addition, the coolant enabled high-temperature and low-pressure operation, both conducive to efficient power production.

EBR-I was brought to full power with the Mark-I core on December 21, 1951, with an operational loading of 52 kilograms of uranium-235 and subsequently operated with three different core loadings including a plutonium-fuel core.

On August 26, 1966, President Lyndon B. Johnson participated in ceremonies at the INEL officially designating EBR-I as a national historic landmark.

The historic reactor facility is open to the public seven days a week from Memorial Day weekend through Labor Day. Group tours can be arranged any time of the year.

BOILING WATER REACTOR EXPERIMENTS (BORAX)

BORAX-I was the first in a series of five INEL reactors to pioneer intensive work on boiling water reactors. In these reactor types, the coolant/moderator boils in the core and passes saturated steam directly to the turbine for power generation.

BORAX-I was constructed in 1953 to demonstrate the feasibility of this type of reactor concept. It was designed for 1.4 megawatts (thermal). The facility was deliberately destroyed in July 1954 to determine its inherent safety under extreme conditions.

BORAX-II was constructed in late 1954 for further tests. New core combinations were installed using varying enrichments of uranium-235 in the metal fuel plates. The power level was 6 megawatts (thermal).

BORAX-III, operated in 1955, was designed for 15 megawatts (thermal) with a 2000-kilowatt turbine generator to investigate use of boiling water reactors for generating electric power. On July 17, 1955, it produced sufficient power to experimentally power and light the city of Arco, Idaho—a world first. The reactor generated approximately 2000 kilowatts (electrical) over a period of about two hours, distributed as follows: 500 to light Arco, 500 to power the BORAX facility, and 1000 to power the Central Facilities Area at INEL.

BORAX-IV operated from December 1956 until June 1958. This reactor, 20 megawatts (thermal), was used principally to test high-thermal capacity fuel elements made from mixed oxides (ceramics) of uranium and thorium.

The ceramic core of uranium-thorium-oxide fuel elements demonstrated the feasibility of stable operation with this fuel. This fuel can be operated at higher temperatures, is less reactive with water coolant in case of cladding rupture, is cheaper to manufacture, and has higher burnup possibilities. The reactor also produced measurable quantities of the artificial, thorium-derived fuel, uranium-233. BORAX-IV was operated satisfactorily with several experimentally defective fuel elements in the core. Relatively minor contamination of auxiliary equipment resulted.

BORAX-V, with a design power of 40 megawatts (thermal), provided an extremely flexible facility for determining the safety aspects and feasibility of an integral, nuclear superheat system. BORAX-V achieved criticality on February 9, 1962, and on October 10, 1963, produced superheated (dry) steam wholly by nuclear means for the first time. The reactor demonstrated that improved efficiency from manufactured steam is obtainable by incorporating as a design feature a number of superheated fuel assemblies in the reactor core lattice.

ZERO POWER REACTOR NO. 3 (ZPR-III)

A split-table machine used to achieve criticality by bringing two halves of a fuel configuration together, ZPR-III was used to determine the accuracy of predicted critical mass geometries and critical measurements in connection with various loadings for makeup of fast reactor core designs. The cores of EBR-II, Fermi, Rapsodie, and SEFOR reactors were originally mocked up in this facility.

Theoretical predictions of the system's performances took into account neutron reactions important to the neutron chain over a wide range of energies, e.g., from 50 kilovolts to a few million volts. Experimental critical assembly results in this field were almost completely lacking before the advent of the ZPR-III facility in October 1955. The reactor was placed on standby in 1970 and is now on display in the EBR-I visitor center.

MATERIALS TESTING REACTOR (MTR)

MTR first achieved nuclear startup on March 31, 1952, the second reactor to be operated at the INEL. The historic reactor went into retirement April 23, 1970. Its materials testing workload was taken over by the new and larger Advanced Test Reactor. The MTR was decommissioned in 1974.

The choice of core structural materials and fuel elements for every reactor designed in the country since 1952 has been influenced by information obtained from tests in the MTR. High-flux radiation fields available in this reactor made greatly accelerated screening tests possible, which were of immediate benefit to reactor design. In its earliest stages, it contributed vitally to work on pressurized water reactors and later to the Yankee and Dresden power stations, organic reactors, liquid-metal-cooled reactors, and even homogeneous reactors. Successful MTR operation was in itself a great experiment resulting in a family of plate-type reactors.

The reactor was operated at 30 megawatts (thermal) until September 1955 when the thermal output was increased to 40 megawatts.

MTR logged more than 125,000 operating hours and more than 19,000 neutron irradiations. During August 1958, the MTR became the first reactor to be operated using plutonium-239 as fuel at power levels up to 30 megawatts, the original design power of the reactor. The demonstration proved the feasibility of fabricating a plutonium fuel core capable of withstanding high-power, high-flux conditions. The test also demonstrated that a reactor fueled with plutonium can be satisfactorily controlled.

ENGINEERING TEST REACTOR (ETR)

ETR first achieved nuclear startup in 1957. At that time it was the largest and most advanced materials test reactor in the world.

ETR was a 175-megawatt (thermal) reactor, built to provide more testing space and flexibility than the older MTR. ETR was used to evaluate fuels, coolant, and moderator characteristics under environments similar to those in many types of power reactors.

In 1972, the ETR was modified for a new role to support DOE's breeder reactor safety program. Test programs relating to reactor core design and operation were performed by means of the Sodium Loop Safety Facility (SLSF) inserted into the ETR core. As the testing progressed, the reactor was again modified with a new top closure accommodating the irradiation loop. Other modifications included adding a helium coolant system and sodium-handling system.

ETR went into retirement in 1982. It was the first complete reactor facility to be deactivated and documented immediately after shutdown.

ETR CRITICAL (ETRC) FACILITY

ETRC was a full-scale, low-power nuclear facsimile of ETR. The facility, achieving criticality on May 20, 1957, was used to determine in advance the nuclear characteristics of experiments planned for irradiation in ETR. It also determined the power distribution effects for a given ETR fuel and experiment loading.

Since no two ETR loadings were identical, the ETRC allowed prediction of the ETR's nuclear environment when completed experiments were removed or new ones were added. This information was necessary to calculate the experiment irradiation and determine core life, control rod withdrawal sequences, reactivity worths, and core safety requirements.

Proposed fuel and experiment loadings were first mocked up in ETRC and manipulated until a desired power distribution throughout the core was attained, satisfying pertinent safety requirements.

The reactor tests provided substantial savings of time and money through necessary low-power testing without interruption of full-power ETR operation. ETRC was retired from service with the ETR in 1982.

REACTIVITY MEASUREMENT (RMF) FACILITY

RMF, a detector reactor, which measured reactivity changes in materials irradiated in the MTR or ETR, was operated for more than eight years following startup on February 11, 1954. The RMF was used to assay new and spent fuel elements and to assist in experiment scheduling by evaluating reactivity losses and flux depression caused by in-pile apparatus. The RMF was retired April 10, 1962.

SPECIAL POWER EXCURSION REACTOR TEST (SPERT) FACILITIES

Four SPERT reactors were designed and operated to study a wide range of variables such as plate design; core configuration; coolant flow; and reflector, moderator, void, and temperature coefficients.

The SPERT reactor experiments fell into four major classifications: (a) static experiments to determine such core characteristics as void and temperature coefficients, (b) step tests in which the system is suddenly made supercritical, (c) ramp tests where reactivity is added to the reactor at a constant rate, and (d) stability tests involving either spontaneous or externally induced oscillations.

These tests were performed under various conditions of temperature, pressure, and coolant flow on differing core designs. All operations were conducted from a central control building, one-half mile from the reactors. Most of the experiments conducted in the SPERT reactors were safely below the threshold of core damage. Fuel damage was sustained in only a few tests.

SPERT-I was placed in operation June 11, 1955, and decommissioned in the fall of 1964. It was an open-tank, light-water-moderated and reflected reactor, originally using 92 percent enriched uranium fuel. The reactor tank, about 4 feet in diameter and 14 feet high, was filled with water to a level about 2 feet above the core.

In general, the SPERT-I tests were characterized by sudden power rises, arrested by inherent shutdown or self-limiting tendencies of the reactor. In a typical excursion, reactor power would attain a peak in a fraction of a second and, without manipulation of control rods, drop to much lower, though generally steady, levels.

Before being phased out in the fall of 1964, SPERT-I demonstrated the damage-resistant capabilities of low-enrichment (4 percent uranium-235) uranium-oxide fuel pins similar to those used in water-cooled reactors powering large central stations.

SPERT-II sustained initial criticality March 11, 1960. This facility consisted of a pressurized water reactor with coolant flow systems designed for operation with either light or heavy water pressures up to 375 pounds per square inch, temperatures up to 400°F, and flow rates up to 20,000 gpm. SPERT-II was placed in standby in 1964 and decommissioned in 1980.

SPERT-III became operative December 19, 1958. It was considered to be the most versatile facility yet developed for studying the inherent safety characteristics of nuclear reactors. This reactor provided the widest practical range of control over three variables: temperature, pressure, and coolant flow. SPERT-III was placed in standby upon completion of its programmed operations in 1968. The facility was decommissioned in 1980 and is now used for the Waste Experimental Reduction Facility.

SPERT-IV construction was completed in October 1961, and criticality was achieved July 24, 1962. This was an open-tank, twin-pool facility that permitted detailed studies of reactor stability as affected by varying conditions including forced coolant flow, variable height of water above the core, hydrostatic head, and other hydrodynamic effects.

The Capsule Driver Core (CDC) was operated for several years in the SPERT-IV reactor to gain information on fuel-destructive mechanisms pending completion of the new Power Burst Facility. The CDC program in SPERT-IV ended in 1970.

SYSTEMS FOR NUCLEAR AUXILIARY POWER TRANSIENT PROGRAMS

An extension of the SPERT program, known as the Safety Testing Engineering Program, or STEP, was conducted in the 1960s by Phillips Petroleum Company in the Initial Engineering Test Facility at Test Area North. STEP involved engineering-scale field testing to demonstrate the safety of aerospace as well as land-based reactor systems.

The successful testing of a SNAP-10A auxiliary power space reactor on April 1, 1964, was one of the early STEP undertakings. Designated SNAPTRAN-3, the test showed that if such a device should fall into water, it would immediately destroy itself instead of building up a high inventory of radioactive fission products. Successful follow-on tests of the SNAP-10A system were SNAPTRAN-1 and SNAPTRAN-2.

AIRCRAFT NUCLEAR PROPULSION (ANP) PROGRAM

Test Area North was originally the site of the ANP program where significant progress was made in the 1950s toward development of a nuclear airplane. The program involved building and testing three Heat Transfer Reactor Experiments (HTRE-1, HTRE-2, and HTRE-3, which developed 20, 14, and 32 megawatts of heat energy, respectively). These air-cooled reactors proved the feasibility of operating an aircraft turbojet engine with nuclear heat.

Three low-power reactors were also operated at TAN in support of the ANP program: the Shield Test Pool Facility Reactor (SUSIE), Critical Experiment Tank (CET), and Hot Critical Experiment (HOTCE). Since the ANP program was discontinued (by order of the President) on March 28, 1961, several other programs have been assigned to the area.

SPHERICAL CAVITY REACTOR CRITICAL EXPERIMENT (SCRCE)

SCRCE in Test Area North was an outgrowth of a program begun in the 1960s to investigate the feasibility of a space propulsion reactor concept.

The purpose of the reactor was to test the nuclear feasibility of heating hydrogen propellant to approximately 10,000°F by a ball, or core, of low-density gaseous uranium hexafluoride (UF6). The ball of uranium gas would be held in place by the hydrogen flowing around it, something like a ping-pong ball suspended in a stream of air. Uranium core temperatures as high as 100,000°F were considered possible.

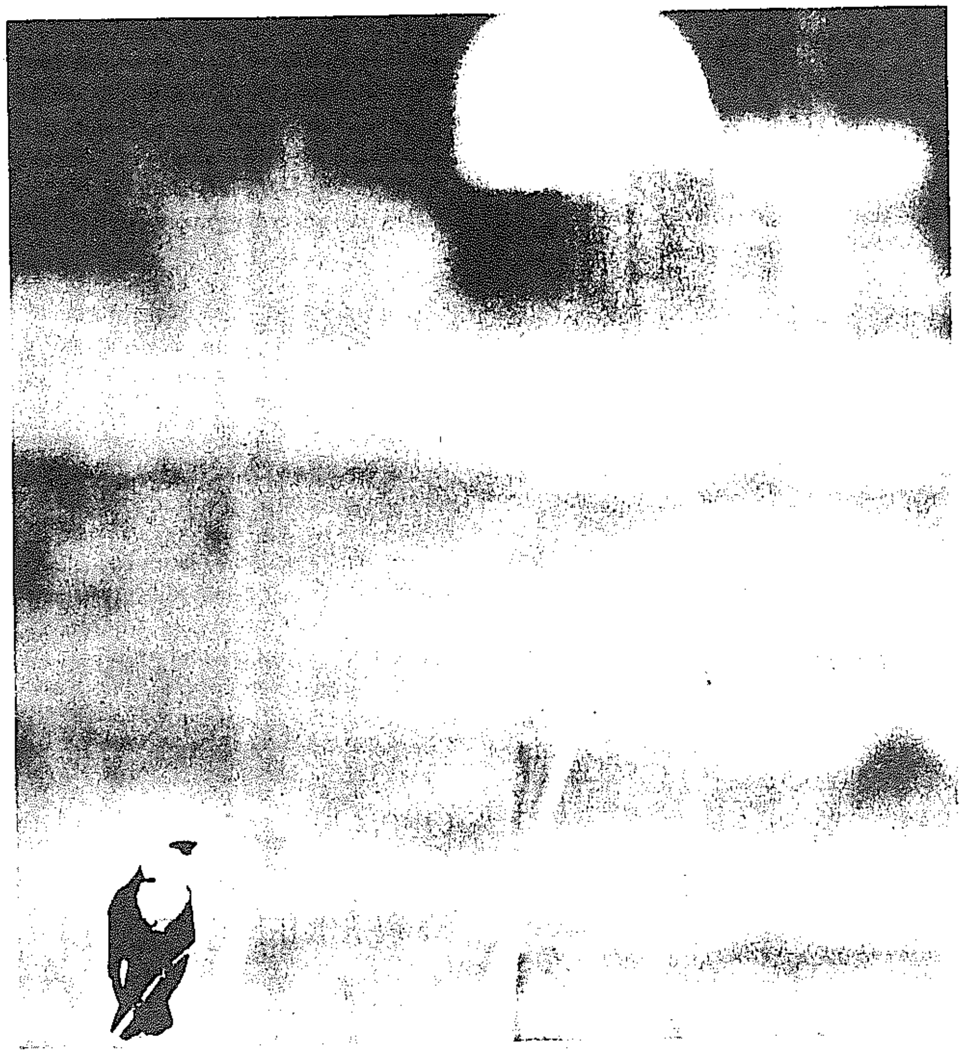
On May 17, 1967, the world's first gas-core reactor, SCRCE, was successfully operated.

FAST SPECTRUM REFRACTORY METALS REACTOR (710)

This low-power critical facility was operated from 1962 to 1968 to collect data on a proposed fast spectrum refractory-metal reactor concept called the 710 reactor. The concept involved using metals, such as tungsten and tantalum, in developing a compact, very high temperature reactor for generating power in space.

HIGH-TEMPERATURE MARINE PROPULSION REACTOR (630-A)

The 630-A Reactor Critical Experiment was operated to explore the feasibility of an air-cooled, water-moderated system for nuclear-powered merchant ships. Further development was discontinued in December 1964 when decisions were made to lower the priority of the entire nuclear power merchant ship program.



The LOFT reactor

LOSS-OF-FLUID TEST (LOFT)

The LOFT reactor was inactivated in 1986, following completion of the LP-FP-2 experiment, the most significant severe fuel damage test ever conducted in a nuclear reactor. This test, which involved the heatup and melting of an 100-rod experimental fuel bundle, is providing information on the release and transport of fission products that may occur during an actual commercial reactor accident where core damage occurs. The facility is now available for new programs.

The internationally famous LOFT reactor, a scaled version of a commercial pressurized water reactor, was the only reactor in the world used for total pressurized water systems simulations of loss-of-coolant accidents (LOCAs).

Thirty-eight nuclear power tests were conducted on various accident scenarios between 1978 and 1985. A major aspect of the program was to investigate the capability of emergency core cooling systems to prevent core damage during a LOCA. Experiments at LOFT have simulated small-, medium-, and large-break LOCAs, sometimes complicated with other events such as "loss of offsite power."

Experimental data from the program have been used to evaluate and improve computer predictive codes that NRC uses to assess the adequacy of licensing requirements for commercial nuclear power plants, and to aid NRC in making regulatory judgments. Information generated from the LOFT program has been instrumental in helping the nuclear industry maintain its safe operating record.

EXPERIMENTAL BERYLLIUM OXIDE REACTOR (EBOR)

Modifications of an existing facility at Test Area North were begun in May 1963 to house EBOR. Work was terminated in 1966 before construction completion. The project's objective was to develop technology for using beryllium oxide as a neutron moderator in high-temperature, gas-cooled reactors. Among the reasons for the cancellation was the encouraging progress achieved, concurrent with EBOR construction, in developing graphite as a moderator, which lessened the importance of developing the alternate moderator.

GAS-COOLED REACTOR EXPERIMENT (GCRE) AND MOBILE LOW POWER PLANT NO. 1 (ML-1)

From 1957 through 1965, a nine-year, military-oriented program in the Auxiliary Reactor Area (then known as the Army Reactor Area) saw near fruition of the U.S. Army's quest for a compact, light weight, mobile power reactor that could be transported by air or tractor-trailer, with minimal intervals between shutdown and restartup in a new location.

The most successful portion of the army program involved GCRE and the ML-1. GCRE was a water-moderated, nitrogen-cooled, direct- and closed-cycle reactor generating 2200 kilowatts of heat, but no electricity. It achieved criticality February 23, 1960. After accomplishing its mission (the proof-of-principle phase of a mobile nuclear power plant followed by three months testing of a prototype reactor package for ML-1), it was placed on standby April 6, 1961.

ML-1 reactor operation occurred in a separate facility, beginning with criticality March 30, 1961. Operation ended May 29, 1964, after a series of power runs climaxed by 664 consecutive operating hours. A determination by the Army in late 1965 resulted in phasing out the ML-1 program because of an inability to identify a specific current mission and questions about cost-effectiveness.

NUCLEAR EFFECTS REACTOR (FRAN)

A small-pulsed reactor, capable of supplying bursts of high-intensity fast neutrons and gamma radiation, was transferred to the Site in mid-1967 from the Nevada Test Site, where it had been operated by Lawrence Livermore Laboratory.

Beginning with first criticality August 28, 1968, in the former ML-1 reactor building, FRAN was used for a short time to test new detection instrument performance then being developed for reactor control purposes. The reactor was moved to DOE's Lawrence Livermore Laboratory at Livermore, California, in June 1970.

ORGANIC MODERATED REACTOR EXPERIMENT (OMRE)

OMRE was constructed and operated at the INEL for several years to demonstrate the technical and economic feasibility of using a liquid hydrocarbon as both the coolant and the moderator, a reactor concept developed and partially financed by Atomics International.

The primary purpose was to study the radiation and thermal stability of the organic materials used and the associated physical property changes under actual reactor operating conditions, OMRE was phased out in April 1963 after accomplishing this purpose. OMRE first achieved criticality September 17, 1957. OMRE was decommissioned in 1979, and the area was released for unrestricted use.

SPLIT TABLE REACTOR (STR)

STR was a low-power facility at Test Area North. Its nuclear core was placed in two halves of a vertical, aluminum, honeycomb-like matrix. The reactor could not be operated until the two halves were brought together to form the critical fuel mass. STR served to mock up reactor design concepts for both thermal and fast neutron reactor systems. It obtained basic physics and design data for such concepts.

SEMISCALE

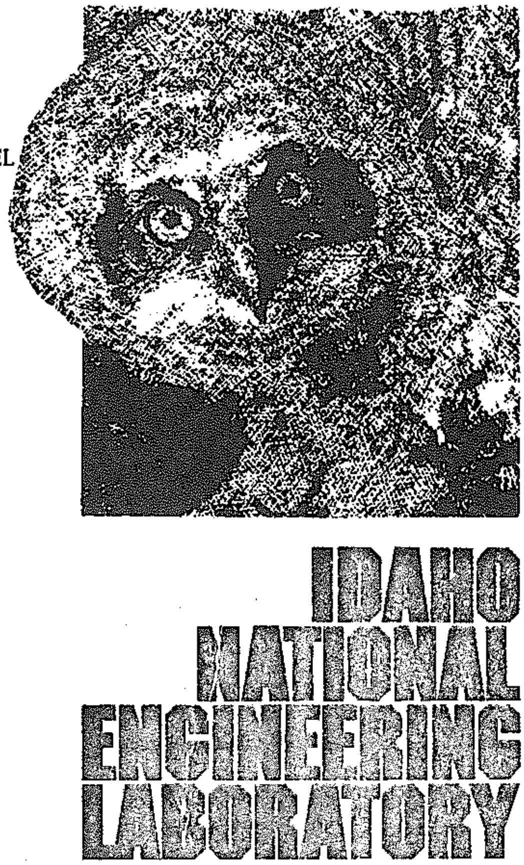
Semiscale is a nonnuclear facility at the INEL Test Area North used for investigating the behavior of pressurized water reactor systems. Specifically, the facility conducts tests on thermal-hydraulic phenomena that can occur during reactor power transients.

The facility has undergone many modifications since it started operating in 1965 and, through hundreds of tests, has played an important role in developing computer safety codes for nuclear power plants. Semiscale was the nation's major source of experimental data on reactor system performance until results from the INEL Loss-of-Fluid Test Facility became available in 1978.

Semiscale is also used to develop and test advanced instruments for reactors and experimental facilities. The facility has an electrically heated core with piping and components scaled down from a large nuclear power plant.

A unique aspect of the facility is that it can be quickly modified to meet new test conditions. This ability was demonstrated during the Three Mile Island Unit 2 accident in 1979 when the facility was used to provide information on the hydrogen buildup in the TMI-2 reactor vessel. Modifications were quickly installed, and the information was available within 24 hours of the request.

Located near the Semiscale Facility is the Two-Phase Flow Loop, a nonnuclear facility with the capability of delivering controlled steam/water flow to a test section. The facility, used to investigate some of the intricate steam and water relationships that may occur during a nuclear power plant accident, is instrumented to measure two-phase flow variables such as flow rate, mass, and energy. The Two-Phase Flow Loop is one of the largest test loops in the world.

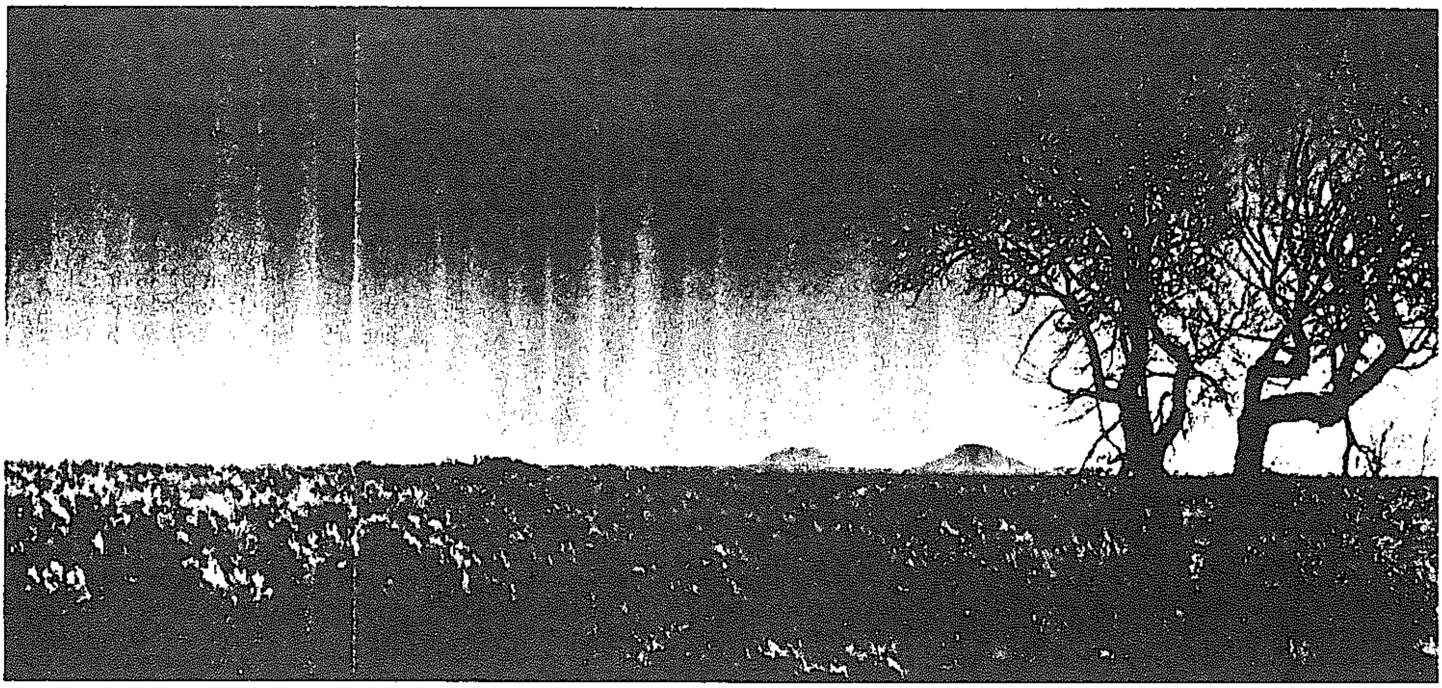


AREA HISTORY

The Snake River Plain is classified in the Pleistocene epoch, which began one million years ago and is one of the most recent geologic time categories. Fossils of prehistoric mammals have been found in excavations at the INEL Site. It is postulated that the fossils are from camels and mastodons that inhabited the region about 35,000 years ago. A fossil taken from carboniferous strata, encountered during well drilling at approximately 100 feet below land surface, has been determined to be over 40,000 years old.

Recent archaeological investigations disclosed evidence that man has been in the Eastern Idaho region for perhaps 10,000 to 12,000 years. Fur trappers were the first white men to enter the area.

Thyery Godin, a French-Canadian trapper representing the English Northwest Company, discovered what was then known as the Godin River in 1820. Later it became known as the Big Lost River because of the phenomenon of the river's "disappearance" in the area now circumscribed by the Site boundaries. Alexander Ross, representing the Hudson Bay Company, visited the Godin River in 1824 and mentioned the "Three Pilot Knobs," which could have been the Three Buttes on the INEL Site or the Teton Mountains, which were also referred to by this name. The Lost River Sinks and the Three Buttes were shown on map sketches made by Captain Bonneville, U.S. Army, in 1832-34. In the winter of 1832-33, he referred to the Snake River Plain as the great plain of the Three Buttes.



In the late 1870's, the INEL Site area was crossed by a trail used for large cattle herds moving eastward from Oregon to eastern markets and ranges made available in Wyoming by treaties with the Indians. Two stagecoach lines also crossed the plain near the Twin Buttes, which served as landmarks for early gold seekers. A branch of the Oregon Short Line Railroad Company was constructed in 1910. Cerro Grande, now only a location name at the southern boundary of the Site, was the terminus until the rails were extended to Arco and the mining town of Mackay.

An area within the INEL Site boundary was once a part of the Big Lost River Irrigation Project, one of the historically colorful reclamation projects in the West. It was authorized under the Carey Act of 1894, which provided that each state could be given land suitable for irrigation if the states did the reclaiming. Idaho accepted the application on the basis that private capital could be induced to construct the works and that the state would provide supervision.

A dam on the Lost River was started in 1909 to provide storage to irrigate some 100,000 acres, 30,000 of which were known as the Powell Tract, lying within what is now the boundaries of the Site. During 1910, canals, ditches, and channel structures were constructed. The project was plagued with grave errors of engineering, financial difficulties, and legal and political controversies. Construction of the Powell Tract was discontinued in the spring of 1911. The old canals and structures are still prominent landmarks.

A similar project on the Little Lost River involved a small tract of land on the northwest side of the Site. The Mud Lake Project in the northeast also included land within the Site boundary. Both projects were the result of overly optimistic estimates. The dry canal systems are all that remain.

During World War II, the U.S. Navy utilized about 270 square miles of the plain as a gunnery range. An area southwest of the naval area was once used by the U.S. Army Air Corps as an aerial gunnery range. The present Site included all of the former military area and a large adjacent area withdrawn from the public domain for use by DOE. The former navy administration shop, warehouse, and housing area are today the Central Facilities Area of the INEL Site.

Two stagecoach lines crossed the plain near the Twin Buttes, landmarks for early gold seekers

APPENDIX

FACILITIES AT THE IDAHO NATIONAL ENGINEERING LABORATORY

Reactors Operating or Operable (As of 1986)

Name	Page	Abbreviation	Operating Contractor
1. Advanced Reactivity Measurement Facility No. I	10	ARMF-I	EG&G Idaho
2. Advanced Test Reactor	9	ATR	EG&G Idaho
3. Advanced Test Reactor Critical	9	ATRC	EG&G Idaho
4. Argonne Fast Source Reactor	8	AFSR	ANL
5. Coupled Fast Reactivity Measurement Facility	10	CFRMF	EG&G Idaho
6. Experimental Breeder Reactor-II	7	EBR-II	ANL
7. Large Ship Reactor "A"	21	A1W(A)	WEC
8. Large Ship Reactor "B"	21	A1W-(B)	WEC
9. Natural Circulation Reactor	21	S5G	WEC
10. Neutron Radiography Facility	8	NRAD	ANL
11. Submarine Thermal Reactor	20	S1W(STR)	WEC
12. Transient Reactor Test Facility	8	TREAT	ANL
13. Zero Power Plutonium Reactor	8	ZPPR	ANL
14. Power Burst Facility	11	PBF	EG&G Idaho

Reactors Dismantled, Transferred, or on Standby Status

1. Advanced Reactivity Measurement Facility No. II	10	ARMF-II	PPCo
2. Boiling Water Reactor Experiment No. 1	28	BORAX-I	ANL
3 Boiling Water Reactor Experiment No. 2	28	BORAX-II	ANL
4. Boiling Water Reactor Experiment No. 3	28	BORAX-III	ANL
5. Boiling Water Reactor Experiment No. 4	28	BORAX-IV	ANL
6. Boiling Water Reactor Experiment No. 5	29	BORAX-V	ANL
7. Cavity Reactor Critical Experiment	31	SCRCE	GE, INC
8. Critical Experiment Tank	30	CET	GE
9. Engineering Test Reactor	29	ETR	INC, ANC, EG&G Idaho
10. Engineering Test Reactor Critical	29	ETRC	INC, ANC, EG&G Idaho
11. Experimental Beryllium Oxide Reactor	32	EBOR	GA
12. Experimental Breeder Reactor-I	28	EBR-I	ANL
13. Experimental Organic Cooled Reactor (mothballed before startup)	*	EOCR	PPCo
14. FAST Spectrum Refractory Metals Reactor	31	710	GE
15. Gas-Cooled Reactor Experiment	32	GCRE	AGC
16. Heat Transfer Reactor Experiment No. 1	30	HTRE-I	GE
17. Heat Transfer Reactor Experiment No. 2	30	HTRE-II	GE
18. Heat Transfer Reactor Experiment No. 3	30	HTRE-III	GE
19. High Temperature Marine Propulsion Reactor	31	630-A	GE
20. Hot Critical Experiment	30	HOTCE	GE
21. Loss-of-Fluid Test Facility	31	LOFT	INC, ANC, EG&G Idaho
22. Materials Testing Reactor	29	MTR	PPCo
23. Mobile Low Power Reactor No. 1 (Army)	32	ML-1	AGC
24. Nuclear Effects Reactor	32	FRAN	INC
25. Organic Moderated Reactor Experiment	32	OMRE	AL
26. Power Burst Facility	11	PBF	INC, ANC, EG&G Idaho
27. Reactivity Measurement Facility	29	RMF	PPCo
28. Shield Test Pool Facility Reactor	30	SUSIE	GE
29. SNAP-10A Transient No. 1	30	SNAPTRAN-1	AI/PPCo
30. SNAP-10A Transient No. 2	30	SNAPTRAN-2	AI/PPCo
31. SNAP-10A Transient No. 3	30	SNAPTRAN-3	AI/PPCo
32. Special Power Excursion Reactor Test No. I	30	SPERT-I	PPCo
33. Special Power Excursion Reactor Test No. II	30	SPERT-II	PPCo
34. Special Power Excursion Reactor Test No. III	30	SPERT-III	PPCo
35. Special Power Excursion Reactor Test No. IV	30	SPERT-IV	PPCo
36. Spherical Cavity Reactor Critical Experiment	31	SCRCE	ANC
37. Split Table Reactor	32	STR	GE, INC, ANC
38. Stationary Low Power Reactor No. 1[a]	*	SL-I	CE
39. Zero Power Reactor No. 3	29	ZPR-III	ANL

a. Accidently destroyed during shutdown, January 3, 1961, following 931.5 megawatt days of operation.

* Not covered in this brochure.

Other Facilities in Use

Name	Page	Abbreviation	Operating Contractor
1. Argonne National Laboratory-West Area	7	ANL-W	ANL
2. Central Facilities Area	24	CFA	EG&G Idaho
3. Coal-Fired Steam Generating Facility	*	CFSGF	WINCO
4. Computer Science Center (in Idaho Falls)	25	CSC	EG&G Idaho
5. CFRMF Neutron Radiography Facility	*	—	EG&G Idaho
6. Expended Core Facility	21	ECF	WEC
7. Experimental Field Station	*	EFS	DOE-ID
8. Fluorinel Dissolution and Fuel Storage Facility	14	FAST	WINCO
9. Hot Cell Facility (TRA)	18	—	EG&G Idaho
10. Hot Fuel Examination Facilities	8	HFEF	ANL
11. Hot Shop Facilities (TAN)	18	—	EG&G Idaho
12. Idaho Chemical Processing Plant	12	ICPP	WINCO
13. INEL Research Center (in Idaho Falls)	26	IRC	EG&G Idaho
14. Naval Reactors Facility	20	NRF	WEC
15. New Waste Calcination Facility	13	NWCF	WINCO
16. Process Experimental Pilot Plant	16	PREPP	EG&G Idaho
17. Radiation Measurements Laboratory	10	RML	EG&G Idaho
18. Radioactive Waste Management Complex	15	RWMC	EG&G Idaho
19. Radiological and Environmental Sciences Laboratory	4	RESL	DOE-ID
20. Reactor Training Facility	*	RTF	EG&G Idaho
21. Remote Analytical Laboratory	14	RAL	WINCO
22. Semiscale Test Facility	33	STF	EG&G Idaho
23. Standards Calibration Laboratory (CF-698)	24	SCL	EG&G Idaho
24. Stored Waste Examination Pilot Plant	16	SWEPP	EG&G Idaho
25. Technical Services Center (CF-688, 689)	24	TSC	EG&G Idaho
26. Technical Support Buildings A&B	2	TSA; TSB	EG&G Idaho
27. Test Area North	17	TAN	EG&G Idaho
28. Test Reactor Area	9	TRA	EG&G Idaho
29. Waste Experimental Reduction Facility	16	WERF	EG&G Idaho
30. Willow Creek Building (in Idaho Falls)	2	WCB	EG&G Idaho

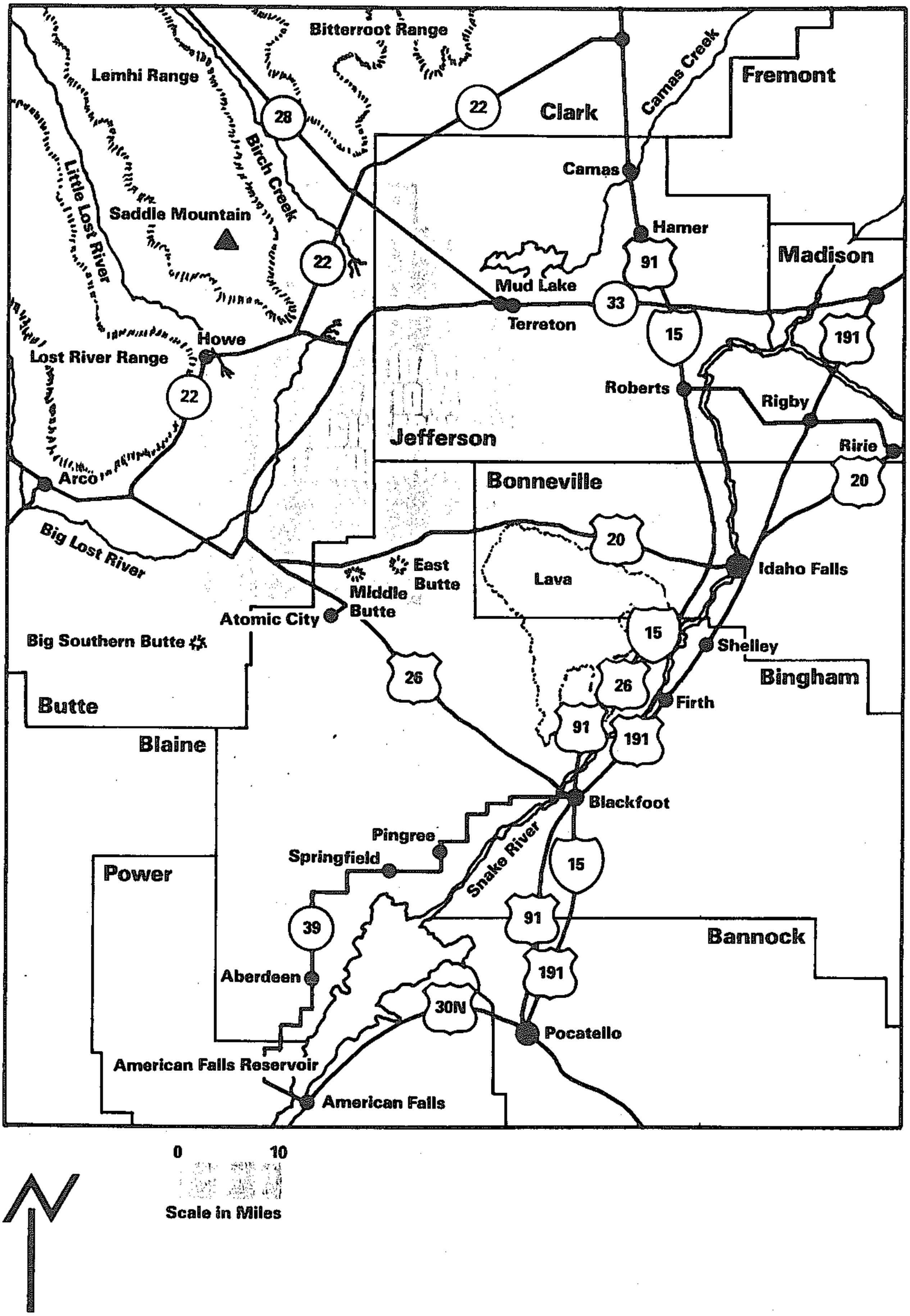
Facilities Under Construction

1. Fuels Processing Facility Project	14	FPF	WINCO

Facilities Dismantled, Transferred, or on Standby Status

1. Alcohol Fuels Plant	*	—	EG&G Idaho
2. Raft River Geothermal Project	*	—	EG&G Idaho
3. Waste Calcining Facility	13	—	PPCo, AL, ENICO
4. Semiscale Test Facility	33	STF	EG&G Idaho

* Not covered in this brochure.


Bitterroot Range
Lemhi Range
Clark
Camas Creek
Fremont
28
22
Birch Creek
Little Lost River
Saddle Mountain
Camas
Hamer
Madison
22
91
Mud Lake
33
Terreton
15
191
Howe
Lost River Range
22
Roberts
Rigby
Jefferson
Ririe
Arco
Bonneville
20
Big Lost River
20
Lava
East Butte
Middle Butte
Idaho Falls
Atomic City
Big Southern Butte
15
Shelley
26
26
Bingham
Butte
Firth
91
191
Blaine
Blackfoot
Pingree
Springfield
Snake River
Power
15
39
91
Bannock
191
Aberdeen
30N
Pocatello
American Falls Reservoir
American Falls
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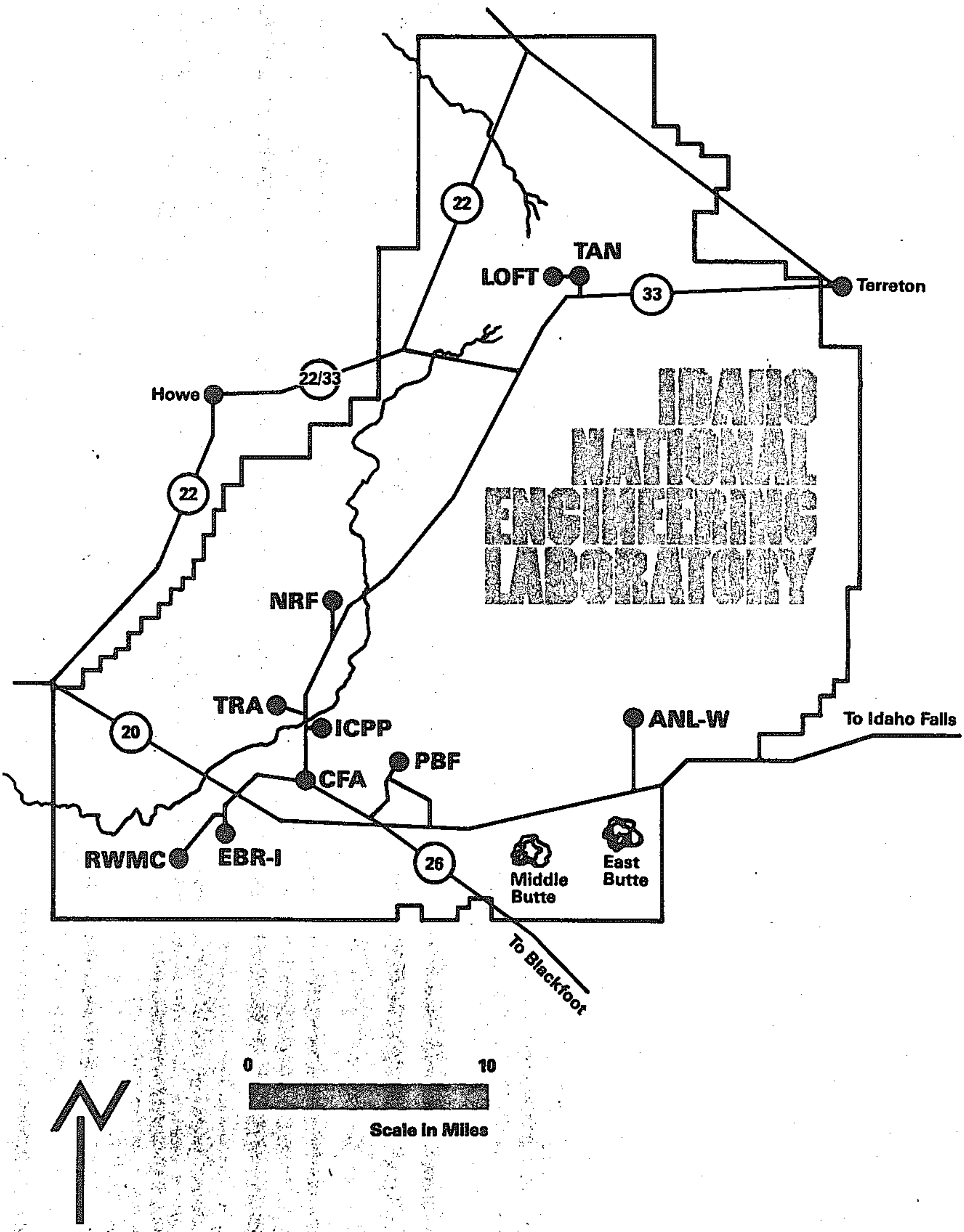
Scale in Miles



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Prime Contractor
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Joseph E. Stiegler
Manager
Transportation System
Development Department

Sandia National Laboratories
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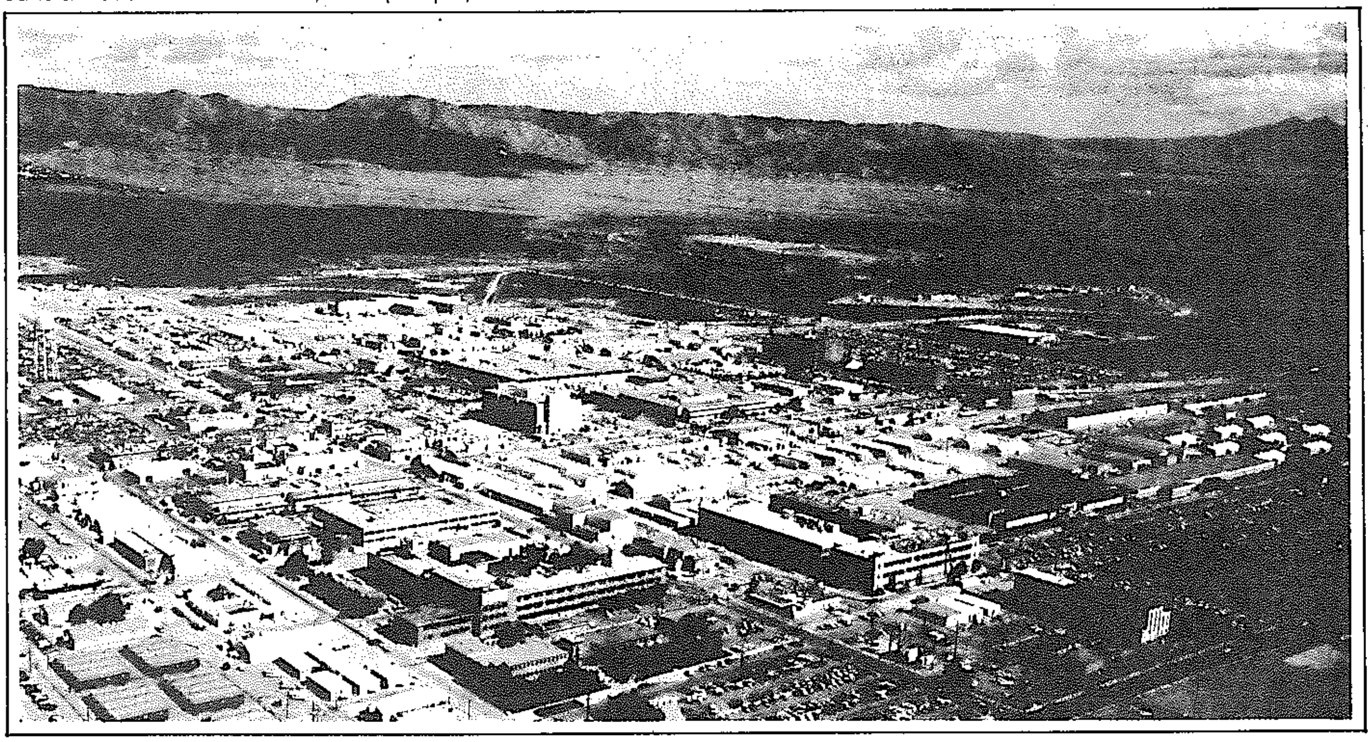
Sandia National Laboratories

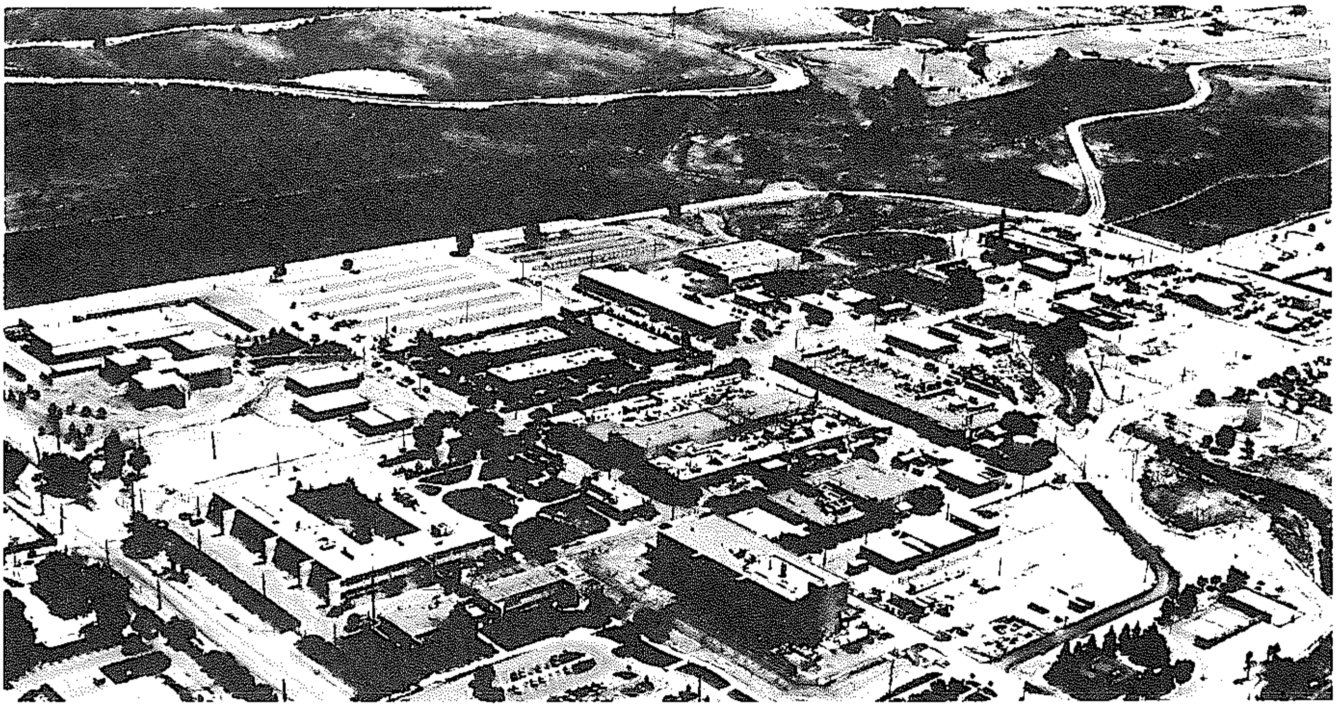
Albuquerque, New Mexico　　Livermore, California

Sandia National Laboratories, Albuquerque, New Mexico



Sandia National Laboratories (then Sandia Laboratory) was established in 1945 and operated by the University of California until 1949, when President Truman asked AT&T to assume the operation as an "opportunity to perform an exceptional service in the National interest." Today AT&T Technologies, Inc., continues to operate Sandia for

Sandia National Laboratories, Livermore, California



the Department of Energy on a no-profit, no-fee basis. The Labs' responsibility is national security programs in defense and energy, with primary emphasis on nuclear weapon research and development. Sandia also does a limited amount of work for the Department of Defense and other federal agencies on a non-interference basis.

January 1988

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CONTENTS

SANDIA OVERVIEW

Sandia National Laboratories is one of the nation's largest research and development engineering facilities. Sandia headquarters and its main laboratory are located on Kirtland Air Force Base on the southeast edge of Albuquerque. Sandia Livermore, located in the San Francisco Bay area just east of the city of Livermore, adjoins Lawrence Livermore National Laboratory and was established in 1956 to provide a close working relationship with that lab. Test ranges are operated near Tonopah, Nevada, and on Kauai, Hawaii.

About 60 percent of Sandia's research and development effort—including applied research in semiconductors, materials, etc.—involves the weaponization of nuclear explosives for national defense; the remainder involves energy programs and advanced military technologies.

Sandia employs about 8400 persons. Almost 60 percent are in technical and scientific positions, and the remainder are in crafts, skilled labor, and administrative classifications. About 7200 persons work in Albuquerque and about 1080 in Livermore. The total includes more than 100 at the Tonopah Test Range, the Nevada Test Site, and elsewhere.

Sandia operates a broad range of facilities, many of them unique. They are used for a wide variety of projects, ranging from basic materials research, a relatively large program, to the design of specialized parachutes. Assets, owned by DOE and acquired at a cost of more than $922 million, include about 500 major buildings containing some 4 million square feet of floor space. They are located on land totaling approximately 562 square miles, most of which is at Tonopah Test Range.

The facilities include state-of-the-art equipment for environmental testing, radiation research, combustion research, computing, and microelectronics research and production (see pages 13-15).

Other major facilities include a full-service Technical Library for employees; a Primary Standards Laboratory; transonic, supersonic, and hypersonic wind tunnels; and design, fabrication, and process development laboratories.

Sandia is a major factor in its area and state economies. In FY 1987, purchases totaled almost $244 million in New Mexico and $53 million in California (see page 18 for payrolls).

NUCLEAR WEAPON PROGRAMS

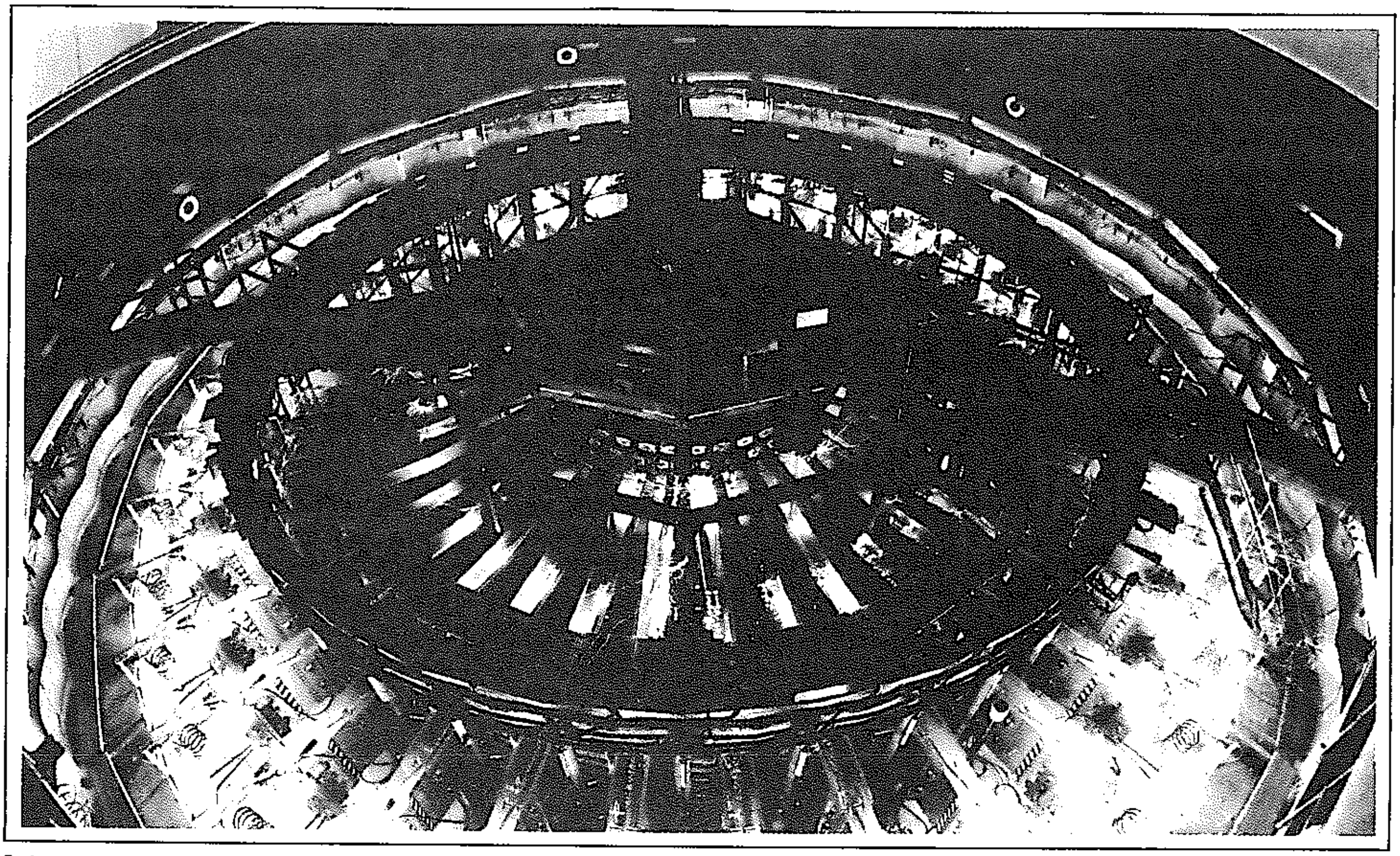
The Labs' primary mission is to monitor and maintain the health and safety of the nuclear-weapon stockpile; to develop and engineer new nuclear weapons for production; to explore advanced weapon concepts and perform research that will keep the nation second to none in this arena; and finally to maintain for the future all of these capabilities as a continuing strategic national defense resource.

Sandia's work on nuclear ordnance—principally weapon safety, control, arming, fuzing, and firing systems; aerodynamics and structures; and related testing and instrumentation—is carried out in close cooperation with Los Alamos National Laboratory (LANL) and Lawrence Livermore National Laboratory (LLNL), DOE's other two nuclear weapon laboratories. Operated by the University of California, LANL and LLNL design the nuclear explosive systems used in the weapons.

Sandia does not manufacture or assemble weapons; this work is performed by other contractors, using design information provided by Sandia, LANL, and LLNL. After the weapons reach stockpile, Sandia quality assurance evaluators periodically obtain representative samples and test them in laboratory and field exercises to ensure that they continue to operate safely and reliably. Procedural manuals, used by military personnel with the weapons in the field, are prepared and verified. Sandia is thus involved with a nuclear weapon from its inception until its retirement from stockpile.

Several important changes have occurred in nuclear weapon research and development over the years. The Limited Test Ban Treaty that went into effect in 1963 halted nuclear explosion tests above ground, causing the weapon laboratories to focus their attention on underground testing and on developing means of simulating nuclear weapon effects in the laboratory. The latter development has led to expanded use of nuclear reactors and particle beam accelerators to study the effects of radiation on materials. A new facility at Sandia, the Simulation Technology Laboratory, provides improved capabilities for testing nuclear weapon systems for vulnerability and survivability within intense radiation environments.

Weapon designers have been challenged in recent years with the requirement to develop smaller, more versatile weapons that can be used with a



Saturn x-ray simulation machine is used to test effects of high x-ray radiation doses on weapon components

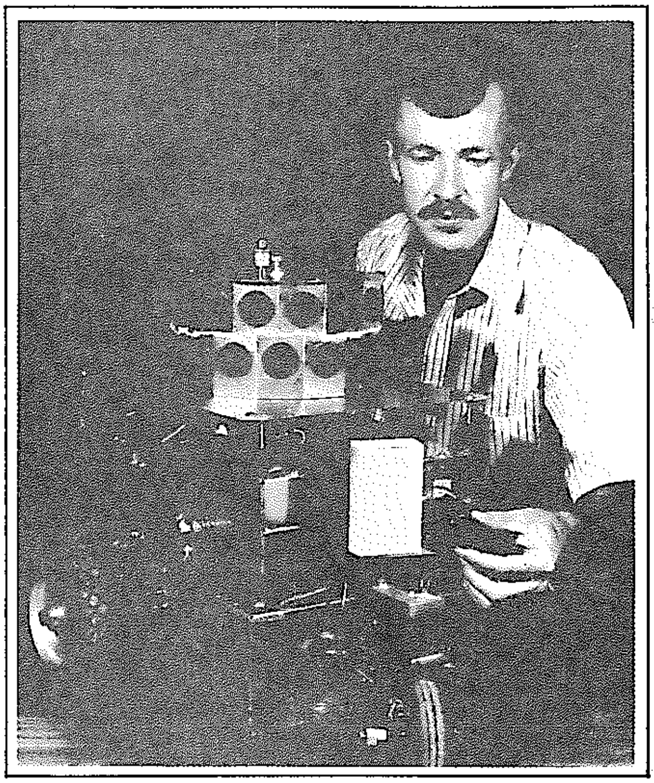
variety of new delivery systems. At the same time, designers also are involved in development of weapons that operate reliably in all types of environments, even after years of deployment or storage, and to improve systems that protect against accidental or unauthorized detonation.

To enhance military deployment strategies, Sandia develops nuclear weapon concepts that provide a variety of delivery options. For example, systems that penetrate a short distance into the earth before detonating are an outgrowth of Sandia earth-penetration (terradynamics) technology. Such weapons could be used to create barriers and craters to impede movement of equipment and personnel or to destroy targets below ground.

The Labs has made major contributions to the development of smaller, lighter missile warheads containing miniaturized components. The integrated arming, fuzing, and firing systems of such missiles as the Navy's Poseidon and Trident rely heavily on this technology.

Sandia has pioneered concepts and hardware that prohibit unauthorized use of nuclear weapons. One such concept involves security locks, generally known as Permissive Action Links, which contain sophisticated microelectronic logic systems and are small enough to be embedded in critical weapon components. Another development enables a weapon to disable or destroy itself if it detects tampering.

In addition to developing security measures internal to nuclear weapons, the Labs conducts an extensive program of research, development, and implementation of safeguards and security techniques to protect weapons and nuclear materials, in transit and at sensitive facilities. The goal is to protect such items from terrorist, insider, and other threats. The program has two key elements. First is the development of components for intrusion sensing and assessment, access denial, entry control, communications, and response. The second involves the application of systems engineering principles to combine components into a coordinated, operational system.



The Sandia Interior Robot was built to research the possibility of using autonomous robots as roving sentries inside high-security buildings

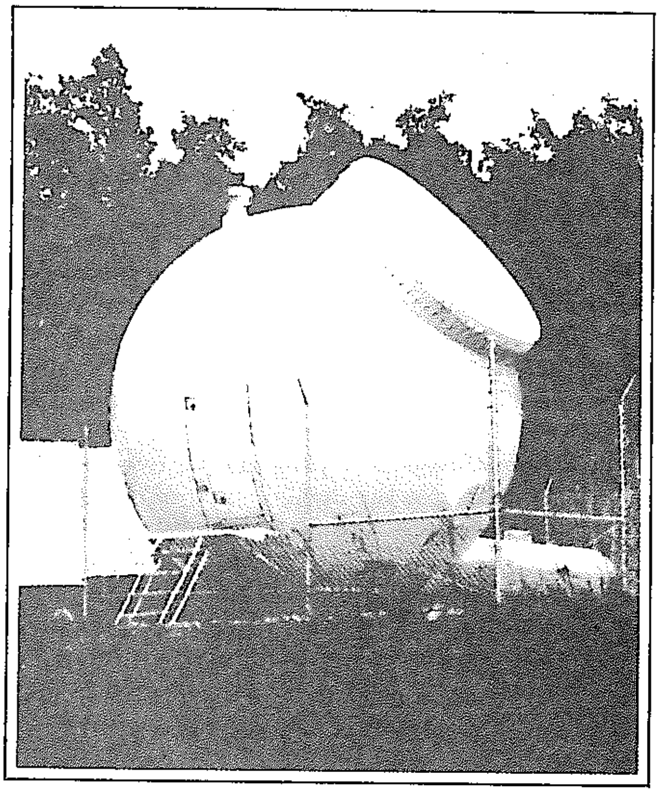
OTHER DEFENSE PROGRAMS

Verification Technology

Sandia has the largest arms-control verification technology program in the country and has contributed significantly to the development of major satellite and ground-based verification systems.

The work began with the early development of satellite instrumentation for monitoring arms-control agreements. Since then, the Labs has maintained a leading role in reliable miniature electronics and sensor technology. Sandia provided data processing logic and sensor systems for the Vela satellite program, begun in 1963 to detect nuclear detonations in the atmosphere and in space. Sandia continues to design and provide similar systems for satellite programs that include the Global Positioning System (GPS) multi-satellite constellation.

In ground-based systems, Sandia pioneered automated seismic stations used to monitor nuclear weapon testing. The Labs' verification technology expertise is now being applied to develop systems for monitoring compliance with proposed agreements to limit nuclear weapon delivery systems.



Unmanned automatic seismic stations like this Sandia-developed prototype are now deployed to monitor worldwide compliance with weapons test ban treaties

Other Weapon Systems

Technologies developed to satisfy DOE-funded nuclear weapon, verification, and security program requirements are used for other applications, including "smart" weapon systems, and other systems for specialized Department of Defense (DoD) and National Aeronautics and Space Administration space applications.

Agreements between the DoD and DOE have increased the Labs' work in advanced conventional weapons. Work in this area combines certain aspects of nuclear weapon engineering with technologies in the developing area of intelligent machines. For example, terrain-aided navigation systems, which use inertial navigation systems and associated flight computers initially developed for re-entry vehicles, are being applied to land vehicles and aircraft.

Sandia is among the world's leaders in the development of rugged, miniaturized computers. Originally developed for weapon uses, the

SANDAC (for Sandia Airborne Computer) puts the computational power of a desk-sized general-purpose computer into a very small package. The latest model, SANDAC V, is a full 32-bit multi-processor design. A three-parallel-processor unit in a navigational configuration uses only 30 watts of power, weighs just 5 pounds, and occupies about 210 cubic inches (less space than a shoe box). SANDACs are used to control re-entry vehicles and to help pilots of complex helicopters manage numerous control and weapon functions under battle conditions. The miniature computers are being considered for ground-based and space-station robotic systems.

Pulsed-Power Research

Sandia is the DOE center for pulsed-power research. This technology is applied to produce intense pulses of energy for feasibility studies of controlled thermonuclear fusion, for nuclear-weapon-effects simulation, and for directed-energy weaponry research for the Strategic Defense Initiative (SDI) program. The Saturn and HERMES-III accelerators, capable of 20-terawatt (TW) power levels, are being developed for nuclear-weapon-effects testing as part of the Simulation Technology Laboratory.

Sandia's Particle Beam Fusion Accelerator II (PBFA II) is the world's most powerful particle accelerator and is considered to be the first machine with the potential for igniting a controlled fusion reaction in the laboratory. PBFA II consists of 36 pulsed-power modules, arranged around a central hub, which deliver power from all directions. Built at a cost of $48 million, PBFA II delivers at least 100 trillion watts of power. Sandia is DOE's lead laboratory for inertial confinement fusion experimentation with light ions. Considerable collaboration takes place with the Naval Research Laboratory, Cornell University, LANL, and LLNL.

Pulsed power is applied to developing directed-energy weapons using lethal electron beams for endo- and exo-atmospheric applications, microwaves, optical lasers, x-ray lasers, and hypervelocity projectiles. These technologies will be studied and developed in Sandia's new Strategic Defense Facility, scheduled to be fully operational in 1991.

Strategic Defense Initiative

The Labs is involved in laboratory and field research to study concepts that may be used as a

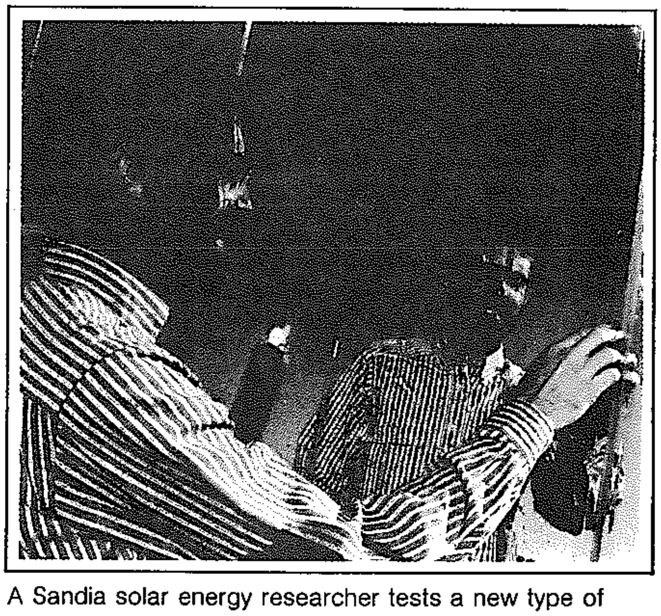
part of the Strategic Defense Initiative program. In addition to pulsed-power research and technology, SDI work at Sandia includes research in physics to establish the feasibility, in mission analysis to study the viability, and in engineering to determine the producibility of concepts. Analysis and testing are done to determine the survivability and vulnerability levels of prospective weapon systems. Specialized design of various components and materials also is done. Sandia's SDI projects for DOE are with nuclear-based systems; other, non-nuclear SDI work is done for DoD. A three-building Strategic Defense Facility is being constructed in Albuquerque, and a Defense Engineering Laboratory is planned for Livermore.

ENERGY PROGRAMS

Solar/Wind

Improved use of solar energy is being investigated by Labs researchers, who have major roles in several of DOE's solar R&D programs. The Labs' lead role in photovoltaic conversion includes crystalline cell research and development, photovoltaic (PV) concentrator development, and systems research and evaluation. The PV Design Assistance Center and the PV Advanced Systems Test Facility are both operated by Sandia for DOE.

Sandia also is the DOE lead center for the Solar Thermal program and operates the nationally recognized Solar Thermal Test Facility in



A Sandia solar energy researcher tests a new type of lightweight heliostat mirror made by gluing silver polymer film over very thin metal

Albuquerque. This facility is developing and evaluating direct-absorption central receivers, stretched-membrane heliostats and parabolic dishes, advanced engines for dishes, thermochemical heat transport, and high temperature photo-thermal chemical processing.

In the Wind Energy Program, Sandia is evaluating advanced airfoil technology for vertical axis wind turbines near Amarillo, Texas, and provides research support on viscous flow and material fatigue for various wind systems.

Fossil Energy

Sandia has a number of projects related to fossil energy extraction or conversion technology. One such effort is the Multi-Well Experiment located in the Piceance Basin of Colorado. The objective of this project is to develop technology for locating and extracting natural gas trapped in tight, lens-shaped sandstones of the western US.

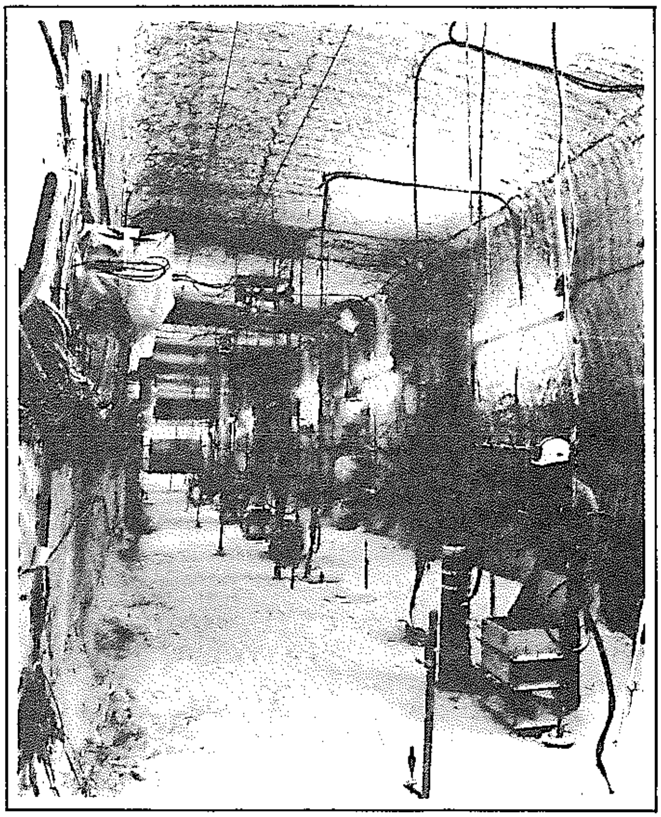
Coal science projects are directed toward increasing the understanding of coal structure, chemistry, and reactivity for various applications. Emphasis is placed on using advanced catalysts to pretreat or liquefy coal and on understanding the fundamentals of coal combustion. Computer-aided design techniques are being used to develop catalysts capable of converting methane, a gas, to methanol, a liquid fuel, in a single step.

Other projects include developing advanced diagnostic instrumentation for improved oil recovery, providing geotechnology for the Strategic Petroleum Reserve, developing hard rock penetration technology and high temperature instrumentation for geothermal resource recovery, and exploring the feasibility of extracting energy from underground magma bodies.

Fusion/Fission

Sandia is involved in a range of activities for the Magnetic Fusion Energy program. Labs scientists and engineers are working with the US and international fusion community to develop materials and devices that can survive the high heat flux environment of an operating fusion reactor. Sandia staff serve as the international resource for determining estimates of the tritium inventory in the walls of an operating reactor.

Laboratories personnel are working on DOE and Nuclear Regulatory Commission projects related to



Experiment room at the underground Waste Isolation Pilot Plant in Southeastern New Mexico

the safe and efficient use of nuclear fission energy. Sandia is actively engaged in the development and application of probabilistic risk assessment methodologies for all segments of the nuclear fuel cycle. In addition, advanced reactor concepts for earth and space-based applications are being developed and evaluated.

Nuclear Waste Management

The Labs has several project responsibilities in nuclear waste management. Work extends from geologic site evaluations to repository conceptual designs, and from there to risk-assessment studies. Two large projects are in process. One is the Waste Isolation Pilot Plant (WIPP), which has been constructed for the storage of low-level nuclear waste in underground salt beds in southeastern New Mexico. The second is the Nevada Nuclear Waste Storage Investigations project, which is assessing the feasibility of storing high-level nuclear waste in underground volcanic tuff in southern Nevada. New-generation radioactive material transportation systems are designed and evaluated at the Transportation Technology Center, also located at Sandia.

FACILITIES

Environmental Testing

Extensive environmental test facilities are located in the headquarters area and in a 60-square-mile hazardous-test area 6 miles south, also on Kirtland Air Force Base. In addition to classic environmental test capabilities for component and system development, Sandia has a variety of unique facilities. Examples include two of the world's largest centrifuges; blast tubes operating from 1 to 2000 psi; a facility for impulse testing using sprayed, light-initiated high explosives; a low field, broadband electromagnetic radiation facility; a radiant heat facility; a facility to provide large quantities of molten metal oxides for studying nuclear reactor accident conditions; a 10,000-foot rocket sled track; a multiple stroke lightning facility capable of simulating natural lightning strokes; drop towers (including one to simulate water impact); and a 5,000-foot aerial cable facility for extreme impact tests.

Radiation Research

The Laboratories operates 17 major radiation facilities, which are used in radiation effects investigations, reactor safety studies and inertial confinement fusion research. The Annular Core Research Reactor, which is used for weapon effects tests and reactor safety studies, can be pulsed to a peak power of 35,000 MW and Sandia Pulse Reactors II and III to about 150,000 MW. The HERMES II electron beam accelerator, used in a variety of x-ray experiments, can be discharged in 60 billionths of a second, with a peak power of more than 1 trillion watts.

Other facilities located in the main laboratory complex at Albuquerque include a Van de Graaff accelerator and a Cockcroft-Walton accelerator. At Livermore, an advanced Tritium Research Laboratory permits safe handling of tritium compounds during a variety of experiments.

Combustion Research

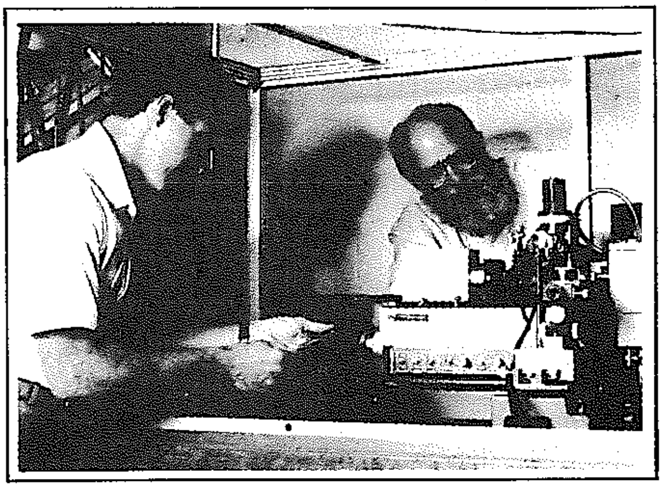
The Combustion Research Facility (CRF) at Sandia National Laboratories/Livermore develops and applies new research tools, particularly laser diagnostics and computer modeling, to fundamental and applied problems in combustion research. To encourage the transfer of advanced technology, the facility is available for use by qualified combustion researchers from other organizations.

Located outside the Sandia security area, the 51,500-square-foot facility permits easy access for visitors. The CRF has three centrally located lasers and a facility beam distribution (periscope) system. The CRF provides state-of-the-art diagnostics capabilities, access to a broad range of computers, and a sophisticated, microprocessor-controlled laboratory safety system.

Computer Systems

Scientific computing at Albuquerque and Livermore is done with a CDC CYBER 170/730, a CYBER 180/855, three CRAY-1Ss, a CRAY X-MP/24, and a Cray 416. A growing number of DEC VAX-class computers serve specific user areas. Data reduction needs are satisfied by three Data General MV/10,000s. Most business and management information needs are handled by two Sperry 1100s and a distributed network of IBM 43XX processors. A new NCUBE/ten parallel supercomputer is used for computer science research in massive parallelism. Many other computers serve specialized needs throughout Sandia.

Sandia's computer applications include more than 250 computer-aided engineering graphics workstations. Approximately 40 percent of all design definition is being completed with these stations. The Laboratories plans to attain full CAE/CAD/CAM (computer-aided engineering/design/manufacturing) capabilities by the end of the decade. Sandia is the lead laboratory to coordinate the integration of CAD/CAM capabilities within the DOE nuclear weapon complex.



These Sandia Livermore scientists worked cooperatively with Lawrence Livermore National Lab to develop this x-ray microanalyzer that allows different kinds of measurements to be made simultaneously from a single material sample

Microelectronics Facilities

Sandia is heavily engaged in specialized microelectronics research and development. Its work is concentrated in two major facilities—the Center for Radiation-hardened Microelectronics (CRM) and the Compound Semiconductor Laboratory (CSL). The CRM concentrates on conventional silicon-based microelectronics, while the CSL develops new types of materials and components made from a combination of elements.

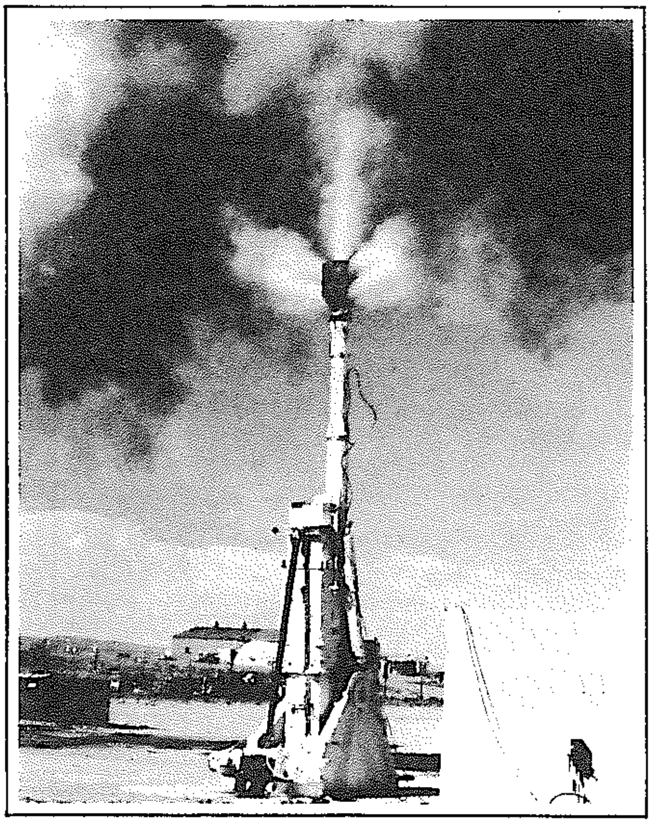
The CRM develops very large scale integrated circuits for applications requiring resistance to high levels of radiation, e.g., weapon systems, satellites and space probes. These circuits are at least 100 times more resistant to radiation than commercial counterparts. The CRM is completely equipped to design, fabricate, and test integrated circuits containing hundreds of thousands of devices. It makes substantial use of the laminar air flow clean room principle invented at the Labs in 1961, and now used worldwide in clean room airborne contamination control.

The new Radiation-hardened Integrated Circuit facility has been designed to support development and pilot production of ultra-large scale integrated circuits, using techniques that eventually will allow circuits with features smaller than 1 micron in diameter. The facility has a 12,500-square-foot Class 1 clean room that limits airborne particles to not more than one microscopic particle per cubic foot.

A significant advance in semiconductor science occurred at the CSL when a Sandia team devised the strained-layer superlattice (SLS) semiconductor. The SLS semiconductors consist of many very thin alternating layers of semiconductor materials. SLS technology allows fabrication of semiconductors with completely new electronic and optical properties.

Test Ranges

The Tonopah Test Range (TTR), a major test facility for DOE-funded weapon programs, is a permanent outdoor testing laboratory with unique capabilities for gathering data from a variety of test vehicles. This 577-square-mile area is located on the north end of the Nellis Bombing and Gunnery Range and about 32 miles southeast of Tonopah, Nevada. The range provides state-of-the-art integrated instrumentation systems for test vehicle tracking and data acquisition. Multiple radars, optical trackers, telemetry stations, a central computer



Test gun fires 8-inch diameter projectiles at the Tonopah Test Range

complex, and communication systems provide full coverage for tests involving artillery fired projectiles, rocket-launched payloads and high-performance aircraft-delivered test units. Although the range's primary mission is to support DOE programs, a number of DoD development programs have used TTR's unique test capabilities and instrumentation network. In addition to the airborne vehicular tests, various areas of the range have been used for explosives studies involving blast effects, case ruptures, shock wave phenomena and cratering.

The Kauai Test Facility (KTF) in Hawaii, resident on the Pacific Missile Range Facility, has a rocket-preparation and launching capability for high-altitude scientific research and re-entry vehicle studies. The emphasis at the KTF is on research and development of materials and components. Various small rockets are launched to test various types of non-nuclear payloads. Recent DOE-sponsored launches have included exploratory and development testing of water entry penetrators.

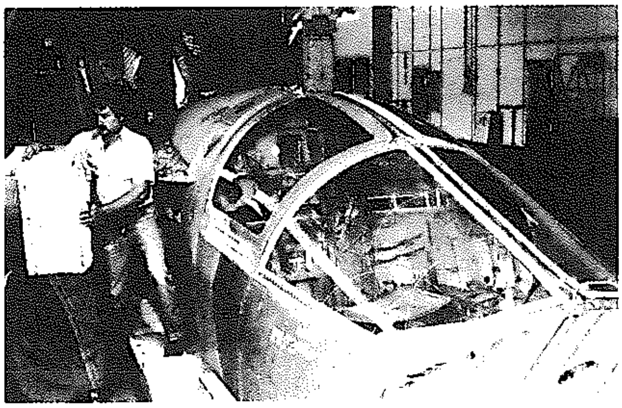
TECHNOLOGY TRANSFER

Sandia actively transfers tax-funded technology to private enterprise, universities, and state and local government. US enterprise is the primary beneficiary of Sandia's Technology Transfer program; national security is enhanced by improving commercial capabilities and opportunities. The Technology Transfer and Management Department (telephone 505-846-4945) is the contact point for the program.

Technical staff members from throughout the laboratories supply information in both proactive and responsive efforts. Some of the methods used are conference presentations; news releases; publication of papers, journal articles, and reports; sponsorship of meetings on subjects of particular interest; alliances with industry; hosting visitors and conducting visits to unclassified facilities; and answering questions received by mail or telephone. Sandia interacts extensively with universities by sponsoring research programs and allowing laboratory staff to teach selected courses.

Some technology transfer activities have had significant economic impact. For example, Sandia scientists and engineers recognized some "weak links" in petroleum recovery and developed techniques that now save the industry hundreds of millions of dollars annually. Other examples of successful transfer activity include laminar air flow clean room technology, advanced integrated circuit and semiconductor technology, vertical axis wind turbine designs, and solar energy technology.

A 14,000-square-foot Technology Transfer Center, completed in 1984, is used to host meetings between Sandia personnel and industry representatives. The Labs spends about 1 percent of its annual research and development budget on technology transfer activities.



Sandia specialists are designing and testing a new parachute system that will slow the descent of the crew escape module on an F111 aircraft to 25 feet/second

STATISTICS

SANDIA EXPENDITURES—FY 1987

	($ in millions)
Defense Programs	$ 626
Conservation and Renewable Energy	39
Fossil Energy/Strategic Petroleum Reserve	11
Nuclear Energy	3
Energy Research	20
Civilian Radioactive Waste Management	23
Total—Department of Energy	722
Federal Agency Reimbursables	306
Total Operating	1028
Capital Equipment	48
Major Construction	70
TOTAL	$1,146

Payroll ($ in millions)

FY	Albuquerque*	Livermore	Total
1987	$295.7	$45.8	$341.5
1986	275.2	42.7	317.9
1985	269.8	42.2	312.0
1984	249.1	40.3	289.4
1983	231.1	37.7	268.8
1982	208.5	34.2	242.7
1981	187.4	30.4	217.8
1980	165.5	26.8	192.3
1979	148.1	23.8	171.9
1978	134.2	21.8	156.0

Number of employees

FY	Albuquerque*	Livermore	Total
1987	7,317	1,080	8,397
1986	7,216	1,061	8,277
1985	7,179	1,074	8,253
1984	7,326	1,101	8,427
1983	7,024	1,106	8,130
1982	6,862	1,088	7,950
1981	6,935	1,083	8,018
1980	6,792	1,055	7,847
1979	6,588	1,030	7,618
1978	6,470	1,008	7,478

*includes Tonopah and Nevada Test Site

SANDIA NATIONAL LABORATORIES TECHNICAL CAPABILITIES

SCIENCE AND ENGINEERING

Physical Sciences
Physics of Surfaces and Near-Surface Layers
Physics of Solids
Computational Physics and Mechanics
Quantum Chemistry
Solid Dynamics and Shock Wave Phenomena
Interaction of Radiation with Matter
Laser Physics
Particle Beam Research
Research Reactors
Superconductivity

Applied Mathematics
Discrete Mathematics
Mathematical Physics
Theoretical Mathematics
Parallel Processing
Statistical Analysis
Cryptology
Artificial Intelligence
Optical Computing
Computer Sciences
- Parallel Processing
- Artificial Intelligence
- Computer Engineering

Reliability Assurance
Reliability Assessment
Statistical Design and Analysis
Human Factors
Quality Control

Electrochemistry
Electrochemical Power Sources
Physical Chemistry
Thermal Analysis
Corrosion

Earth Sciences
Numerical Methods
Geophysics
Analysis of Faults
Geochemistry
Drilling, Magma, and Diagnostic Technologies
Rock Mechanics

Materials and Processes
Metallurgy
Composites
Surface Characterization and Film Deposition
Polymers
Ceramics and Glasses
High-Temperature Characterization of Materials
Materials Analysis

Electronics
Compound Semiconductors
Microcircuits
Pulsed High Energy Technology
Electro-optics
Radiation Hardening
Microwave Devices
Imaging Radars

Electromechanics
Guidance and Control
Robotics
Dynamics Mechanisms Analysis

Aerosciences
Aeroballistics
Aerodynamics
Atomic and Fluid Physics
Aerothermodynamics
Aerophysics
Atmospheric Environments
Fluid Dynamics
Aeromechanical Design

Combustion Sciences
Combustion Diagnostics
Physics and Chemistry of Combustion
Combustion Modeling
Coal and Engine Combustion
Pollution Control

Explosives
Detonation Physics
Chemistry of Explosives

Quality Assurance
Program Implementation
Test Program Development
Product Acceptance and Production Control
New and Field Material Surveillance
Data Base Management

Educational Background of Technical Staff Members (MTS)

Technical Fields	
Electrical Engineering	779
Mechanical Engineering	636
Physics	367
Computing	234
Chemistry	154
Nuclear Engineering	122
Chemical Engineering	106
Mathematics	99
Civil Engineering	86
Aeronautical Engineering	80
Materials Science	67
Earth Sciences	52
All others	162
Total	2944

Highest Degrees Attained	
Doctorate	1161
Master	1330
Bachelor	407
Associate	41
None	5
Total	2944

Educational Level of Administrative & Technical Support Staff (MLS)

Highest Degrees Attained	
Doctorate	50
Master	421
Bachelor	174
Associate	16
None	46
Total	707

Personnel by Categories

Technical Staff (MTS)	2944
Administrative & Technical Support Staff (MLS)	707
Section Supervisors	164
Technical Staff Associates	361
Senior Technical Aides	1048
Technical Aides	758
Staff Assistants Technical	62
Management Aides	589
Graded Employees	1764
Total	8397

Note: Information current as of September 24, 1987.

ANALYSIS AND TESTING

Systems Analysis
Problem Definition and Assessment
Concept Formulation
System Evaluation
Concept Optimization

Engineering Analysis
Structural Mechanics
Stress Wave Analysis
Explosives Technology
Heat Transfer
Mechanism Analysis
Environment Analysis
Controls Engineering
Reactor Safety Analysis
Solids Modeling

Testing
Environmental Simulation
- Mechanical Loading
- Radiation Loading

Development and Evaluation
- Aerosciences
- Material Response
- Nondestructive Testing
- Solar Energy Systems

Field Testing
- Tonopah Test Range
- Nevada Test Site
- Kauai Test Facility
- Mobile Facilities
- Remote Ranges

Safety Assurance
Safety Assessment
Safety Assurance
System Safety

Instrumentation
Data Acquisition
- Transducers
- Telemetry
- Communications
- Optical Recording
- Mobile and Transportable Instrumentation

Special Instrumentation
- Meteorological
- Photometric
- Laser Diagnostics
- Quality Assurance
- Radiation Analysis
- In Situ Measurements and Control

SUPPORT FUNCTIONS

Fabrication
Metals
Glasses and Ceramics
Composites and Plastics
Electrical Components
Heat Treating and Finishing

Information Science
Information Management
Information Dissemination
Reference and Translation

Computation Systems
System Planning, Development and Support
Central Computing
Remote Computing
Interactive Graphics
Computer-based Special-Purpose Systems

Design Definition
Design Drawings and Specifications
Computer-Aided Design
Information Control Sytems

Measurement Standards
DOE Standardization Program Management
Direct-Current Electrical Quantities
Microwave Quantities
Radiation
Mechanical
Environmental

Environmental Health
Hazard Control
Radiation Dosimetry
Radiation Counting
Effluent Documentation

Sandia National Laboratories Organization

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O. E. JONES
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20
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Secretary & Treasurer
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- Information Integrity

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- Education and Training
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- Benefits
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- Purchasing
- Purchasing Planning and Services
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- Pulsed Power Applications
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- Applied Mathematics
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