**MECHANICAL ENGINEERING DEPARTMENT
QUARTERLY REVIEW**

April through June 1978

Scientific Editor: R. G. Stone
General Editor: M. B. Bathgate

June 30, 1978

Work performed under the auspices of the U.S. Department of
Energy by the UCLLL under contract number W-7405-ENG-48.

MASTER

UNCLASSIFIED

LAWRENCE LIVERMORE LABORATORY

University of California Livermore, California 94550

UCRL-50016-78-2

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In work for the Nuclear Regulatory Commission, the Laboratory is developing an assessment procedure to evaluate the effectiveness of a potential nuclear facility licensee's material control system. The key in the procedure is the generation of adversary event sets by a directed graph and fault tree methodology.

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INTRODUCTION

This issue of the *Mechanical Engineering Department Quarterly Review** covers the activities of Nuclear Test Engineering Division (NTED). The division consists of four sections: Diagnostics, Nuclear Explosives Systems and Energy, Engineering Mechanics, and Mechanical Technicians. The Division's activities are carried out in three main areas: nuclear testing, energy-related programs, and reimbursable programs for the U.S. Nuclear Regulatory Commission.

Nuclear Testing

Our nuclear testing activities involve device systems, diagnostics, containment, and special projects. We are responsible for the design, structural evaluation, fabrication, and proof-testing of device canisters, which position and protect nuclear explosives in underground environments. We synthesize, procure, and certify the arming and firing system, and are responsible for the necessary bunkers, cables, and interface with the control room. At the ground-zero site, we enforce appropriate controls over the nuclear explosive and firing system to comply with the Department of Energy safety rules. The necessary diagnostic instrumentation is provided by us, and it requires the development and fielding of mechanical equipment and the designing of downhole emplacement systems. An article beginning on page 4 describes our design of DCARR, a spectrograph for measuring low-energy x rays.

We have the task of developing containment systems that meet zero radioactivity release requirements. In addition, we are working toward advancing the technology and improving the materials and hardware. This involves carrying out ground shock analysis, thermal fluid investigations, gas flow studies, and structural analysis. Several special projects are being pursued. One, the development and fielding of a prompt post-shot cavity gas-sampling system, is discussed in an article beginning on page 9. Another special project is the development of an Integrated Planning System, which will produce milestone charts, optimize resources, and improve the logic in establishing the nuclear explosives test schedule.

Energy Related Programs

We are participating in three major energy-related programs: oil shale, coal gasification, and liquefied natural gas spill effects. A principal objective in the Oil Shale Project is the development and verification of a computer model of the Rubble-In-Situ Extraction process. Our role is to provide the engineering design of several retorts and the necessary instrumentation for collecting the data for the verification. The engineering design addresses the retort vessels, gas control system, product collection system, and shale preparation. In an article beginning on page 14, we describe our recently completed thermal analysis of a retort to better understand the backheating phenomenon. Our forthcoming participation will include actual commercial retorts. For the Coal Gasification Project, we are providing the engineering, well drilling, surface plant piping, and necessary instrumentation. We will also be providing test facility management and operational support during testing. We have developed a traveling thermocouple probe, described in the article beginning on page 20. Our involvement with the Liquid Natural Gas Spill Effects Project includes the engineering for site selection and facility design for the planned 1000-m³ spill test at the Nevada Test Site. In addition, we are designing and fielding the instrumentation for the 5-m³ test facility at China Lake. Other, less major, projects supported by NTED include Geological Waste Isolation, DNA Shock Tube Studies, and Geophysics.

Reimbursable Programs for the U.S. Nuclear Regulatory Commission

The safety of nuclear power plants and associated facilities is the major concern of the U.S. Nuclear Regulatory Commission (NRC). Over the past five years, our interaction with the NRC has grown significantly. We have been involved with over 40 separate reimbursable contracts concerning more than 30 technical areas; the areas are listed in Fig. 1 with a diagram of the nuclear fuel cycle. Most of the facilities of the fuel cycle have been addressed in our work. These include fuel processing and fabrication, power generation, interim waste storage, and spent fuel reprocessing. Our work can be grouped into three categories: seismic engineering, applied mechanics, and systems engineering and risk estimation. In seismic engineering, we evaluate and develop analytical techniques, establish design criteria and design requirements, carry out dynamic structural analyses, assess the seismic

*Previous issues were called *Mechanical Engineering Department Quarterly Reports*.

Nuclear Fuel Cycle Facilities	Past & Present *	Future	Enr†	CTF†	React	SFS†	Rep†	WS†
<u>Seismic engineering</u>								
Nuclear fuel reprocessing and fabrication plants	•						•	
Soil-structure interaction	•		•	•	•	•	•	•
Spent-fuel storage pools	•					•		
Submerged structures	•				•	•		
Quantification of design conservatism	•				•			
Nuclear fuel services reprocessing plant	•						•	
Seismic analysis of a mixed-oxide fuel fabrication plant	•		•	•				
Analysis of D. to Canyon Site response spectrum	•				•			
Effects of abnormal natural phenomena on existing plutonium fabrication plants	•		•	•				
Site specific spectra	○	•			•			
Seismic safety margin research program	○	•			•			
Systematic evaluation program		•			•			
Technical assistance on generic plants		•	•	•	•	•	•	•
Parameter estimation technique (PARET)	○	•			•			
<u>Applied mechanics</u>								
BWR containment studies	○	•			•			
Response of PWR feedwater piping to water hammer	•				•			
Modeling considerations in the vibration analysis of reactor internals	•				•			
Transportation of spent nuclear fuel rods		•				•		
Tornado-borne missile impact studies	•		•	•	•		•	
Technical assistance on generic plants		•	•	•	•	•	•	•
Missile impact effects on structural barriers		•			•			
<u>Systems engineering & risk estimation</u>								
Seismic trip studies	•				•			
Material control study	○	•			•	•	•	•
Waste management program	○	•				•	•	•
Reactor safety study	•				•			
Engineering support for seismic safety margin research program					•			

* In this column, ○ indicates an active program and • indicates a completed program.

† These are defined in the above picture.

Fig. 1. Past, present, and planned future reimbursable projects for the Nuclear Regulatory Commission.

hazard at specific sites, and quantify the conservatism in the design of nuclear power plants. Our work quantifying conservatism is presented on page 26. Our applied mechanics activities include scale model testing, fluid-structure interaction studies, transportation of hazardous materials investigation, and mis-

sile impact effects studies. A current project analyzing casks used in transportation of nuclear fuel rods is described on page 29. Our work in systems engineering and risk estimation involves seismic trip studies, materials control (see article page 36), waste management, and reactor safety.

DCARR: A SPECTROGRAPH FOR MEASURING LOW-ENERGY X RAYS

One of the more interesting challenges that is assigned to a diagnostic engineer is to be intimately involved in the design from conception to fruition of a new detector system.¹ For the last 2 years, we have been involved in such a design with DCARR, the Differential Critical Angle Reflection Refraction detector system.² This detector was designed to measure low-energy x rays, 500 to 5000 eV, with a high degree of resolution, 250 eV. DCARR was developed because these low-energy measurements are of interest in the diagnostics of x-radiation in nuclear tests and available equipment could not make measurements at this low an energy in field tests. DCARR is a versatile piece of equipment that can also be used as a laboratory tool, such as in measuring the low-energy x rays emitted by lasers and various x-ray machines.

The First DCARR Design

Our developmental work has involved the design of two physics configurations for the detector. Delineation of the energy bin to be measured is accomplished principally by use of critical angle reflection. Figure 2 shows the original design. The detector, also shown in Fig. 3, is approximately 100 mm by 100 mm by 1.27 m. In this design, Point 1 shows the representative continuous spectrum entering the detector system. Point 2 shows the first reflected spectrum determined by the mirror angle. Point 3 shows the second reflected spectrum determined by the scintillator angle. E_2 must be less than E_1 . Point 4 shows the refracted spectrum, the difference between E_1 and E_2 , and the part we want to measure. Notice how concise the ends are. Sharpness of the energy cut-off depends on both the flatness and roughness of the mirror surface. With this detector system, E_1 - E_2 can be made as narrow or as broad as desired (within 4 keV) by selection of critical angles. When the reflection angle is small, x rays with energy from zero to a cut-off energy dependent on the reflectance angle are totally reflected. For the carbon mirrors used in DCARR, the cut-off reflectance angle varies from approximately 20 minutes for 5-keV x rays to 1° for 1 keV. Reaching this lower limit was one of the most challenging parts of the design because it required working to angles of less than a degree holding to a tolerance of ± 1 minute. (One normally work with angles greater than 5° with angular tolerance of 10 minutes.) Those x rays not reflected by the scintillator surface are absorbed by the scintillator which fluoresces, with the amount of

light produced recorded through the use of a photodetector and appropriate recording system.

In the design of the first detector, a plastic fluor is used as the scintillator and must be inside the vacuum. The photodetector—either a photodiode or photomultiplier—must be outside the vacuum system. The plastic fluor has to be cemented to a glass substrate in order to polish the fluor flat and smooth. The substrate also acts as a light pipe, a vacuum barrier, and the surface for holding the scintillator angle. Tests have indicated that because of the differences in thermal expansions between the plastic and glass it will cause the bond to break at about 46° C.

For the first field application this September, it was planned to bury the detectors in grout which could reach a maximum temperature of 66° C during the hydration process. This is unacceptable due to bond breakage. Also, grout has a tendency to move in strange ways as it solidifies. To reduce these effects, the detectors will be placed in sand plugs. A thermodynamics study³ showed the maximum sand plug temperature as expected is 42° C. These detectors will be instrumented in the field test to verify the calculations.

The Alternate DCARR Design

Because the photodetectors are extremely sensitive and the scintillator is susceptible to background radiation, the configuration shown in Fig. 2 was judged to be too sensitive for the large signals expected in nuclear tests. We decided to use an XRD detector for nuclear explosive tests and an alternate design (shown in Figs. 3 and 4) was conceived. This design is identical to the first one except for the XRD detector and that a filter is used to define the lower energy of the bin. The filter material and thickness control the cut-off. Note in Fig. 4 that the cut-off is as sharp as in Fig. 2. This is because attenuation of the x rays by the filter is energy-dependent.

The existing XRD detector design was incorporated into the detector system. This design uses a beryllium cathode and a copper anode with two coaxial feedthroughs through the vacuum barrier. The anode has a positive bias voltage applied and when the x rays are absorbed in the beryllium cathode electrons are emitted and collected on the anode, thus giving an electrical signal from the cathode proportional to the amount of x rays absorbed.

Theoretically, use of the XRD detector in the critical reflection angle mode should be excellent. But, because the XRD detector is insensitive and

Fig. 2. DCARR physics configuration and spectral response.

x-ray intensities produced by laboratory equipment are low, we have been unable to verify this in lab tests. Plans are to verify this principle on some future nuclear event by comparing results from the XRD detector to the original model DCARR detectors in the same energy range, which can be calibrated.

During tests of XRD detectors in a reflection mode, it was found that the XRD efficiency improved to a maximum at an incidence angle of 3° to 5°.

Common Design Features

From an engineering standpoint, the crux of both detector systems is the carbon mirror and slits, not only in manufacture but also in placement. The slits define the incidence beam to eliminate scattering from surfaces other than the mirror. Also they prevent the incident beam from striking the detector which is particularly important in the measurement of higher energy x rays (smaller angle of incidence). It is important that the incident beam of x rays strike the mirror at the prescribed angle both during calibration and in the test application. So a base line, which passes through both slits and the center of the mirror surface, was established for fabrication and

alignment of the detector system during calibration and in the field. For a 50-mm-long mirror, the maximum slit widths will range from 0.3 mm to 0.76 mm, depending on the incident angle of the incoming beam to the mirror surface. We considered making the mirror small enough to act as a slit, but this introduced other problems that were more serious than the manufacture of the slits.

Slits used for laboratory experiments are made of thin tantalum and the slit openings approach the maximum 0.76-mm slit size. In the field units where the intensities of x rays are orders of magnitude larger, slit widths are reduced to 0.058 to 0.28 mm, with manufacturing and assembly tolerances of ± 0.0064 mm. Since the slit is made up of several materials approximately 7.6 mm thick, it will be necessary to have a taper of approximately 0.1. Initial tests indicate that the slits 0.18 mm or larger may be made by the EDM process and those smaller will have to be made by grinding two "L" shaped pieces for each material. Assembly of the slits must be made using a comparator microscope.

The slits can be aligned for detector calibration by taking the mirror out, turning the x-ray machine on, and adjusting the detector for maximum beam intensity. For field applications, the initial alignment

Fig. 3. The DCARR detector mechanical assembly for both physics configuration.

is made optically, which requires the removal and re-installation of the slits and mirror. For the detector calibration to remain valid, the slits must be reinstalled in their original positions and the mirror must maintain the prescribed angle, all within a very tight tolerance. This is accomplished by precision pinning. As a final alignment verification, the system is evacuated and a piece of film behind each slit is exposed by an x-ray machine placed at the point of interest for the detector. After the film is processed, the

two image sizes are compared. If they are the same, everything is ready; if not, the detector will be adjusted until the image sizes are the same.

Current Status

We have built, calibrated, and installed at the Nevada Test Site the first eight detectors (see Fig. 3). These will be tested in a nuclear field event scheduled for early September. One of each type will be used to

Fig. 4. Alternate DCARR physics configuration and spectral response.

measure background, and five of the alternate XRD detectors and one of the original design will be used for low-energy x-ray diagnostics. Following the field

test, we will determine if any significant design changes are needed, and we will study methods to simplify the design, thereby reducing costs.

Fig. 5. Downhole system.

GAS SAMPLING A POST-SHOT CAVITY

We have designed a gas sampling system that allows immediate recovery of radioactive gases from the cavity produced by a nuclear test at the Nevada Test Site. The samples are analyzed by the Radiochemistry Department to determine a number of parameters about the test, such as: (1) the fast neutron flux inside the device by observation of the $^{86}\text{Kr}(n, 2n)^{85}\text{Kr}$ reaction, and (2) the device thermonuclear yield by determining the amount of ^4He formed.

While usable information can be obtained with gas samples from the post-shot drill hole, the short half-life of many of the gases being analyzed makes a system that can be used immediately after the test extremely valuable. This dictates that the channel for conducting the gases should be emplaced before the nuclear test. Designing a system which can survive the violent ground motion associated with the test and still present no change of cavity gas escape was a challenging problem which took many years of development to solve.

The first attempts at gas sampling began in 1967. A number of different approaches have been used to carry the gas to the surface. For instance, initial systems used the emplacement pipe filled with steel balls or rock. Later, hoses filled with mercury were tried. Success rates with these systems were on the order of 30% and limited to the smaller shots.

The Gas Sampling System

We finally developed the present gas sampling system which consists of the downhole system that collects and transports radioactive gas uphole to the uphole system that pumps and stores it. Many successful tests have been carried out with this system.

The Downhole System. The downhole system is shown schematically in Fig. 5. A nonrotating 1-in. diam wire rope, encased in a 1-1/4-in. heavy duty hose, extends down into the emplacement hole. The wire rope inside the hose prevents it from completely collapsing. The sample gas is transported in the annulus between the wire rope and hose as shown in Fig. 5. The wire rope extends to the probable cavity radius. It has a rated strength of 350,000 N, which is well above the combined weight of the downhole system and stemming load.

The 1-1/4-in. high-pressure hose does not extend all the way to the cavity. It was feared that the high temperatures from device detonation might melt the rubber in the hose and completely seal it off from the cavity gas. We rely on the interstices within the wire rope and stemming material to carry the

cavity gas to the end of the hose. The hose itself is segmented into approximately 10-m lengths. Each segment ends in 37° female flare fitting. The segments are joined together by a special male union. These unions are in turn anchored to the wire rope by three set screws. The stemming load and weight of the hose are thus transferred to the wire rope.

The next feature within the downhole system is the vacuum demand valve (Fig. 5), which prevents sample gas from flowing in the hose unless it is actively drawn up by the uphole system. It is designed so that the pressure in the hose above the vacuum demand valve is always below atmospheric pressure. A simplified representation of the vacuum demand valve is shown in Fig. 6. Before the valve is emplaced, the reference chamber (1) is temporarily opened to the local atmospheric pressure at the test site. If no suction is provided by the uphole system, chamber (2) will also be at atmospheric pressure and the spring will hold the valve closed. As a vacuum is drawn on chamber (2), the net force between the two chambers eventually overcomes the spring force and the valve opens. Note that any pressure below the valve serves to make the valve harder to open. The lower vacuum demand valve is configured to open at approximately 0.85 atm while the redundant upper vacuum demand valve is set at approximately 0.75 atm.

Although the wire rope is more than capable of supporting the weight of the hose and initial stemming load, it can be broken if it is gripped at two places in the ground that have a relative motion to each other during the nuclear test or subsequent cavity collapse. Consequently, several special precautions are used to isolate the hose and cable assembly from the two rigid plugs used in all underground nuclear tests. (See Fig. 5.) The section of the hose which passes through the lower plug is covered with foam rubber and wrapped with tape. The low shear strength of the foam allows the hose to pull through this plug. In a like manner, the section of the hose that passes through the upper plug is coated with a highly viscous grease-like material called hydroseal. Its purpose is to allow the hose to slip through the plug but at the same time maintain a seal at the upper plug.

Approximately a hundred meters of extra cable and hose are snaked from the top of the emplacement hole as shown in Fig. 7. This slack allows the hose to be pulled into the hole during cavity collapse and keeps the hose from being pulled out during surface cratering.

The Uphole System. The uphole system pumps and stores sample gas from the nuclear-created cavity. Since this gas is a radioactive and explosive mixture of hydrogen, air, and inert gases, great care must be taken in handling it to assure that none escapes or is ignited. For instance, as much of the system as possible is maintained at below atmospheric pressure. For parts of the system operating above atmospheric pressure, bellows valves with redundant packed seals were used.

The uphole system shown in Fig. 8, consists of three major modules: a containment skid, containment tanks, and a sample-taking skid.

A two-stage compressor is mounted on the containment skid. It has a throughput of approximately 2 l/s which is used to clean the downhole system and sample skid of gas from the previous sample. Since this gas is radioactive, it cannot be exhausted to the atmosphere but must be stored in containment tanks. These tanks are 1900-l propane tanks typical of the type used by households with no natural gas supply. Although these tanks are rated to over 1.7 MPa, they are pressurized no higher than 0.7 MPa with excess sampling gas and are buried under 1 m of dirt to minimize the radioactive exposure of personnel. Before nuclear tests, the containment tanks are prefilled to one-third their capacity with a fire retardant gas (Halon) to prevent ignition of the hydrogen-air sampling gas stored in them.

A schematic of the major components on the sampling skid is shown in Fig. 8. A three-stage compressor in series with a metal bellows compressor is used to pressurize the sample bottles to 3.4 MPa. At this pressure, these 300-l bottles have a capacity of approximately 10 atmospheric liters of gas.

A typical sample-taking sequence occurs about 10 minutes after a nuclear test when the containment compressor is started and valve VA11 is opened (see Fig. 8). For 5 to 10 minutes, sample gas is bypassed through the sample skid and stored in the containment tanks. When instrumentation indicates it is desirable to take a sample, Valve VA19 is opened, Valve VA16 is closed, and the compressors on the sample skid are turned on. After a delay of approximately 8 s to purge the high-pressure lines on the sample skid of any residual gas, Valve VA12 is closed and the first of the sample bottle valves (VA1) is opened. After the sample bottle is pressurized to 3.4 MPa, the sample bottle valve is closed, Valves VA12 and VA16 are opened, and VA19 is closed. The nitrogen flush (VA13) is then used to clean both the

Fig. 6. Vacuum demand valve.

sample skid and the containment skid of the radioactive gas.

As a safety precaution, the system is thoroughly leak checked before the test using a SF₆ sensitive leak detector. During the gas sampling experiment, air collected within the shelters housing the uphole system is continually monitored for leaks. After the nuclear test, the shelters housing the sampling skid were ventilated and all power to the system turned off before personnel were allowed in the shelters.

Instrumentation and Command and Control. Various pressure transducers are used in the gas sampling system to monitor system operation. Other instrumentation includes two radiation monitors located on the hose between the sample skid and containment skid. Their purpose is to determine the radiation level of the gas passing through the hose to determine whether a sample of this gas should

be taken. During post-shot measurements, the radioactivity of the sample gas eventually decreases to make the radiation monitors unusable as diagnostic devices. They are then supplemented with a more sensitive ion chamber.

The main command and control station for the gas sampling system is located in the command post at the Nevada Test Site. It consists of a control panel, a televised display and a series of analog meters which provide a backup for the TV display. On the control panel is a flow diagram of the system. Pushbutton switches controlling the valves and compressors are located at the appropriate location on the flow diagram.

The colored flow diagram displayed on the TV monitors has symbols representing the valves and sample bottles that change color to indicate their status. Readings from the instrumentation displayed at the appropriate place on the flow diagram are updated approximately every second. We have found that this command and control system, especially the

Fig. 7. Uphole system.

Fig. 8. Schematic of gas sampling system.

TV display, greatly facilitates safe and efficient operation of the gas sampling system.

Future Gas Sampling System

At the present time, the major problem with the gas sampling experiment is maintaining the integrity of the downhole system during the collapse of the cavity. In the last fielding of the gas sampling experiment, samples were obtained up until the time of cavity collapse when the sampling hose broke somewhere between the upper and lower vacuum demand valves. Although usable gas samples were obtained from the post-shot drill hole after this

time, it is still highly desirable to design the downhole system so it could survive the cavity collapse.

The approach to be taken in future gas sampling experiments is to employ two separate downhole systems. One will follow present practices and be placed in stemming material. The other will be contained within the emplacement pipe. It is hoped the different environments in which these are employed will improve the probability that at least one hose will survive.

Two new gas sampling skids capable of taking 10 samples apiece are being built. They incorporate a compressor which can handle both the bypass func-

tion of the containment compressor and the sampling function of the high-pressure compressor. Gas flow will be measured by a turbine flow meter which is not sensitive to uncertainties in gas density as

is the present square-edged orifice flow meter. In general, the new sampling skids are expected to be even more reliable than the present one.

THE ANALYSIS OF HEAT LOSSES AND THE DESIGN OF A BACKHEATER CONTROL SYSTEM FOR THE LLL OIL SHALE RETORT

We have been supporting the LLL Oil Shale Program, an experimental effort aimed at obtaining a comprehensive understanding of the oil shale retorting process. The primary thrust of the four-year-old program has been laboratory research and planning for expected field conditions. The goal is providing the technical base that will be required to develop a modified *in-situ* method to a commercial scale.

In tests at the Laboratory, a 0.3-m-diam, 1.5-m-long retort is used to simulate the actual vertical underground *in-situ* process. In the field, the volume to be retorted is large so that the heat losses are small. Since the laboratory retort is small, the heat losses are significant and must be compensated for by backheating the reactor sidewall. This has been accomplished with stationary electric heaters wrapped around the sidewall every 152.4 mm. The control algorithm for these heaters uses measured peak temperatures, temperature profiles calculated by a detailed one-dimensional process model, and heat loss coefficients based on a previous steady-state test. However, the heat loss through the wall is not steady state.

We desired to improve the sidewall backheating control system to better simulate the adiabatic *in-situ* situation. Our goal was to modify the control system so it did not require the use of temperature profiles generated by the process model and to make it fully automated. We performed a thermal analysis of the retort to obtain detailed information on reactor temperatures and heat flows. This analysis also provides a tool needed in designing a new backheating control system.

Thermal Analysis Model

The 0.3-m-diam oil shale retort was modeled using the TRUMP thermal analysis code. The model, which is axisymmetric, is shown in Fig. 9. It is finite differenced using the FED computer code. The active retort length is divided into 70 increments, and the Vermiculite insulation is divided into four increments in the radial direction. The gas and shale nodes are coincident, but the gas flows downward from one gas node to the next. We used the parameters from the latest test done in the retort, Run S11, as the problem input and compared the calculated results to this run.

In the actual tests, the heat needed to decompose the kerogen in the shale to form oil is obtained

by burning the residue carbon. This combustion heat is simply modeled by generating heat in each successive shale node as the flame front moves downward. No attempt is made to model the complex chemical processes. Instead, the magnitude of this heat generation is adjusted to yield gas temperatures that agree with test data from Run S11. The power from the 10 electric sidewall heaters is controlled by the heater control analog used in Test Run S11.

Model Results and Discussion

To compare to Run S11, the model was first run with the sidewall heaters turned on (Fig. 10). The shale combustion heat generation was adjusted to yield the Run S11 temperature levels. The agreement between the test and calculated temperature profiles is good.

To yield some insights into the effects of sidewall heat losses, we ran the model without heaters (Fig. 11) and with adiabatic sidewalls (Fig. 12). In both these runs, only the ignition heater is left on.

Fig. 9. Thermal analysis model of oil shale retort.

Fig. 10. Centerline gas temperatures from the Run S11 test are compared with calculations using S11 input data. Inner and outer shell temperature calculations are also shown.

In the run with adiabatic sidewalls, all thermal connections between the gas and the inside shell and between the shale and the inside shell are severed. The connections at both ends of the retort and between the ends and the sidewall remain intact.

The comparison between the run with heaters (Fig. 10) and the runs without heaters (Figs. 11 and 12) shows a decrease of 60°C in the maximum peak gas temperature. The comparison between the run with heaters (Fig. 10) and the run with adiabatic walls (Fig. 12) shows an increase of 170°C in the maximum peak gas temperature.

The increase in the temperature of the adiabatic run is consistent with the overall energy balance of the calculational run with heaters (Table I). If the heater control analog were adequate, the energy input from the heaters should equal the heat loss to the surroundings plus the heat needed to heat up the structure (the empty retort). Table I shows that the heaters should have supplied another 33,665 kJ of energy, which is a 33.7 % deficit. Given the heat capacity of the retort, this much additional energy would have resulted in an average retort temperature increase of 130°C.

Fig. 11. Centerline gas temperatures from the Run S11 test are compared with calculations using Run S11 input data but with no sidewall heater. Inner and outer shell temperature calculations (also assuming no sidewall heater) are also shown.

Table 2 compares the calculated heater energy and heat loss with the actual test values. A direct comparison is difficult due to the additional length of the test run. In the model, the combustion front moves at a constant 2.69 m/day (Run S11 average peak advance rate). So, it takes 13.61 h to traverse the entire reactor length. The actual test run time was 17.17 h. This difference is due to the additional time spent during test ignition and shut down.

The energy balances in Table 2 show that the heaters in the calculation supply less of the required energy than do the heaters in the actual test. The heaters in the calculation supply 33.7 % too little

energy, compared to 24.6 % too little energy in the actual test. This is due to a slight difference in the manner in which the heater control algorithm is applied, as well as the difference in run time.

As a result of the heaters supplying 33.7% too little energy to the calculational model, the maximum peak gas temperature in the adiabatic run increases by 170°C. To determine what the actual maximum peak gas temperature would have been if Run S11 were actually adiabatic, we used the 24.6% shortage in heater energy for the actual test. The maximum peak gas temperature is estimated to increase 120°C or from 1040°C to 1160°C. This higher

Fig. 12. Centerline gas temperatures from the Run S11 test are compared with calculations using S11 input data with adiabatic sidewalls. Inner and outer shell temperature calculations (also assuming adiabatic sidewalls) are also shown.

Table 1. Overall energy balance from calculational model.

Heater Energy Input	Heat (kJ)
Ignition	8,987
Sidewall	57,216
Total	66,203
Energy Output	
Heat loss	74,219
Structure heat	25,649
Total	99,868

temperature would have damaged the inner stainless steel retort shell. Thus, for most of the retort runs, some heat loss must be allowed near the combustion front to maintain the integrity of the retort.

Future Heater Control

Since we gained an understanding of the retort heat loss problem through the thermal study, we could improve the sidewall backheater control system. Ideally, this heater control should minimize the local heat flow from the shale to the inner shell. However, the thermal studies showed that exact control is impossible on a local level due to: (1) the

Table 2. Comparison of the calculated heater energy and heat loss with actual test values.

	Model (kJ)	SH test (kJ)	Difference (kJ)	Difference (%)
Heater energy				
Ignition	8,987	8,987	0	0
Sidewall	57,216	78,250	-21,034	-26.9
Total	66,203	87,237	-21,034	-24.1
Heat loss	74,219	89,628	-15,409	-17.2
Run time ^a	13.0697	17.1667	-3.557	-20.7

^aIn hours.

axial heat conduction along the stainless steel inner shell, and (2) the fact that the temperature changes at the flame front take place at a distance about equal to the 152.4-mm heater spacing. An additional difficulty is that the flame front may not burn in a horizontal plane. Since local control is impossible, we decided to control the global shale to inner shell heat loss in each of the 10 retort sections surrounding each heater. Heat balances of the wall are used to determine this shale to inner shell heat loss.

Figure 13 shows the thermocouple placement for a section of retort through the heater wall. This pattern is repeated at the diametrically opposite locations. The temperatures from each pair of thermocouples is averaged.

The heat flows necessary for the wall heat balance are made using the control volume edge to center temperature differences. Between levels 5 and 6, the axial heat flows in the inner shell are determined by the temperature differences (T_1-T_2) and (T_3-T_2). The axial heat flow in the outer shell is negligible. In the insulation, the axial heat flows are determined by the temperature difference averages

$$(1/2 T_1 + T_4 + 1/2 T_7) - (1/2 T_2 + T_5 + 1/2 T_7)$$

and

$$(1/2 T_3 + T_6 + 1/2 T_7) - (1/2 T_2 + T_5 + 1/2 T_7).$$

Since the axial heat flows in the outer shell are small, the radial temperature loss out of the wall is determined by the temperature difference average

$$(1/2 T_4 + T_5 + 1/2 T_6) - T_7.$$

The change in stored wall heat is determined, respectively, in the inner wall, the insulation, and the outer shell by the time rate of change in the temperature averages

$$(1/2 T_1 + T_2 + 1/2 T_3),$$

and

$$(1/2 T_4 + T_5 + 1/2 T_6),$$

and

$$T_7.$$

Using these temperature differences and rates of temperature change, an approximate value of the shale-to-inner shell heat loss can be determined by heat balance for each retort section. The objective of the heater control is to supply sufficient heater power to make this heat loss zero. Since a minimum number of thermocouples is used, experimental constants will probably be needed to obtain the actual heater power required.

This heater control system is planned for implementation in the near future.

Fig. 13. Thermocouple location for heat balance on retort wall.

THE DESIGN AND DEVELOPMENT OF AN AUTOMATED TRAVELING THERMOCOUPLE MECHANISM

We recently designed, developed, and tested an automated traveling thermocouple assembly for use at the Hoe Creek No. 2 *In-Situ* Coal Gasification Experiment. This experiment is part of a program to develop efficient and economical *in-situ* coal-to-gas conversion technology. To develop this technology, temperature is one of the data that must be recorded and studied during the coal gasification process.

At Hoe Creek, temperatures as high as 1000°C at depths over a hundred meters below the surface of the earth had to be reliably recorded. In addition, three-dimensional temperature data throughout the coal seam were required. Fixed thermocouples, which are emplaced before the coal gasification experiments, have been a problem because they often fail during the months-long experiments. Since there is no way to replace them, needed data is lost. The automated traveling thermocouple assembly we developed overcame this problem by having a replaceable thermocouple tip.

The traveling thermocouple assembly consists of a standard thermocouple probe (Fig. 14) attached to over a hundred meters of wire that is lowered into a thermowell-lined hole at the coal-gasification site. The thermocouple wire is attached to machinery

(Fig. 15) at the top of the hole with a motor and drum mechanism for lowering the probe. This provides for quick and efficient measurements. Special design features allow for recovery of all but the bottom 15 m of the wire in event of collapse of the thermowell or other hang up of the probe.

We designed the system to meet the following specifications and constraints:

- Temperature repeatability to $\pm 5\%$.
- Temperature range of 0-1000°C.
- Static temperature measurement with a $\pm 10\%$ accuracy with respect to the point adjacent to it on the external surface of the thermowell (measured by a fixed thermocouple).
- Capability for both continuous as well as stepwise probe motion.
- Probe motion that is programmable and controlled both manually as well as automatically.
- Probe depth readout with an accuracy to ± 15 cm.
- Probe speed that averages 60 cm/s for any given move.
- Thermocouple junction that can be easily checked and easily replaced.

Fig. 14. Thermocouple cable shown with three different type probe cages.

Fig. 15. The mechanical components of the traveling thermocouple.

- Time response that is as rapid as possible — less than 1 minute.
- Surface equipment easily movable from hole to hole.
- System operation from a 115 V, 60 Hz power source.

Engineering Design

Downhole Equipment.

The Temperature Probe. For the temperature measuring probe, we chose Inconel 600 sheathed, type K, 1/16-in.-diam chromel-alumel thermocouple wire with a grounded junction. Analysis showed that no special provisions for minimizing free convection or contact resistance are necessary for downhole temperature measurement during the *in-situ* coal gasification process. Heat was therefore conducted to the probe through an open air probe cage surrounding the thermocouple tip. This served to secure and protect the thermocouple wire.

The cage was screwed to an extremely flexible conduit, shown in Fig. 14. Inside this flexible tubing ran the thermocouple wire. The conduit protected the thermocouple wire from any mechanical damage

caused by an inadvertent tensile or compressive load. It could sustain a tensile load up to failure at 1600 N. After 15 m of conduit, we attached a specially designed downhole connector (Fig. 16) to connect the Inconel-sheathed thermocouple wire to a more flexible and less expensive Teflon-sheathed extension wire. The connector was wrapped around the drum and terminated in a "cold junction." The connector was swaged to the flexible tubing in such a fashion that the tubing would break on the downhole side at 600 N of tension in case the probe or bottom 15 m of wire got caught. This let us save the rather expensive connector and made it easy to recover. Such a design is needed since the thermowell will eventually fail during the long coal gasification experiment.

The connector, probe, and sinker were all designed to easily pass through the thermowell, a 1-in. pipe with a 30-cm bend radius.

Thermowell. The thermowell provides an appropriate environment in which temperature can be measured. It must be resistant to thermal as well as mechanical failure. Since the thermowell acts as a barrier from the corrosive gases, failure of the thermowell precludes further data acquisition.

Fig. 16. Downhole connector is used at the Inconel-Teflon junction.

Although it was desirable, a pressure seal between the thermocouple cable and the thermowell was not used due to budget constraints. Since 1000°C temperatures were expected in a very corrosive environment, Inconel 600 was used for the thermowell in the heated coal zone. For economic reasons, regular black pipe was used to extend the thermowell to the surface. Two 6.4-m lengths of Inconel pipe were welded together at the Lab and installed as a unit at Hoc Creek. The Inconel pipe was connected to the black pipe via a coupling. Ordinary pipe couplings were used as connectors for the black pipe. A sleeve was used at every coupling to provide a smooth internal transition from pipe to pipe. The top of the thermowell was terminated with a ball valve.

Thermal expansion of the thermowell was provided for at the bottom. The expansion joint which we designed allowed the pipe to grow longitudinally without bowing or blocking the thermowell. The pipe needed the ability to slip with respect to the coal seam and grout. This slip was assured by preventing the grout from attaching itself

to the thermowell by lubricating the pipe with a silicone grease. Three identical thermowells were emplaced.

Above-Ground Instrumentation. Figure 15 shows the movable above-ground instrumentation.

Rotary Mechanism. The Teflon-sheathed extension wire was wound on the rotary mechanism (Fig. 15), designed to drive the drum in such a fashion as to oscillate the probe in the coal seam. A closed-loop servo positioning system, Index-Syn, accepted digital input to automatically control the traveling thermocouple. The rotary mechanism used the drum to drive the level wind mechanism. The winch used the cable to drive a sheave which was attached to a resolver and counter. The resolver provided positional feedback.

Level Wind. The level wind mechanism worked very well. This was probably the most unique mechanism on the rotary winch system. Level winds which have a short length of cable between the drum and the first sheave usually are powered through a gearbox. Since the restoring force from the cable on

the sheave is high (shown in Fig. 17a). The gearbox drives the fork at the proper speed to lay the cable efficiently on the drum. If, as in Fig. 17(b), there is a large length of cable between the drum and the first sheave, there usually is no gearbox. The cable is moved by threads on the drum. Since the angle is small through which the cable is moved, the restoring

force of the cable is small. The maximum possible force of the drum thread on the cable is larger than the existing restoring force. Therefore, the cable is wound efficiently on the drum.

One objective of the traveling thermocouple winch mechanism was to wind the cable efficiently on the drum without a gearbox for the level wind and yet

Fig. 17. Comparison of different level winds. We used the one in 17 (c).

keep the unit small and compact. In order to do this, a two degree of freedom sheave was used. Figure 17 (c) shows that the solution was to make one sheave axis collinear with the cable. By making the cables parallel to one another, no component of the restoring force was seen. By allowing the sheave to rotate co-axially with the line of action of the cable over the sheave, the level wind was formed. The cable was powered by threads on the drum.

This type of level wind works extremely well as long as it is kept horizontal and the higher loads on the sheave attaching bolts can be accommodated.

Electrical Interface. Control for the traveling thermocouple was provided by a timer circuit. Through the timer, the dwell at each data point could be manually adjusted. The original concept called for digital computer control, but we were uncertain if it would be ready in time. Thus, a timing control system was designed to serve as a backup computer and was ultimately used in the experiments.

Temperature Data Results and Discussion

The traveling thermocouple was used for over two months (October-December 1977) at Hoe Creek. Three instrument holes placed at different areas of

the burn front had thermowells in them and were suitable for using the traveling thermocouple. These three holes also had seven fixed thermocouples outside the thermowell to get data for comparison. Nine other holes were part of the temperature measurement scheme and had fixed thermocouples only.

Our original plan was to move the traveling thermocouple as the burn front progressed and leave it in each hole "permanently" until the front progressed to the next hole. In practice, the failure of the fixed thermocouples in the various instrument holes caused us to move back and forth between the traveling thermocouple holes to get data.

The majority of the data were recorded starting 9 m from the bottom of the seam and going down at 60-cm intervals. The thermocouple tip was held at each 60-cm point for 4 minutes before measurements were made. The dwell time was set based on tests at the Lab that showed the time constant was 48 s.

Figure 18 summarizes measurements made on several days at one well and is typical of our results. Note that temperatures as high as 1100°C were recorded by the thermocouple. Although the figure shows only data recorded as the probe descended,

Fig. 18. The traveling thermocouple temperature profiles for a Hoe Creek thermowell.

data were also recorded while the probe was ascending. Comparison of these two data sets showed that temperature hysteresis was present. This indicated that the time constant determined at the Lab was too short, and the dwell time should be lengthened or other corrections made. According to the Lab-determined time constant, after the 4-min dwell time the probe should have been within 99% of the steady-state temperature. The hysteresis however showed that the probe was not this close to the steady-state temperature. Apparently, thermal energy was being stored in the flexible conduit, the sinker weight, and probe cage and was not allowing the probe to respond as quickly as predicted. In our design we had minimized this mass as much as possible; further minimizing it would give a faster response. Another parameter which could be changed is using a metal with a lower capacity to store heat to surround the junction.

Although this is an area where improvement is possible, it is not crucial to the results. Even though an ideal probe would show no hysteresis, the 10% difference between the ascending and descending data did not hinder its value to the program. The probe was able to respond at a sufficiently fast rate to

record temperature gradients as steep as $14.2^{\circ}\text{C}/\text{cm}$.

The data in general appeared to be very good. The temperature peak could easily be determined. The accuracy for determining the peak is believed to be good since the average for the ascending and descending plots could be calculated. A good feel for the actual temperature is obtained by recording data while moving the probe in both directions.

Conclusions

The traveling thermocouple worked very well. All the design constraints and specifications were met. The probe never hung up in the hole. The above-ground system responded to automated commands by quickly and efficiently moving the probe to the programmed vertical position. The traveling thermocouple proved to be a rugged piece of hardware that outperformed the fixed thermocouples.

Although the traveling thermocouple is a prototype temperature measurement system built specifically for Hoe Creek 2, the design was kept as flexible as possible to allow later use at other coal gasification sites. It is a very useful system where temperature measurements are critical since unlike fixed thermocouples it can be replaced if it fails.

SEISMIC ENGINEERING—QUANTIFICATION OF DESIGN CONSERVATISMS

Seismic considerations play a major role in nuclear power plant design. Over the past five years, we have helped the Nuclear Regulatory Commission (NRC) improve and update its technical basis for licensing of nuclear power plant facilities. Present seismic criteria set by NRC regulation, guides, standards, and licensing review procedures define a seismic methodology for power-station design. This same criteria forms the starting point for NRC's review of all nuclear fuel cycle facilities. We are involved in safety-related issues covering all the key methodology areas shown in Fig. 19. Shown is an earthquake source at some distance from the plant, the seismic waves reaching the plant, and the plant responding to the ground motion. Designing a plant to withstand earthquakes requires the employment of the various methodologies embodied in the design sequence. The first four steps are illustrated in Fig. 19. The techniques of Step 1, seismic input, express the earthquake hazard in terms of free-field seismic motions based on geological seismological considerations. From the free-field seismic motion, the methodologies of Steps 2, site response, and 3, soil-structure interaction, provide plant basemat ground motion. The techniques of Step 4, structural response, are used to develop structural member

forces and/or floor spectra from a given basemat motion. Step 5, seismic stress-strain, encompasses the methods to calculate earthquake stresses from the member forces and/or floor spectra. Nonseismic loads are determined by the methodology of Step 6, nonseismic stress/strain. Seismic and nonseismic loads are combined using the procedures of Step 7, combination of seismic and nonseismic stress/strain. The methodology of Step 8 (allowable failure stress-strain/load) provides suitable failure criteria. The evaluation of structural failure, Step 9, is carried out with appropriate techniques. In general, most of our NRC programs address the adequacy of the methodologies of all nine steps, establish a technical basis for design methods, and provide recommendations for improving the methodologies.

For convenience, we categorize our seismic engineering projects under three headings: (1) the evaluation and assessment of methodologies used in the design of nuclear facilities, (2) the quantification of design conservatisms (including the quantification of the effect of compounding design conservatism), and (3) case review of specified plants and sites. In this article, we describe our efforts in the quantification of design conservatism. In quantifying design conservatism, we address the seismic design

Fig. 19. Key seismic methodology steps. Shown in the figure are: (1) seismic input, (2) site response, (3) soil-structure interaction, and (4) structural response. Other key methodology steps include: (5) seismic stress/strain, (6) nonseismic stress/strain, (7) combination of seismic and nonseismic stress/strain, (8) allowable/failure/stress/strain/load, and (9) evaluation of failure.

methodology steps of Fig. 19 and focus on two different levels. One involves an individual quantification of conservatism at the various methodology steps (Fig. 19 and Table 3) in the seismic design chain. The other involves developing a methodology to realistically combine such individual conservatisms and obtain an overall conservatism factor for the entire chain.

Individual Quantification of Seismic Design Conservatisms

To date, our work on individual quantification has addressed seismic input, soil-structure interaction, structural response, and mechanical systems. Table 3 lists the topics examined in each area.

NRC seismic input guidelines are developed with a large number of actual earthquake time histories. Conservatism in the design input comes from two sources: (1) the frequency band set by NRC guidelines encompasses the entire band of actual time histories, and (2) synthetic time histories which envelope the NRC guidelines are generally used for design. We study the first source by comparing the response of a typical structure when the input motion is defined by NRC guidelines to its response when the input motion is defined by individual time histories. Source (2) is studied by comparing the response of a typical structure when input is synthetic time histories to its response when input is NRC guidelines.

Our study of conservatism in soil-structure interaction design methodology concerns the use of linear solutions methods for nonlinear soil behavior. A commonly used example of this is embodied in the combined application of the computer codes SHAKE and FLUSH, which utilize iterative techniques to obtain "equivalent" linear solutions. The Humboldt Bay Power Plant is analyzed in our study since the only available usable seismic response records of a nuclear plant were obtained there during the June 7, 1975, Ferndale earthquake. SHAKE/FLUSH analytical results are compared using the NRC guidelines seismic input, and the actual seismograph records from the event. In the future we also plan to study the differences between linear and nonlinear techniques.

Table 3 lists topics being investigated in the next steps of the Fig. 19 design sequence, structural analysis of the plant buildings and mechanical systems. Our work on the elastic-plastic behavior of power-plant braced frames is a good example of our effort to quantify the conservatisms involved with structural analysis. We selected for study a braced frame, representative of those used in power plant construction. This was first analyzed using elastic

methods, statically and dynamically. The threshold g values were calculated for the brace design limit for buckling and for the theoretical elastic or inelastic buckling loads. We found the reserve capacity in the structure's elastic response equaled the safety factor utilized in elastic design.

The frame was then analyzed using two-dimensional elastic-plastic analyses for eight time histories. In general, the buckling and yielding of the braces controlled the response. Based on a limiting strain criterion, we found that the ultimate capacity of the frame is a 30-cm displacement. From the results of our analyses, we determined the g -level corresponding to 30-cm displacement. This displacement is defined as the ultimate capacity of the frame. By comparing the threshold g values and the ultimate values, the reserve capacity of the structure was established. Results examined to date indicate that the frame can sustain an earthquake 1.5 times greater than the current design level.

Overall Quantification of Seismic Design Conservatism

The second level of our work on design conservatism involves developing a methodology to realistically combine the individual conservatisms to obtain an overall conservatism for the entire seismic design chain shown in Fig. 19. Our activity was significantly increased in January, 1978, with the initiation of the Seismic Safety Margin Research Program, a sizable, multiyear effort sponsored by

Table 3. Study areas for individual quantification of seismic conservatisms at the various methodology steps in the seismic design chain of Fig. 19.

Seismic Input
• Wide band spectra
• Enveloping synthetic time-history
Soil Structure Interaction
• USNRC Regulatory Guide 1.60 and soil-structure interaction
• Nonlinear soil structure interaction
Structural Response
• Elastic-plastic
• Damping values
• Conservative operation basis earthquake
• Statistical load combinations
• Three input components
• Structural resistance
• Three directions of earthquake motion
Mechanical System Response
• Elastic-plastic
• Mechanical resistance
• Peak widening of floor spectra
• Nonlinear structural analysis and floor response spectra
• Subsystem/major system interaction

the NRC.⁴ The major objective is developing mathematical models that realistically predict the probability of radioactive release from seismically induced events in nuclear power plants. Using these models, we will provide the NRC licensing staff with methods for determining inherent safety margins in the seismic design of nuclear power plants. This will allow the NRC to achieve an appropriate balance between construction costs, conservatism, and the complexity of necessary analytical design methods.

To achieve this, we will first develop a probabilistic-based system model (or models) to integrate the entire design methodology and an event fault tree characterization of the nuclear power plant. We will then perform systems analyses to calculate the probability of radioactive releases as a function of various earthquake source parameters, such as ground acceleration. The results of these analyses, in terms of release probabilities, will then be used as a measure for quantifying safety margins and improving design methodology.

To estimate the overall conservatism in the seismic design methodology, we will develop and perform system analyses on two different integrated system models. One model will represent the current methodology (e.g., as defined in the NRC Standard Review Plan). The other will establish a basis of comparison (or reference base) for judging the

results of the current seismic methodology. To accomplish this, we will replace elements of the currently used methodologies with the most recent advances in the state-of-the-art, which are less restrictive. In addition, because no method of analysis yet available is completely general, additional "ad hoc" improvements can be developed using several calculational methods and incorporating available experimental data.

All models developed will identify those structures, systems, and components for which some combination of their failures could lead to a release. The interface between the seismic methodology and the event-fault tree characterization will be curves or expressions which relate to the probability of failure of identified structures, systems, and components as a function of a seismic source parameter.

Phase I schedules, manpower requirements, and funding (\$3 million) have been established. This 20-month phase is currently underway. (For a more detailed description, see Ref. 4.) Seven tasks are being performed: (1) plant site selection, (2) seismic input characterization, (3) soil-structure interaction, (4) structural building response, (5) structural subsystem response, (6) fragility definitions, and (7) system analysis. Subsequent phases will be defined by the results of Phase I.

ANALYSIS AND TESTING OF CASKS FOR TRANSPORTATION OF SPENT NUCLEAR FUEL RODS

We are involved in diverse safety-related projects in applied mechanics for the Nuclear Regulatory Commission (NRC), including hydrodynamic testing and analysis, vibration analysis, transportation of hazardous materials, and effects related to tornadoes. An example of our effort in applied mechanics is our project on transportation of spent nuclear fuel rods.

Casks used for shipping fuel rods are massive, weighing up to 54,000 kg. Large casks, as the one shown in Fig. 20, are transported on railroads, and small ones are transported on trucks. We are developing improved methods for the NRC to use in their evaluation of such casks submitted for licensing. Current regulations specify a set and sequence of conditions that a container must undergo without excess leakage. These conditions include: impact on a very hard surface, attempted puncture, fire, and then water submersion. Loss of radioactive material can occur by leakage around the closure or seals, or by loss of shielding.

We are addressing the two critical areas of puncture and sealing in a combined experimental

and analytical program. The major objectives of this program are to develop new analytical methods to augment and perhaps replace our reliance upon experimental methods for the evaluation of puncture processes and seal performance.

Prior to our work, the evaluation of seal performance and structural resistance to puncture had always been done experimentally. No effective analysis techniques existed that could solve the geometric and material nonlinearities of contact surfaces, such as at a seal or punch interface. A major breakthrough was achieved by the Mechanical Engineering Department's Code Development Group with the introduction of a slide line methodology into the dynamic explicit finite element code DYNA2D. For the first time, dynamic nonlinear contact problems could be solved with finite element codes. However, the quasi-static nature of the forces associated with sealing or static puncture still could not be attempted with any degree of success. In response to our need to tackle these problems, the Code Development Group successfully implemented a slide line procedure into an implicit finite

Fig. 20. Rail cask for spent fuel transport.

element code that can solve both dynamic and static problems. This computer code, NSAP2D, and its successor, NIKE2D, are also more efficient and less costly to run than DYNA2D. Without these finite element codes as an analytical tool, the major objectives of this program could not have been achieved.

Puncture Analysis and Tests

Spent-fuel shipping casks use either lead, or depleted uranium, U_{238} , for radioactive shielding, sandwiched between sheets of stainless steel. Our objective was to test and analyze the puncture process of layered plates consisting of either lead or depleted uranium, sandwiched between two plates of stainless steel.

Previous experimental evaluation of puncture has been based on an energy criteria. Static tests were made, and the integral of the force multiplied by the distance determined the energy required to puncture the plates. For dynamic puncture, the energy concept was extended by determining the dynamic energy required to puncture a plate. Multiplying the weight of a plate by the drop height required for puncture gives the energy required for puncture. In general, more energy is required to puncture a plate or layered plates under dynamic conditions than under static conditions. More energy is absorbed by structural vibrations, and by the punch during a dynamic drop. However, the energy required depends on the structure being tested and does not appear to be the best puncture criteria.

In our work, we analyzed the structure being tested, predicted its response, and selected a satisfactory puncture or failure criteria that could be applied to any structure. Figure 21 shows one of the layered plates that was analyzed and tested. First, the puncture conditions were determined by tests. Second, the analysis of the same configuration was examined to confirm the agreement between the analysis and the test. Third, the stress conditions at failure, as computed by the computer code, were studied and compared with selected failure criteria. We performed this process for both static and dynamic puncture conditions.

Static Tests and Analysis. For the static puncture conditions, excellent agreement was obtained between the measured test parameters and the values predicted by the computer analysis. See Fig. 22. We then needed to find a unifying puncture criteria to form a common base for determining when a plate would fail. Based on the analysis, we found that as the punch load increases the intensity of the shear stress on the failure surface increases. Failure consistently occurs when the critical shear stress contour reaches the opposite side of the lower plate. The value of this shear stress is equal to 0.6 times the ultimate tensile stress. This shear failure criteria provides the unifying conditions that allow the puncture event to be accurately predicted for the lead-backed plates. Figure 22 shows the comparison between the predicted static puncture conditions and the experimentally measured puncture conditions for lead-backed plates.

Fig. 21. Deformed shape of a 1.37-cm-thick stainless steel plate at its experimentally determined failure load.

For plates where uranium replaced the lead, the test results are different. Since lead provides little or no structural strength, the lower stainless steel plate must carry the load. The depleted uranium plate carries a significant structural load. When the strain in the outer fiber of the uranium plate exceeds approximately 6% to 7%, it fails. Sufficient load has been developed in the punch at this point that the shear failure criteria in the lower stainless steel plate is exceeded and puncture occurs.

Dynamic Tests and Analysis. The static tests were a prelude to more extensive dynamic testing. The dynamic tests and analyses are being completed, but initial agreement between test results and the computer predicted response is good. Preliminary results show that the puncture during dynamic impact is also governed by a shear failure criteria for the lead-backed plates. No dynamic calculations have been made yet for the uranium-backed plates, but a similar failure criteria to that found during the static tests is expected.

Seal Test and Analysis

A comprehensive program is underway to analytically and experimentally investigate the leak rate of five seal-flange configurations. The flanges are one-quarter and one-half the size of those used in spent-fuel railroad casks. This corresponds to 25.4- and 50.8-cm-diam seals. Each seal configuration is shown in Fig. 23. These are: (1) Grayloc, (2) Conoseal, (3) polymer O-ring, (4) hollow metal O-ring, and (5) Batzer. Each seal uses a different principal to achieve a high seal-flange interface pressure. Although these designs are different, our computer code NIKE2D, can calculate the key interface pressures. This pressure is essential for good sealing and low leak rates. In addition, each seal has been used in applications where a leak rate less than 10^{-5} atm cc/s has been achieved. However, scale effects, temperature effects, and large differential pressures across the seal may significantly affect the leak tightness of some of these seals. Our test program is designed to measure these effects. Table 4 illustrates the test matrix that is being used with each of the seal-flange configurations for both the 25.4 and 50.8-cm-diam flanges.

Each seal was designed with trial and error techniques while following general design principles. A leak rate model governing permeation-type leaks (not orifice leaks) has been proposed by Roth in his book on vacuum technology⁵ and is shown in graphical form in Fig. 24. The leak rate is plotted for helium tracer gas at 25°C with 1-atm pressure differential across the seal. Other significant parameters that are fixed for this curve are the length of the contact

surface (L), the width of the contact area (W), a measure of the seal hardness (R), and a measure of the surface roughness (A). For these fixed but significant parameters, it is obvious the leak rate is a strong function of the seal-flange interface pressure (P).

Our finite element codes with slide surfaces can calculate this seal-flange interface pressure, and the contact surface width. The behavior of each seal when it comes into contact with a sealing surface represents a nonlinear contact problem. An example of the computer calculated stress and deformation is shown in Fig. 25. These computer calculations, in conjunction with leak rate models similar to those in Fig. 24, are used to estimate the expected leak rate for a particular seal. The calculated leak rate is checked against the rate measured in the laboratory to verify and refine the analytical techniques. These analytical tools should provide an effective means to estimate the range of leak rates that could be achieved by any new seal design. In addition, our tests (Table 4) will provide hard data on leak rates on large seals under realistic operating conditions.

Summary

Our major objective has been to improve structural analysis techniques for shipping container components. In particular, we are concerned with the two major mechanisms associated with radioactive

Fig. 22. Calculated and experimental puncture force required to penetrate lead-backed stainless steel plates.

(a) Grayloc seal fixture,
made-up position.

(b) Conoseal and
fixture.

O-ring installed

O-ring under pressure

(c) Polymer O-ring under pressure.

(d) Hollow metal O-ring with a 0.25-mm-silver coating.

(e) Batzer flange and seal.

Fig. 23. Various kinds of seals.

Fig. 24. Leak rate as a function of seal-flange contact stress for a helium tracer gas at 25°C. Values, which are defined more fully in the text, are: $\Delta P = 1 \text{ atm}$, $R = 500 \text{ kg/cm}^2$, $L = 10 \text{ cm}$, $W = 0.2 \text{ cm}$, $A = 1 \times 10^{-4} \text{ cm}$, and one interface.

leakage, loss of sealing and loss of shielding from puncture. A recent development in finite element analysis, slide surfaces, achieved by the Mechanical Engineering Code Development Group, has enabled us to attack these two nonlinear contact problems for the first time with computer codes. Our comparisons to date with laboratory tests have shown excellent agreement and offer a great deal of promise for the use of these techniques in the future. The techniques open up new analysis avenues that can augment and potentially replace expensive tests as a method for evaluating the structural integrity of spent-fuel shipping casks.

Fig. 25. Computer-predicted contact stress in a silver-coated hollow metal O-ring in a compressed position.

Table 4. Test matrix for seal leak rate tests.

Test number	Bolt force (F=specification force)					100	Internal pressure (kPa)					Temperature (°C)					Seal diam (cm)		
							1390	2758	4135	5515	6895	-40	21	38	118	204	282	25.4	50.8
	0.5F	0.75F	1.0F	1.1F	1.1F														
1	X					X							X					X	X
2		X				X							X					X	X
3			X			X							X					X	X
4				X		X							X					X	X
5	X					X							X					X	X
6		X				X							X					X	X
7			X			X							X					X	X
8				X		X							X					X	X
9	X					X		X					X					X	X
10		X				X		X					X					X	X
11			X			X		X					X					X	X
12					X			X					X					X	X
13	X								X				X					X	X
14		X							X				X					X	X
15			X						X				X					X	X
16									X				X					X	X
17	X									X			X					X	X
18		X								X			X					X	X
19			X							X			X					X	X
20				X						X			X					X	X
21	X									X			X					X	X
22		X								X			X					X	X
23			X							X			X					X	X
24				X						X			X					X	X
25		X								X			X					X	X
26		X	X							X			X					X	X
27		X	X	X						X			X					X	X
28		X	X	X						X			X					X	X
29			X							X			X					X	X

MATERIAL CONTROL STUDY: A DIRECTED GRAPH AND FAULT-TREE PROCEDURE FOR ADVERSARY EVENT SET GENERATION

Introduction

The Lawrence Livermore Laboratory is conducting a Material Control and Accounting Study for the Nuclear Regulatory Commission (NRC), Office of Nuclear Regulatory Research. As part of their duties, the NRC is responsible for the licensing of new nuclear facilities. Since the safeguarding of nuclear materials has become increasingly important in recent years, the NRC must be able to systematically evaluate the material control systems of proposed nuclear facilities and to guarantee their effectiveness to the public. Each facility has a material control system to protect against the theft of special nuclear material, such as plutonium and uranium 235. In the two-year-old study, the Laboratory has been developing an assessment procedure to evaluate the effectiveness of a potential nuclear licensee's material control system.⁶

The assessment procedure, shown in the block diagram in Fig. 26, draws upon two types of data: license applicant information and the NRC/LLL data base. Applicant data include the plan of the

facility physical plant, operational procedures, descriptions of special nuclear material processing, and the details of the material control and accounting system. The NRC/LLL data base will contain the mathematical models (such as models of the performance of the special nuclear material detection monitors) necessary to evaluate an applicant's submittal.

The first step in the assessment procedure is to identify targets within the facility that contain theft-attractive special nuclear material. The second step is to determine the adversary actions and conditions of the material control system that could allow successful diversion of special nuclear material, that is, generate the adversary event sets. Simulation is required for those adversary event sets where timeliness and ordering of events is important for successful diversion. The qualitative and quantitative analysis of the event sets and the simulation results allow the effectiveness of the material control system to be determined.

Fig. 26. The LLL assessment procedure.

Adversary Event Set Generation Procedure

The key in the LLL assessment procedure for evaluating the effectiveness of a material control system is the generation and analysis of adversary event sets. We have developed a procedure based on a directed graph (digraph) and fault tree methodology by which the event sets can be generated and analyzed. This methodology has been used by Japp and Powers⁷ to assess the safety of chemical processing systems. We extended this methodology to model intentional diversionary or malevolent acts by an adversary so that these acts appear in the event sets.

The procedure for the generation and analysis of the event sets is described below, in detail, step by step, as delineated by the block diagram in Fig. 27.

General System Schematic. The first step in the procedure (Fig. 27) is the formulation of a general schematic for system modeling. Information from piping and instrumentation diagrams, the physical plant layout, and material control related procedures

is used to formulate the schematic for system modeling. The general schematic delineates the unit model digraphs needed to model the system and the overall system interactions. The unit models include models of adversary movement in the facility, monitors, process equipment, and procedures.

Unit Model Digraphs. In step 2 (Fig. 27), unit model digraphs, the basic building blocks of the procedure, are generated. As described in Refs. 7 and 8, digraphs are functional cause-and-effect network models that describe the relationship between various system variables and the conditions that are necessary for these relationships to exist. In addition, digraphs can show events such as adversary actions that may nullify or change the relationships between variables. Digraphs are useful since they are multivalued network models and they can readily model the dynamics of the relationships between variables. The advantage of generating unit model digraphs is that a separate

Fig. 27. Procedure for generation and analysis of adversary event sets.

analysis can be performed on system components without performing an entire systems analysis. These unit models are analogous to mini-fault trees described by Fussell et al.⁹

System Digraph. The third step in Fig. 27 is the generation of the system digraph, which is constructed from the unit model digraphs for a selected top event variable (the top event is the variable being modeled). The system digraph is obtained by deductively following the information flow given in the general system schematic. The material control system is modeled as a control system designed to counter the actions of the adversary. All the potential ways the material control system may respond to prevent special nuclear material theft are modeled in terms of "adversary cancellation loops" on the system digraph. These loops are similar in concept to the negative feedback and negative feedforward loops designed to cancel disturbances in process variables.

System Fault Tree. In the fourth step (Fig. 27), the system fault tree is generated from the system digraph via a synthesis algorithm. The top event in the fault tree corresponds to a disturbance in the top event variable of the system digraph. The top event variable for the material control study is M_{DIV} defined by

$$M_{DIV} = \begin{cases} +1 & \text{if successful diversion of} \\ & \text{special nuclear material occurs} \\ 0 & \text{otherwise} \end{cases}$$

A zero value for a variable on the system digraph corresponds to a true or expected value. Any other value, hence, corresponds to a deviation or disturbance. The top event in the system fault tree for the material control study is $M_{DIV} = +1$. All loops in the system digraph that model the corrective actions of the material control system must fail for a disturbance in the top event variable to exist.

For successful diversion of special nuclear material to occur, all adversary cancellation loops must fail. These loops fail as the result of:

- random monitor failure
- inadequate monitor measurement sensitivity
- human error, including slow guard response
- adversary activity, including equipment

tampering and collusion.

The synthesis algorithm creates an AND logic gate in the fault tree each time a cancellation loop in the system digraph fails.

Once generated, the fault tree can be evaluated qualitatively and quantitatively to assess the vulnerabilities of the safeguard system.

Qualitative Analysis. The qualitative analysis of the fault tree provides much valuable information without using numerical data. It includes performing Boolean manipulations of the basic events, generating the adversary event sets, structurally ranking the basic events, determining the collusion requirements, and evaluating the effect of power loss on the material control system.

A structural ranking of the basic events in the event sets helps to identify important basic events for further analysis. This type of ranking is a function of the number of event sets in which a basic event appears in and the relative length of those event sets.

Common cause analysis is used to determine the collusion requirements (how many and who) and the effects of power loss of key components of the material control system for successful special nuclear material theft. In addition, a vital location analysis can be performed to determine the locations which must be visited for successful tampering.

The computer codes Fault Tree Analysis Program (FTAP)¹⁰ and the Set Equation Transformation System (SETS),¹¹ designed to generate and handle numerous, high-ordered minimal cut sets, are used to perform the qualitative analysis.

Quantitative Analysis. To further identify the weaknesses of the material control system, a quantitative analysis is performed. This analysis assesses the impact of material control system components with various failure rates and detection probabilities, the effect of maintenance policies, and the ease with which component tampering can occur. The IMPORTANCE computer code¹² is used to perform the quantitative assessment.

Inputs required for the quantitative analysis are a listing of all event sets, probability data for the basic events, and the assumption of statistical independence of the basic events.

The probability of successful theft of special nuclear material can be calculated for four specific cases:

- (1) No material control system tampering, no alarm signal generated.
- (2) No material control system tampering, slow safeguards response.
- (3) Material control system tampering, no alarm signal generated.
- (4) Material control system tampering, slow safeguards response.

A sensitivity analysis of the probability of successful theft for the above cases as a function of the amount of special nuclear material stolen is also done. Quantities of special nuclear material investigated are 0.5 g, 200 g, and 5 kg. The maximum expected performance of the material control system

occurs when there is no system tampering. However, clever adversaries may tamper with the material control system to render it ineffective. In the tampering analysis, the following adversary attributes and material control system characteristics are considered:

- Type of tools and resources required for tampering
- Accessibility of components to potential adversaries
- Monitoring of equipment for tampering
- Availability of tools and resources required for tampering
- Personnel required for tampering.

The probability of successful tampering is then a function of the probability that each of the above can occur with either no or slow material control system response.

Demonstration of the Event Set Generation Procedure

The event set generation and analysis procedure has been applied in the assessment of the material control system in a prototype nuclear facility, the Test Bed.¹³

The Test Bed is based upon the plutonium nitrate storage area of the AGNS facility in Barnwell, S.C., but with substantial modifications. These modifications are added to further develop and test the assessment procedure and are not criticisms or "fixes" to the AGNS current design. The modifications include the addition of check valves, limit switches on valves, a computer controlled access system, computerized material control and accounting and procedure monitoring logic, and other safeguards components.

The general form of the system digraph for the Test Bed is shown in Fig. 28. The arrows represent information flow with regard to (1) movement of special nuclear material and people and (2) the material control system response which acts to prevent the theft of special nuclear material. The initial conditions of the Test Bed assessment are shown on the left of Fig. 28.

The fault tree generated from the system digraph by the synthesis algorithm contained 125 gate events and 113 basic events. The qualitative analysis of the fault tree generated 814, 042 event sets for a the case of a slow safeguards response and 4,736 event sets for the case of no safeguards response.

The collusion analysis established that successful diversion can occur only if three particular plant personnel are in collusion or two persons if random failures occur.

The IMPORTANCE computer code determined the probability of each basic event in contributing to the probability of successful diversion. The ranking of these basic events found the following vulnerable points in the Test Bed:

- computer hardware and software
- remote control panel
- crash door alarms
- maintenance policies

Although the Test Bed is a facility with an automated and sophisticated material control system, the assessment has found several basic weaknesses. A similar assessment can determine the effect of strengthening the aforementioned areas. Thus, the Test Bed demonstration has shown the directed graph-fault tree procedure (Fig. 26) is not only an effective assessment tool but also a valuable design tool. The procedure can also be used for safety and reliability analysis.

Future Activities

In the future, the adversary event set generation procedure described in this article will be used to assess the effectiveness of an existing nuclear facility handling special nuclear material. In addition, the assessment procedure will be automated as much as possible. Fault tree computer codes FTAP, SETS, and IMPORTANCE currently exist to qualitatively and quantitatively evaluate fault trees. Computer codes to perform the front end of the procedure, i.e., the manipulation, storage and generation of unit model digraphs, system digraphs, and system fault trees, are currently in the developmental stages.

Fig. 28. General form of system digraph for Test Bed.

PUBLICATION ABSTRACTS

P.F. Durbin, *Radiographic Inspections at the Calipatria Geothermal Test Site*, Lawrence Livermore Laboratory, Rept. UCID-17734 (1978).

The Lawrence Livermore Laboratory (LLL) Nondestructive Evaluation (NDE) section of the Materials Engineering Division is making radiographic inspections to estimate the extent of corrosion and scale buildup at the Geothermal Test Site in Calipatria, California. This study, begun in early 1976, is continuing. The NDE section developed the radiation exposure techniques using ^{60}Co and ^{192}Ir isotopes; Hazards Control and NDE established radiography safety procedures. Five radiographic evaluations were made of the Geothermal Test Site from May 1976 to February 1977. Estimations of scale buildup from radiographs of operating plant pipes, valves, and tanks correlated closely with our actual scale measurements from used plant sections.

R.H. Toland, R.J. Sanchez, and D. Freeman, *Stress-Rupture Life of Kevlar/Epoxy Spherical Pressure Vessels*, Lawrence Livermore Laboratory, Rept. UCID-17755, Part 1 (1977).

We are conducting a study of the life of Kevlar/epoxy spherical pressure vessels subjected to a constant, sustained pressure in time until vessel failure; i.e., stress-rupture. This study has two objectives: to generate baseline data on the stress-rupture life of Kevlar/epoxy pressure vessels, and to apply statistical reliability theory to vessel and strand data to define vessel life statistics and reliability at given pressure (stress) levels. This report, the first of three, describes vessel design, fabrication, and static burst performance, as well as the stress-rupture test plan and test facility.

M.J. Burger, *Document Control Program (DCP)*, Lawrence Livermore Laboratory, Rept. UCID-17746 (1978).

D.F. Arthur, R.G. Dong, R.C. Murray, T.A. Nelson, P.D. Smith, and L.H. Wight, *Evaluation of Methods for Seismic Analysis of Nuclear Fuel Reprocessing and Fabrication Facilities*, Lawrence Livermore Laboratory, Rept. UCRL-52159 (1978).

This is the third and final report in a series aimed at evaluating methods of seismic analysis for critical structures and equipment in nuclear fuel reprocessing plants (NFRPs) and mixed oxide fuel fabrication plants (MOFFPs). The series was requested

and the work funded by the Nuclear Regulatory Commission's Office of Standards Development. Our purpose in these reports is to provide the NCR with a technical basis for assessing seismic analysis methods and for writing regulatory guides in which methods ensuring the safe design of nuclear fuel cycle facilities are recommended. The present report evaluates methods of analyzing buried pipes and wells, sloshing effects in large pools, earth dams, multiply supported equipment, pile foundations, and soil-structure interactions.

E.K. Collins and W. Lai, *Final Air Test Results for the 1/5-Scale Mark I Boiling Water Reactor Pressure Suppression Experiment*, Lawrence Livermore Laboratory, Rept. UCRL-52371 (1978).

A loss-of-coolant accident (LOCA) in a boiling-water reactor (BWR) power plant has never occurred. However, because this type of accident is particularly severe, it is used as a principal basis for design. During a hypothetical LOCA in a Mark I BWR, air followed by steam is injected from a drywell into a toroidal wetwell about half-filled with water. Injection occurs below the water surface in a vertically downward direction through open-end pipes called downcomers. Energy from the reactor is dissipated as the steam mixes with the water and condenses.

During the LOCA, a downward load is induced on the toroidal wetwell as water that is initially inside the downcomers is forced into the water pool below them. The magnitude of this downward load increases until just after gas bubbles appear in the pool. The water surface inside the toroidal wetwell then moves upward, particularly in the region adjacent to the outside of the downcomers, as water is displaced by the gas bubbles. The gas bubbles over-expand, because the flow of gases from the downcomers is not sufficient to counteract the upward displacement of the water. The direction of the net load on the toroidal wetwell then shifts upward and reaches a peak of approximately half the magnitude of the maximum download. The vertical load then decreases to static conditions.

Sponsored by the United States Nuclear Regulatory Commission, a complete three-dimensional facility was designed and constructed. A series of consistent, versatile, and accurate air-water tests simulating LOCA conditions was completed in the Lawrence Livermore Laboratory 1/5-Scale Mark I BWR Pressure Suppression Experimental Facility. Results from this test series were used to quantify the vertical loading function and to study the

associated fluid dynamic phenomena. Detailed histories of vertical loads on the wetwell are shown. In particular, variations of hydrodynamic-generated vertical loads with changes in drywell pressurization rate, downcomer submergence, and the vent-line loss coefficient are established. Initial drywell overpressure, which partially preclears the downcomers of water, substantially reduces the peak vertical loads.

Scaling relationships, developed from dimensional analysis and verified by bench-top experiments, allow the 1/5-scale results to be applied to a full-scale BWR power plant. This analysis leads to dimensionless groupings which are invariant. These groupings show that if water is used as the working fluid, the magnitude of the forces in a scaled facility is reduced by the cube of the scale factor; the time when these forces occur is reduced by the square root of the scale factor.

H.G. Patton, I.F. Stowers, W.A. Jones, and D.E. Wentworth, *Status Report on Cleaning and Maintaining Laser Disk Amplifiers*, Lawrence Livermore Laboratory, Rept. UCRL-52412 (1978).

This report describes the activities and advances in engineering and technology made by the Solid State Mechanical Maintenance Group within the laser program at Lawrence Livermore Laboratory. It includes design and operation of clean rooms, contamination control on optical surfaces, cleaning techniques, and glass damage mechanisms. This report, a much expanded version of a section in the laser program's 1976 annual report, covers work accomplished from July 1976 to April 1977. It has been used as the source for five papers presented at various national conferences.

G.E. Cummings, *Dynamic Response of a Cylindrical Shell Immersed in a Potential Fluid*, Lawrence Livermore Laboratory, Rept. UCRL-52464 (1978).

A numerical solution technique is presented for determining the dynamic response of a thin, elastic, circular, cylindrical shell of constant wall thickness and density, immersed in a potential fluid. The shell may be excited by an arbitrary radial forcing function with a specified time history and spatial distribution. In addition, a pressure history may be specified over a segment of the fluid outer boundary. Any of the natural shell end conditions may be prescribed.

A numerical instability prevented direct solutions where the ratio of the hydrodynamic forces to

shell inertial forces is greater than two. This instability is believed to be the result of the weak coupling between the equations describing the fluid to those describing the shell. To circumvent this instability, an effective mass was calculated and added to the shell.

Comparison of numerical to experimental results are made using a 1/12 scale model of a nuclear reactor core support barrel. Natural frequencies and modes are determined for this model in air, water, and oil. The computed frequencies compare to experimental results to within 15%.

We illustrate the use of this numerical technique by comparing it to an analytical solution for shell beam modes and resolve an uncertainty in the analytical technique concerning the proper effective mass to use.

W.M. Shay, W.G. Brough, and T.B. Miller, *Instrumenting a Pressure Suppression Experiment for a Mark I Boiling-Water Reactor--Another Measurements Engineering Challenge*, Lawrence Livermore Laboratory, Rept. UCRL-80586 (1978). Submitted to the Spring Meeting of the Society for Experimental Stress Analysis, Wichita, Kansas, May 14-19, 1978.

A 1/5-scale test facility of a pressure-suppression system from a Mark I boiling water reactor was instrumented with seven types of transducers to obtain high-accuracy, dynamic loading data during a hypothetical loss-of-coolant accident. A total of 27 air tests have been completed with an average of 175 transducers recorded for each test. An end-to-end calibration of the total measurement system was run to establish accuracy of the data. The instrumentation verified the analysis of the dynamic loading of the pressure-suppression system.

T.T. Chiao, *Some Interesting Mechanical Behaviors of Fiber Composite Materials*, Lawrence Livermore Laboratory, Rept. UCRL-80908 (1978). To be published in the Proceedings of the US-USSR Seminar on Fracture of Composite Materials, Riga, USSR, September 4-7, 1978.

In the course of our studies of the mechanical behaviors of aramid/epoxy and the high-strength S2-glass/epoxy composites, we have observed two interesting but puzzling behaviors. First, when a sustained tensile load is removed any time before stress-rupture failure, the composites retain almost their full original strengths. Second, the contribution of a high-elongation, low-modulus matrix to the performance of a simple glass composite seems to

be surprisingly little. These results have raised questions about the validity of the strength retention test as well as the usefulness of a low-modulus matrix in fiber composites.

M.A. Hamstad, *Nondestructive Testing of Fiber Composites: Application of Acoustic Emission Inspection*, Lawrence Livermore Laboratory, Rept. UCRL-81004 (1978). To be published in The Proceedings of the Fourth Composites in Manufacturing Conference, the Society of Manufacturing Engineers, Los Angeles, California June 7-9, 1978.

Valid nondestructive test results must be quantitatively correlated to the required engineering performance properties of the fiber composite article. We discuss the relationship of the currently unknown defect structure in fiber composite articles with respect to nondestructive testing. Examples of quantitative nondestructive testing of fiber composites are cited from the open literature, including acoustic emission inspection and ultrasonic testing. The unique application of acoustic emission inspection to the location and determination of real flaws in fiber composites is discussed in detail.

B.G. Monahan, V.P. Gede, P.R. Landon, C.A. Colmenares, and T.J. Biel, *Materials Compatibility Study of 316 Stainless Steel at the LLL Tritium Facility*, Lawrence Livermore Laboratory, Rept. UCRL-81028 (1978). To be distributed at the American Nuclear Society Meeting, Santa Fe, New Mexico, May 8-12, 1978.

We are conducting stress corrosion tests on 316 VIM/VAR stainless steel as part of the Tritium Facility's Quality Assurance Program. Initial results under severe conditions indicate earlier-than-expected failure. We are expanding our tests to further examine this problem.

W.L. Server, D.M. Norris, Jr., and M.E. Prado, *Ductile Crack Initiation in the Charpy V-Notch Test*, Lawrence Livermore Laboratory, Rept. UCRL-81034 (1978). To be published in the ASM Proceedings of the 107th AIME Meeting, February 28, 1978, Denver, Co.

Initiation and growth of a crack in the Charpy V-notch test was investigated by performing both static and impact controlled deflection tests. Test specimens were deformed to various deflections, heat-tinted to mark crack extension and broken apart at low temperature to allow extension measurements. Measurement of the crack extension provided an estimate of crack initiation as defined by different criteria. Crack initiation starts well before maximum load, and is dependent on the definition of "initiation". Using a definition of first micro-initiation away from the ductile blunting, computer model predictions agreed favorably with the experimental results.

N.J. Brown and R.T. Maney, *Optical Fabrication of Fluorophosphate Laser Glasses*, Lawrence Livermore Laboratory, Rept. UCRL-81083 (1978). Submitted to the American Ceramic Society Annual Meeting, Detroit, Michigan, May 10, 1978.

This paper discusses the grinding and polishing of a new family of fluorophosphate laser glasses.

C.A. Calder and W.W. Wilcox, *Acoustic Velocity Measurements Across the Diameter of a Liquid Metal Column*, Lawrence Livermore Laboratory, Rept. UCRL-81124 (1978). Presented at the SESA 1978 Spring Meeting, Wichita, Kansas, May 15-19, 1978.

A method is described for the measurement of the acoustic velocity across the diameter of a 1 mm diam liquid metal column before instability and breakup of the column occurs.

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